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REVIEW OF THE STATUS OF CRBR LICENSING TECHNICAL ISSUES RELATED TO HEAT REMOVAL SYSTEM AND SEVERE ACCIDENT ANALYSIS

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NOTICE

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REVIEW OF THE STATUS OF CRBR LICENSING TECHNICAL ISSUES RELATED TO HEAT REMOVAL SYSTEM AND SEVERE ACCIDENT ANALYSIS

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ABSTRACT

A review is presented of the status of licensing technical issues related to heat removal system and severe accident analysis for the Clinch River Breeder Reactor. Discussions are presented on the following topics: operational transients; natural circulation heat removal and the overall capability of the shutdown heat removal system; containment thermal analysis associated with design basis sodium spills and with core-disruptive accidents; reliability and risk analysis; loss-of-heat-sink accident progression with scram; reactor physics issues related to the heterogeneous core design; transition phase of the core-disruptive accident; structural analysis. This work is part of the initial phase of the re-initiation of the licensing review of the Clinch River Breeder Reactor.

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1. INTRODUCTION

1.1 Background

The formal licensing review of the Clinch River Breeder Reactor (CRBR) began in 1975, when the Preliminary Safety Analysis Report was submitted as part of the application for a construction permit to the Nuclear Regulatory Commission (NRC). This application differed from those presented for commercial light water reactors in several ways, which were a direct result of the rather different design, i.e., sodium coolant, highly enriched fuel, breeding blanket, etc. However, perhaps the most interesting aspect of this application was the emphasis placed on reliability analysis in the development of the safety-related design considerations for the plant.

During the licensing review, several technical issues received much attention by the applicant and the NRC. These include, but are not limited to: a) the impact of core disruptive accidents on the reactor vessel and on the containment building, b) the redundancy and diversity of the shutdown heat removal system, c) the capability of the plant to remove decay and sensible heat by natural circulation, and d) the selection of a design basis accident for containment.

In the Spring of 1977, President Carter put forth an energy and non-proliferation policy which did not include plans for completing the CRBR. Shortly thereafter, the Energy Research and Development Agency (ERDA) requested of NRC an indefinite postponement of the hearings associated with Limited Work Authorization. As a result, NRC terminated its formal review of the application.

Notwithstanding the foregoing, funding for the CRBR Project continued. During the past four years, analyses were performed, the design was modified (particularly the reactor core), and major components were fabricated.

The position of the present administration is to complete the CRBR. In order to resume the licensing review of the CRBR, NRC and its consultants must participate in at least two processes: 1) refamiliarize themselves with the safety issues that were associated with the licensing review at the time that it was suspended, and 2) assess the impact of any technological advances of new safety issues that emerged during the past four years.

The purpose of this report is to initiate these processes. The general focus of this report is on the systems and phenomena related to decay heat removal and severe accident analysis. The draft version of this report was submitted to NRC in November 1981. There have been no major changes in the content of the final version of this report. This report was written for interim use only and it should not be regarded as an evaluation of the current design of the CRBR.

1.2 The Role of Brookhaven National Laboratory in the Earlier Phase of the Licensing Review

Brookhaven National Laboratory provided technical assistance to the regulatory staff since the inception of the CRBR licensing review in 1975. The program contained a diverse range of technical topics which included: core disruptive accident analysis (loss-of-flow, transient overpower, transition phase, fuel pin failure dynamics, and loss of heat sink with scram); plant transient behavior (core thermal response to pipe ruptures, pump seizures, and natural circulation events); post accident heat removal phenomena (in-vessel fuel debris relocation, ex-vessel debris cooling, sodium fires, hydrogen combustion, and containment thermal response); reactor physics analysis of disrupted core configurations. Many of the activities in these areas were also carried out during the period 1977-1979 in support of regulatory staff's safety review related to the operation of the Fast Flux Test Facility.

Over the same period of years BNL was active in the development of the SSC code and in simulation experiments related to disrupted core accidents. In addition, analyses were performed on LMFBR piping integrity and on sodium materials behavior.

1.3 Scope of this Report

This report identifies and discusses a broad range of technical issues related to the licensing of the CRBR. It does not include (except for a brief discussion of the transition phase) a discussion of issues related to unprotected transients such as the loss-of-flow and the transient overpower scenarios. A discussion of these transients are the main focus of a parallel effort being conducted by Los Alamos National Laboratory. Several of the issues identified in this report do, however, relate directly to unprotected transients.

Section 2 contains a discussion of operational transients that are generally considered in Chapter 15 of a Preliminary Safety Analysis Report. Section 3 addresses the issues associated with natural circulation heat removal and the overall capabilities of the CRBR shutdown heat removal system. Section 4 focuses on containment thermal analysis associated with design basis sodium spills and with core disruptive accidents. Reliability and risk analysis is

discussed in Section 5. In Section 6, a discussion of the loss-of-heat-sink accident, with scram, is provided. Section 7 contains a discussion of the reactor physics issues related to the new heterogeneous core design. Section 8 adresses issues related to the transition phase of the core disruptive accident. A discussion of structural analysis issues is presented in Section 9. Finally, a summary of this report in connection with the relevent technical issues for CRBR licensing are presented in Section 10.

2. OPERATIONAL TRANSIENTS

For the purpose of this report, the term "operational transients" is intended to include all events that are likely to occur at least once during the operating lifetime of the the CRBR plant. These transients are considered in design basis accidents, and no specific quantification of their likelihood of occurrence is given. In all such incidences, credit is taken for proper operation of the Plant Protection System (PPS). Thus we are not concerned, in this section, with the core disruptive accident scenarios. The likelihood of occurrence of an incidence due to failure of a component or a group of components is discussed in Section 5. This section provides an assessment of the safety analysis contained in Chapter 15 of the CRBRP Preliminary Safety Analysis Report (PSAR).

2.1 Objectives and Issues

The objectives of this section are:

(a) to assess the safety analysis as reported in Chapter 15 of the CRBRP PSAR,

(b) to ascertain that all significant accident scenarios are considered,

(c) to provide a list of subtask areas where work is needed, and

(d) to ascertain that the classification of events and the design criteria used are consistent.

There are many issues which need to be resolved. Most of the concerns noted in this section stem from the following:

(a) Change in Core Design - The current design of CRBRP reactor core uses heterogeneous core, as opposed to the earlier design involving homogeneous core (Refs. 2.1,2.2). This change causes substantial variation in key reactivity feedback parameters. Computations of these parameters are more complex and subject to larger variations/uncertainties than for their counterparts in homogeneous core design.

(b) Control Rod Worths - The number of control rods used in the current design is reduced from a total of 19 to 15. The computation of the worth and margin of a control rod for the heterogeneous core is also more complex because such a computation must be done using space-time coupled method.

(c) Thermal-Hydraulic Analysis - Most of the PSAR analyses for operational transients are done by using either the DEMO-Rev4 (Ref. 2.3) or FORE-2M (Ref. 2.4) computer codes, or both. The DEMO-Rev4 code has been known to have several major deficiencies (Ref. 2.5). Perhaps the most significant one, as it relates to the analysis of undercooling transients, is the usage of pure transport time-delay model in the piping (Ref. 2.6). The CRBR Project has been aware of many of these shortcomings in the DEMO-Rev4 code -- they intend to use DEMO-Rev5 (Ref. 2.7). The FORE-2M code (Ref. 2.4) is used for most reactivity insertion events. A verification of this code is needed. It is not clear, however, as to what extent new analyses will be performed. In any case, there is a clear need to audit all thermal-hydraulic calculations.

(d) Multiple Fault Events - The CRBRP Chapter 15 analyses are based on the consideration of a single failure at a time. An evaluation is then made to ascertain the consequences of each single failure. In view of the TMI-2 incidence, additional effort is needed to examine the consequences of secondary failures or complicating features so that the significance of multiple failures could be addressed.

(e) PPS/PCS Design - The change in the reactor core design, as noted above, can have significant impact on the Plant Protection System (PPS) and the Plant Control System (PCS) designs and their settings. It is not clear whether the impact of the new core has been assessed on PPS and PCS. Furthermore, any interdependency of these two systems needs to be assessed.

(f) Rod Design Criteria - The CRBRP rod (fuel and blanket) design criteria are based on evaluation of the Cumulative Damage Function (CDF), which is at variance with the temperature limits used in LWRs. There is a lack of sufficient data base. It should also be noted that even for an anticipated event of control assembly withdrawal at power, a significant amount of fuel melting in a hot fuel channel is noted in Section 15.2.1.2 of the CRBRP PSAR. Since the peak linear power in the hot blanket channel is more than 20% higher than the hot fuel channel, a significant amount of blanket fuel may also be molten for this anticipated transient. The acceptability of molten fuel either in a fuel or blanket rod must be assessed. The CDF criteria need close scrutiny.

(g) Shutdown Heat Removal - The long term consequences of many of the operational transients place a reliance on the shutdown heat removal system. The issues connected with the SHRs are discussed in Section 3.

A brief list of major issues is given in Table 2.1. Short discussions are provided in subsequent subsections.

2.2 Reactor Physics Analysis

An essential element of the CRBRP performance evaluation is a verified data base for key reactor physics parameters such as the reactivity feedback coefficients, power coefficient, the control rod worths and the available control rod margins. Because of the sensitivity of these parameters to burnup of fissile fuel and breeding of fertile fuel, this data base is needed for the entire fuel cycle from the beginning-of-cycle (BOC), to the equilibrium cycle, to the end-of-cycle (EOC). The need for this data base has increased because of the change from the homogeneous core to the heterogeneous core design in the CRBRP. Another complicating factor is that the criticality calculations for the heterogeneous core are more complex (and subject to larger variations between the computed and measured ratio) than those for the homogeneous core.

The reactor physics data base is also needed for thermal-hydraulic performance evaluation for essentially intact reactor core as well as damaged or degraded core conditions. In this section, required parameters for the intact core conditions are discussed. Reactor physics analysis for the damaged core conditions is discussed in Sections 6 and 7.

2.2.1 Control Rod Worth

The CRBRP heterogeneous core differs from the previous homogeneous core in many ways. Figure 2.1 is a schematic of these two layouts (Refs. 2.1,2.2). The heterogeneous core uses 32.8% enriched fuel (Pu/(U+Pu)), as opposed to 17.4 and 25.1% enrichment in the homogeneous core. The fissile inventory is also significantly larger (10 to 25%). At the same time, the CRBR Project stated that the control rod worth requirement is considerably lower for the heterogeneous core ($17.44 primary for BOC4 as opposed to $26.53 primary for the equilibrium cycle of the homogeneous core). The computation of the control rod worth and the available margin is a complex problem, as it requires two-space dimensions coupled with time kinetics computer codes (see Section 7). An independent assessment is needed to assure that the total available worth with one worst rod stuck will exceed, at all times, the required control rod reactivity.

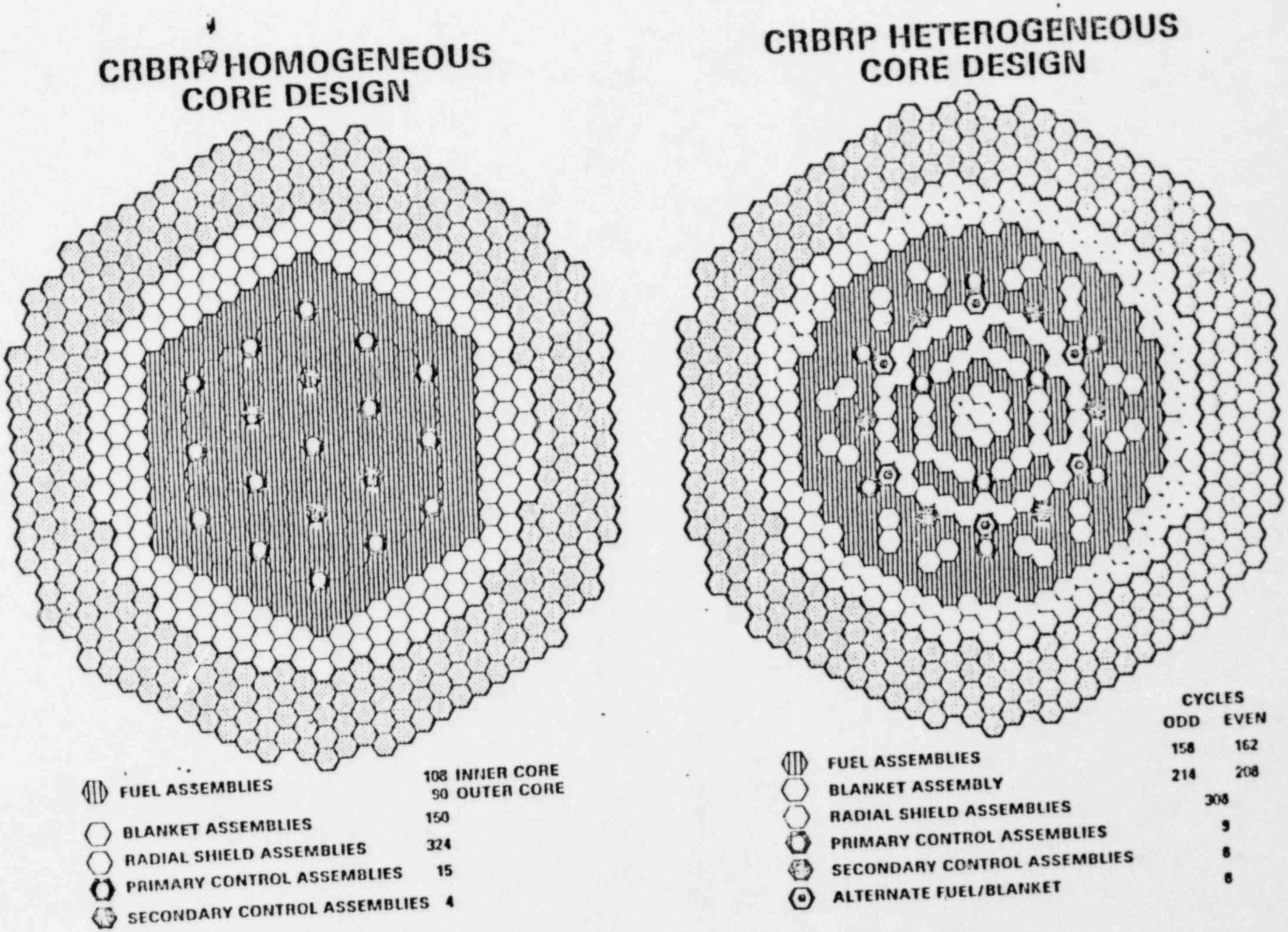

Figure 2.1 CRBRP Homogeneous and Heterogeneous Core Designs.

TABLE 2.1

Issues on Operational Transients

Area	Issues and Comments
Reactor Physics	Criticality calculations Feedback reactivity, including Doppler and sodium temperature coefficients Control rod worth - margin and requirements Reactivity effect due to bowing of fuel and blanket assemblies
Thermal-Hydraulic Analysis	Undercooling events Reactivity insertion events Local fault events Combination of different faults
PPS/PCS Systems	Design specification of PPS/PCS systems Interaction between PPS and PCS systems Sensitivity of settings due to variation in core physics parameters
Rod Design Criteria	Cumulative damage function vs. temperature limits Data base Comparison with LWR design criteria
Steam Generator	Sodium-water reaction Integrity of IHTS and IHX

2.2.2 Reactivity Feedback Coefficients

In going from a "cold" (just critical) to "hot" (power operation) condition the temperature of the core rises, which causes changes in the atom densities as well as microscopic reaction cross-sections. The effects of these changes are expressed in terms of five temperature coefficients of reactivity: a) the Doppler coefficient, b) the sodium temperature coefficient, c) the fuel expansion temperature coefficient, d) the fuel-element bowing coefficient, and e) the power coefficient. All of these coefficients should be verified, particularly the Doppler coefficient which acts very promptly. The CRBR Project has computed this value to be -0.0084. This value for other LMFBRs (with homogeneous core) ranges from -0.0032 to -0.0060. A careful assessment of this prompt negative coefficient of reactivity is essential, as it also impacts on the Plant Protection System (PPS) and the Plant Control System (PCS). The Doppler coefficient is the key feedback reactivity contributor in determining the course of most operational transients.

The reactivity effect associated with the fuel rod bowing can be either positive or negative depending upon the assembly support structure and the radial temperature gradient. This effect was first observed in the EBR-I reactor. Interestingly enough, the accompanying positive reactivity was at first associated with the Doppler effect (which can be positive for highly enriched fuel), but later was confirmed to be due to inward thermal bowing of the fuel assemblies (Ref. 2.6). In the CRBRP the assemblies are held at the top and bottom of the core. Depending upon the radial temperature gradient, fuel assemblies can bow either toward the center of the core, resulting in a positive reactivity contribution, or away from the core, thus causing a negative reactivity contribution. This effect can result in a substantial net positive reactivity. It should be evaluated for different power-to-flow conditions and also for different interassembly gap sizes (nominal gaps to reduced gaps). For CRBRP, the project-computed bowing reactivity contribution for the homogeneous core was as high as +65¢. For the heterogeneous core, this value should be smaller. The reactivity coefficients associated with sodium temperature and with fuel expansions also need to be computed.

Once the reactor attains its operating power and flow, subsequent transient analyses require essentially similar reactivity feedback coefficients. A

generalized representation of the reactivity feedback on reactor power is shown (Ref. 2.6) in Figure 2.2. Any deviation in reactor power causes deviation in the fuel and blanket temperatures, and the coolant temperature. The reactivity effects associated with the change in fuel temperature are predominantly the Doppler and, to some extent, fuel thermal expansion. A deviation in the sodium temperature gives rise to a reactivity contribution due to the resulting change in sodium density. Furthermore, a change in the coolant temperature also results in a change in the duct wall temperature, thereby resulting in an additional reactivity contribution from bowing of fuel and blanket assemblies. The sum of all these individual contributions with the applied reactivity, such as control rod movement, is the total reactivity. The neutron kinetics equations, usually the point-kinetics equations, may now be solved to get a new value for the reactor power.

2.2.3 Reactivity Feedback Due to Core Compaction

A sudden core compaction, due to seismic loading, will result in a positive reactivity. The magnitude of this reactivity change has been estimated, by the CRBR Applicant, to be a 30¢ step for the Operating Basis Earthquake (OBE) and a 60¢ step for the Safe Shutdown Earthquake (SSE). The PSAR formerly considered, in addition, a 90¢ step reactivity insertion. The magnitude of the reactivity insertions, due to core compaction for the OBE and the SSE must be verified for the new heterogeneous core.

2.3 Thermal-Hydraulic Analysis

There are a large number of transients for which thermal and hydraulic performance evaluations must be made. These calculations should be made by using first the primary scram function, and then the secondary scram function to allow for situations where the primary scram signal failed to scram the reactor. For example, in the event of a reactivity insertion of, say, 2¢/sec, the primary scram function is the Flux- Pressure and, if this function is assumed to be inoperative, the secondary scram function is the Flux-Total Flow. The response of the plant for this event (2¢ ramp) must be evaluated for both the primary and secondary scram functions.

For the purpose of thermal-hydraulic analysis, various transients can be grouped into three categories: undercooling, reactivity insertion, and local fault events. A partial list of the undercooling events is:

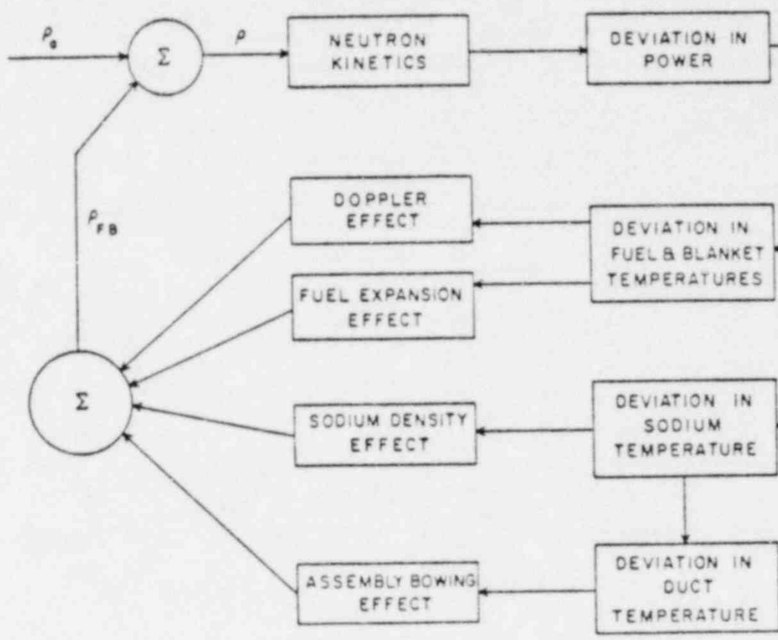

Figure 2.2 Reactivity Feedback Calculational Scheme

Loss of electrical power
Primary pump trip
Primary pump seizure
Coolant leaks
Intermediate pump trip
Intermediate pump seizure
Closure of the evaporator
Closure of the superheater
Loss of main feedwater
Loss of auxiliary feedwater
Steam line break
Failure of the steam bypass system
Intermediate heat transport system leak
Check valve failure
Sodium-water reaction
Pipe breaks
Steam-tube failure
Loss of heat sink
Loss of normal shutdown heat removal

A partial list of the reactivity insertion events is:

Control assembly withdrawal during startup
Control assembly withdrawal at power
Seismic reactivity insertion - step reactivity
Gas bubble passage through the core
Movement of fuel and blanket assemblies
Cold sodium insertion
Control assembly insertion
Mal-operation of PCS

Some of the important local fault events are:

Blockage at the assembly inlet
Blockage within the assembly
Misloading of assembly in wrong inlet module
Interchanging of fuel and blanket assemblies
Side-way compaction of fuel assemblies
Propagation of pin-to-pin failure
Propagation of assembly-to-assembly failure

The approach to be used here will include single faults as well as possible resulting faults. For example, if it turns out that misloading of a fuel assembly (wrong flow inlet module or interchanging with a blanket assembly) goes unnoticed (which is possible), then the consequence of a reactivity insertion for this misloaded assembly must also be investigated.

2.4 PPS/PCS Systems and Their Interaction

The Plant Control System (PCS) is designed to take corrective measures in order to avoid spurious scrams or shutdowns. On the other hand, the Plant Protection System (PPS) is designed to initiate reactor scrams if the perturbation exceeds preset limits. While in many cases the two functions can be complementary, there may be situations where the action by one could counteract the other. For example, the FFTF design initially contained a flow dependent limit within the automatic flux controller power set point circuitry such that the automatic flux controller would reduce power in response to a reduction of primary loop flow. This control feature was included so that a constant flux/flow ratio could be maintained automatically during normal operations. However, if a rapid loss of primary flow event were to occur while operating under automatic flux control, the resultant power decrease delayed the time to trip initiation for both the flux^2/pressure and flux/flow protective functions such that hot channel cladding temperatures exceeded allowable limits. On the basis of these studies, this control feature was deleted from the FFTF design. While this specific feature is not in the CRBRP design, the interaction of the PPS and PCS systems must be studied.

The settings for various PPS functions are based on the neutronic and thermal-hydraulic performance of the reactor core. Any uncertainty or revision in key neutronic or thermal-hydraulic parameters can have significant impact on the acceptability of PPS settings. For example, if the Doppler coefficient of reactivity were to be one-half of the value used in developing PPS settings, a significant amount of fuel could conceivably melt. It is also possible that, in this case, a different PPS function will initiate scram. In other words, the influence of the new core design, with associated feedback coefficients on the system response, must be evaluated in order to verify the efficacy of the PPS.

2.5 Rod Design Criteria

The CRBRP fuel management scheme involves batch replacement of the fuel and blanket assemblies (Ref. 2.2). All fuel and inner blanket assemblies are replaced at two year intervals. The first row of radial blankets is replaced after four years of operation, and the second row of radial blankets is replaced after five years of operation. The safe operation of these rods over

their stated residence time is being quantified in terms of the Cumulative Damage Function (CDF). For emergency events, design criteria calls for the CDF to be less than 1.0 and the accumulated plastic and thermal creep strain to be less than 0.2%. For the faulted events, the criterion used is that the cladding temperature not exceed 2475°F (its melting temperature), and this is conservatively translated into a limit to prevent boiling.

The CRBRP PSAR reports that even for an anticipated transient, like the control rod withdrawal at power, a substantial amount of fuel in the hot fuel channel can be molten. The same should be true for the hot blanket channel (which has roughly 20% more power than the hot fuel channel). The acceptability of molten fuel, even for an anticipated transient, therefore, must be examined. It is conceivable that the molten fuel could ooze through the pellet gaps and contact the cladding tube, thereby resulting in rod failures. The adequacy of the rod design criteria for such an anticipated transient must be assessed.

The CRBRP design criteria appear to have been developed in lieu of a temperature limit for emergency events. This is in variance to the approach used in the light water reactor (LWR) plants. An extensive review and assessment of the CDF criteria, therefore, must be undertaken. Another major comment here is related to the implementation of the design criteria. If the maximum cladding temperature were to be used as a design criterion, implementation can be more direct than the CDF criterion. One would still have to rely on analytical capabilities to substantiate the temperature limits. It appears, however, that the CDF criteria require additional reliance on analysis.

2.6 Sodium-Water Reaction

The materials performance in LMFBRs has been discussed and reviewed in a special issue of Nuclear Technology (Ref. 2.8). It is seen that the steam generators have had a history of tube failures because of a number of factors such as weldment problems, stress corrosion cracking, water chemistry, temperature transients, etc. In LMFBRs (including CRBR) such a failure results in a chemical reaction between sodium and water. Depending upon the number and extent of tube failures (and hence the amount of water available for reaction with sodium), the resulting energy release may cause further propagation of tube failures. On the other hand, the consequences may be controlled if detected early.

Instrumentation is provided to arrest the consequence of a tube failure in the steam generator and the reaction products are removed by a cleanup system. On the other hand, it is quite possible not to detect small leaks until they propagate to involve many tubes. An example is the BR-5 steam generator explosion in 1973 in the USSR.

The problem here has several aspects: a) to determine conservatively the extent of tube failures before corrective action may be taken, b) to compute the hydrogen generation rate and the pressure source term, c) to perform hydrodynamic analysis to ascertain the integrity of the intermediate piping system, and finally, d) to ascertain the integrity of the intermediate heat exchanger.

2.7 Pipe Breaks

Even though the NRC staff has accepted the Project's position of leak before break in the primary cold leg, certain design problems remain to be resolved; the main one being the adequacy of the leak detection system. Additionally, the question of leak before break in the primary hot leg is still an open issue.

REFERENCES

2.1 Clinch River Breeder Reactor Plant - Preliminary Safety Analysis Report, Project Management Corporation (1975), including Amendment 64, dated January 2982.

2.2 P. W. Dickson, "Briefing on Heterogeneous Core Characteristics", Bethesda, Maryland, October 15, 1981.

2.3 W. H. Alliston et al., "Clinch River Breeder Reactor Plant; LMFBR Demo Plant Simulation Model (DEMO), Topical Report", CRBRP-ARD-0005, February 1978 (available from US/DOE Technical Information Center).

2.4 J. V. Miller and R. D. Coffield, "FORE-2M: A Modified Version of the FORE-II Computer Program for the Analysis of LMFBR Transients", CRBRP-ARD-0142, November 1976 (available from US/DOE Technical Information Center).

2.5 R. Pyare and J. G. Guppy, "Transient Analysis of LMFBR Plant with SSC-L and Comparison with DEMO Code", Brookhaven National Laboratory Report, BNL-NUREG-29313 (1981).

2.6 A. K. Agrawal and M. Khatib-Rahbar, "Dynamic Simulation of LMFBR Systems", Atomic Energy Review, 18, 329 (1980).

2.7 W. J. Severson, "Briefing on CRBRP Shutdown Heat Removal - Performance Evaluation", Bethesda, Maryland, November 2, 1981.

2.8 Materials Performance in Nuclear Steam Generators, Proceedings published in Nucl. Technology, 55 (1981).

3. NATURAL CIRCULATION AND SHUTDOWN HEAT REMOVAL

During the earlier phase of the CRBR licensing review, it was a regulatory staff position that no credit would be given for shutdown (decay and sensible) heat removal by natural circulation through the heat transport trains of the plant. This position was based on the lack of a data base to support the potential natural circulation capability of the CRBR design. Additionally, there are numerous concerns regarding the adequate simulation/prediction of the thermal/hydraulic characterization of the system operating under natural circulation conditions. In this section, the issue of natural circulation capability is discussed in light of information that has become available during the past four years. Subsequently, a discussion of the issues related to the performance capability of the shutdown heat removal system components for various operating regimes is provided.

3.1 Natural Circulation Data Base

To adequately portray system operation under natural circulation conditions, there are many inter-related factors which must be represented. The overall driving force for the natural circulation flow is a balance between competing pressure losses and gains throughout the reactor vessel and primary system. The net driving head at these conditions is only a few kPa (1 psi = 6.9 kPa). Some test results (most notably from FFTF) are now obtainable, or will be shortly. These must be assessed closely to determine their true applicability in leading to a reduction of uncertainties in the analytical models and codes.

Most of the LMFBR operating experience (Ref. 3.1) has been in pool-type reactors such as PFR and PHENIX, or in very small experimental loops such as EBR-I, EBR-II, SEFOR, KNK-2 and JOYO. The only large scale loop-type reactors outside of the USSR are Fermi-1, FFTF, SNR-300 and MONJU. Natural circulation tests were not performed in Fermi before the project was canceled, but considerable data were obtained for operational transients. Neither SNR-300 nor MONJU have been completed, so the large scale loop data base is limited to FFTF.

3.1.1 FFTF Natural Circulation Data

The FFTF design relies on passive natural circulation in the heat transport system as a backup to forced circulation as a means of decay heat removal.

As such, the project staff included an extensive array of natural circulation tests (Ref. 3.2) as part of the test program. In the Safety Evaluation Report (Ref. 3.3), the NRC recommends that only limited operation be approved until natural circulation capability has been verified, including verification of the IANUS and FLODISC codes used for Chapter 15 analyses in the FSAR. The natural circulation test program has been completed, but the test data have not been formally released and do not appear to have been analyzed extensively. However, some conclusions can be made based on the non-nuclear natural circulation tests and preliminary results (Ref. 3.5) of the nuclear tests. Specifically,

(1) Heat exchanger performance (both the Intermediate Heat Exchanger (IHX) and the Dump Heat Exchanger (DHX)) was not well predicted at natural circulation conditions. In the case of the DHX this appears to be due to poor modeling (modeling the DHX as a counter-flow heat exchanger rather than a cross-flow heat exchanger). But in the case of the IHX, no such phenomenological basis appears to be present and the project has reduced the heat transfer coefficient in the IHX by a factor of 10 to achieve agreement with the data. The lack of heat exchanger performance data at natural circulation flows was identified by Perkins and Bari (Ref. 3.6) as a problem area, and it apparently remains so.

(2) Pump performance data have been obtained for the FFTF pumps but coastdown times and stopped rotor losses are very design-dependent. Low flow frictional torque and pressure loss data from FFTF could provide a useful comparison to CRBR pump description data, but better data should be available from the CRBR component verification program. Informal discussions with CRBR project staff indicate that water tests have been performed for a CRBR prototype pump, but the data have not yet been submitted.

(3) Upper plenum mixing data in FFTF are very limited. The FFTF project staff interprets (Ref. 3.5) the data as supporting a two-zone mixing model (as a function of momentum and buoyant head) for pony motor flow rates, but they acknowledge severe stratification at natural circulation flows. Even at full flow rates the upper plenum mixing appears to be incomplete and a new phenomenon of bleed flow is identified (Ref. 3.5) to account for discrepancies between the upper plenum temperature and the hot leg temperature. Previous CRBR natural circulation calculations (Refs. 3.7,3.8) assumed stratification of the

upper plenum (thereby reducing the buoyant head term), but the present project's verification program (Ref. 3.9) uses a multi-region upper plenum model which is difficult to relate to the previous assumptions.

(4) Possible stratification in the piping at natural circulation flows was identified as a problem area for both CRBR (Ref. 3.7) and FFTF (Ref. 3.10). From a licensing standpoint, stratification is important in that the effect may be detrimental to natural circulation and is not modeled in the one-dimensional codes (Refs. 3.11,3.12) used in licensing analyses. However, stratification effects can be accounted for even in a one-dimensional code (Ref. 3.13) if the magnitude of the effect is known a priori. While there is no direct measurement of temperature distributions in the FFTF piping, the FFTF project now acknowledges (Ref. 3.5) that the limited data indicate that piping stratification occurred in the primary hot leg during the natural circulation tests.

(5) Check-valve pressure drop is also expected (Ref. 3.14) to be an important contributor to natural circulation performance. FFTF has characterized their check valves in the component testing program but, as with the pumps, the pressure loss characteristics are very design-dependent. The CRBR project staff has reduced their estimate of pressure losses at the check valve (Ref. 3.9), but they have not yet submitted supporting data.

3.1.2 In-Vessel Flow Redistribution

Most of the experimental data and analytical calculations have concentrated on determining flow and temperature patterns at full flow and full power conditions. However, recognition of uncertainties in local flow rates has generated considerable interest in analytical (Refs. 3.15, 3.16, 3.17, 3.18, 3.19) and experimental (Refs. 3.2, 3.20, 3.21) investigations of flow redistribution at natural circulation conditions. The original CRBR project calculations (Ref. 3.8) for natural circulation assumed no flow redistribution within the core, but BNL pointed out (Ref. 3.7) that this procedure underestimated the average blanket temperature difference by about 25%, and boiling would be predicted in the hot channel for the natural circulation event (using the hot channel factors developed by the project). Both the FFTF and the CRBR projects have relied heavily on the FFTF natural circulation tests to verify their flow redistribution models. However, instrument tree bypass flows make the FFTF data difficult to interpret. Only the Fuels Open Test Assembly (FOTA) data

appear (Ref. 3.5) to be accurate at natural circulation flow rates. As previously indicated (Ref. 3.4), the relative difference between FOTAs is not a sensitive measure of flow redistribution.

In-vessel flow redistribution, including recirculation, was shown (Ref. 3.22) to be an important mitigating effect for the loss-of-heat sink scenarios (see Section 6). However, subsequent analyses indicate that recirculation would not be sufficient to delay boiling for FFTF (Ref. 3.23) and that intermediate powered assemblies could be expected to stagnate and boil early in the loss-of-heat-sink transient for CRBR (Ref. 3.24).

3.1.3 Cladding Temperature Limits for Natural Circulation

There has been considerable interest in whether sodium boiling is a viable means of decay heat removal. However, both analyses (Refs. 3.25, 3.26) and experiments (Refs. 3.27, 3.28) tend to be somewhat contradictory. While the debate can be expected to continue, the CRBR project has taken the position that they will use a temperature limit for design events which, conservatively, precludes boiling. In this regard, it should be noted that the sodium boiling point at atmospheric pressure is 1620°F. The boiling point can, of course, be higher in the core when the gravitational head is considered. Neglecting the pressure drop due to forced flow (assuming low-flow natural circulation conditions), the gravitational head of sodium above the core exit region brings the boiling point up to 1700°F.

3.1.4 Applicability of FFTF Data to CRBR

While the FFTF natural circulation test program provides considerable support for the viability of natural circulation in FFTF, the degree of conservatism in the licensing calculations (Ref. 3.3) has yet to be assessed. At the present time, apparent inconsistencies (Ref. 3.5) in the FFTF data make interpretation difficult, but once the data are released for detailed evaluation, some of the inconsistencies may be resolved. The released data should also help to establish the validity of the plant transient codes such as SSC (Ref. 3.30) and DEMO (Ref. 3.11). In any case, the importance of component specific data appears to make it necessary to reevaluate CRBR natural circulation performance in the light of new CRBR component test data and FFTF loop performance data.

3.2 Thermal-Hydraulic Characterization at Natural Circulation Conditions

There are numerous concerns regarding the adequate simulation/prediction of the thermal/hydraulic characterization of reactor system operation under natural circulation conditions. The applicant has an intensive on-going effort in this area aimed at the demonstration of natural circulation capability in the CRBRP. The CRBR Natural Circulation Verification Program (to be submitted) is intended to validate DEMO-REV5, COBRA-WC and FORE-2M. However, none of these codes were used in the natural circulation analysis (Ref. 3.8) and DEMO-REV5 is not yet available. In the previous review (Ref. 3.7), the blanket thermal performance was found to be a dominant concern and could approach boiling even with one set of pony motors on. The previous analysis appears to have little relevance to the present verification effort due to a number of important changes in system characteristics. Most notably, the impact of the new heterogeneous core configuration on natural circulation capability was not evaluated in the previous review. The influence of this new design must be carefully considered on a consistent basis.

The CRBR Project's briefing on decay heat removal (Ref. 3.9) indicated (see Table 3.1) the major changes in their analysis which affect core thermal performance, but they provided little basis to assess the importance of the changes. The summary only raised additional concerns. In particular,

(1) The pump stop time and stopped rotor loss estimates are important changes from the previous submission and are based on prototype data not yet submitted.

(2) The estimated check-valve pressure drop has also been changed significantly and the supporting data should be submitted.

(3) The effects of core flow and heat redistribution to be included in the CRBR project analysis are expected (Ref. 3.7) to raise estimates of blanket temperatures rather than lower them, as implied by the project.

(4) The importance of a new upper plenum model is difficult to assess without having the model available.

(5) The decrease in estimated decay power is said to be due to a decrease in uncertainty estimates, but such a decrease in uncertainty appears to conflict with present NRC requirements (Ref. 3.31).

Table 3.1

FACTORS POTENTIALLY AFFECTING INITIAL ASSESSMENT RESULTS FOR CRBRP

	1976 Predictions	Revised Predictions	Effect On Peak Temperatures
1. Pump stop time	56 sec	>110 sec.	Lower
2. Core flow and heat redistribution	Neglected (fixed flow fractions)	To be included	Lower
3. Reactor upper internals	Neglected	UIS model included	Raise
4. CKV ΔP	.07 psi @ 3% flow	.05 psi @ 3% flow	Lower
5. Pump stopped rotor ΔP	.08 psi @ 3% flow	.11 psi @ 3% flow (Based on water data)	Raise
6. Reactor ΔP	.145 psi @ 3% flow	.178 psi @ 3% flow	Raise
7. Piping/plena heat capacity effects	Neglected	Included	Will smooth temps, small effect on core temps
8. Total decay power	4.15% @ 100 sec. 3.4% @ 300 sec.	3.65 @ 100 sec. 2.9 @ 300 sec.	Lower

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The new heterogeneous core, along with over-cooled inner blankets at beginning of cycle conditions, would appear to make thermal-striping (and associated thermal stresses) a potential problem for the new core, particularly at beginning-of-life.

3.3 Shutdown Heat Removal System Issues

3.3.1 Introduction

The general issue to be addressed in this section is the capability of the various shutdown heat removal systems (SHRS) proposed for the current CRBRP design to remove the decay and sensible heat for all postulated operating conditions. In the aftermath of Three Mile Island, decay heat removal systems are receiving increased attention to ensure that their capabilities are properly verified under all potential modes of operation. The breakdown of this general issue into individual issues is approached by: 1) addressing the issues related to the individual components of the various SHRS, and 2) addressing the issues relating to the operation of the total integrated system. Additionally, general issues relating to the appropriate design limits to be applied to the various events, as well as the identification of the significant events, are discussed.

Some of these issues are probabilistic in nature, while others are mechanistic. The intent here is to address in more detail those issues of a thermal/hydraulic nature, while just highlighting those in the probabilistic or reliability areas, as they are discussed more fully in Section 5.

In this section, the categorization of potential shutdown heat removal events is first discussed, along with a brief description of the various SHRS, to indicate the scope and complexity of these systems. Next, specific issues identified at this time regarding each of the systems and their integrated operations are presented. Finally, an approach to the resolution of the issues and questions is proposed.

3.3.2 Shutdown Heat Removal Systems and Events

The discussion of events including operation of the various shutdown heat removal systems focuses on the categories of scenarios, as summarized in Table 3.3.1. A list of the various SHRS and subsystems is given in Table 3.3.2.

Table 3.3.1 Grouping of SHRS Events

Group	Assumed Power Sources Available	SHRS Components Operating
(1) Normal Shutdown	All off-site and on-site	Main heat transport loops, turbine bypass and main feedwater
(2) Upset	On-site only	PHTS, IHTS, evaporators under natural circulation, auxiliary feedwater and PACC, steam drums
(3) Emergency	Batteries Only	Loop Three PHTS and IHTS only Loop 3 PACC and turbine-driven auxiliary feedwater pump
(4) Faulted	At least diesels and batteries	One or more PHTS loops and DHRS

Table 3.3.2 Shutdown Heat Removal Systems and Components

System	Components
Main Heat Transport System	Primary Heat Transport System (PHTS) including IHX and pony motor Intermediate Heat Transport System (IHTS), including pony motor Steam Generator System Evaporator Module Superheater Module Recirculation Pump Main Feedwater System Protected Air Cooled Condensor (PACC) Turbine Bypass Main Condensor
Steam Generator Auxiliary Heat Removal System (SGAHRS)	Auxiliary Feedwater System (AFWS) One turbine-driven pump Two motor-driven pumps Protected water storage tank (PWST) Protected Air Cooled Condensors (PACC) Superheater Isolation Values Evaporator and Superheater Vent Valves
Direct Heat Removal Service (DHRS)	Vessel Overflow Tank Two sodium electromagnetic pumps Overflow heat exchanger (Na/NaK) Two NaK pumps Two airblast heat exchangers (ABHX) (NaK/Air)

(1) For normal plant shutdown (i.e., assuming no loss of off-site or on-site power), the CRBRP design provides for the removal of decay and sensible heat via forced convection through all main primary heat transport systems (PHTS) and steam generator systems (SGS), see Figure 3.3.1a. All PHTS and IHTS pony motors are operating; the main feedwater pumps and recirculation pumps are operating, and the residual heat is ultimately removed via the main condensor. Thus, no reliance is placed on natural circulation. However, the actions of a complex control system scheme and the subsequent manipulations of pumps and valves are required to assure the smooth operation of this procedure.

(2) For loss of off-site power events (assuming availability of diesel generators), heat is still removed via forced convection utilizing pony motors in the PHTS and IHTS. However, in the SGS, power is not available to the main feedwater and recirculation pumps. Thus, the water side of all evaporators and superheaters must rely on natural circulation. Additionally, operation of the two main subsystems of the steam generator auxiliary heat removal system (SGAHRS), specifically the auxiliary feedwater system (AFWS) and the protected air-cooled condensor (PACC), are automatically initiated (see Figures 3.3.1b and 3.3.2). The SGAHRS requires the complex automatic operation of various control systems, pumps and valves, as well as a water supply from the protected water storage tank (PWST) or main condensate storage tank to assure its mission success. The PACC, which provide the ultimate long term coolability, rely solely on natural circulation on the water side for heat removal.

(3) Under loss of all off-site and on-site power events, heat will be removed by natural circulation within the primary and secondary sodium loops. The SGAHRS will use the steam dump valves to remove heat until the PACC can accept the load with natural circulation of air. Auxiliary feedwater is provided from the PWST by the turbine-driven pump only, which is driven by steam from the steam drum(s).

(4) For shutdown heat removal events in which loss-of-heat-sink (LOHS) through the SGS and/or IHTS is assumed, decay heat is to be removed through the direct heat removal service (DHRS) provided via the reactor vessel sodium overflow system (see Figure 3.3.3). The operation of this system requires the manual realignment of six valves and the startup of various pumps. Additionally, for the present CRBRP design, the DHRS a) cannot operate in a natural circulation mode and thus must have at least the diesel generators available,

Figure 3.3.1

PLANT CONDITIONS AFTER TRIP VERSUS LOSS OF OFFSITE POWER

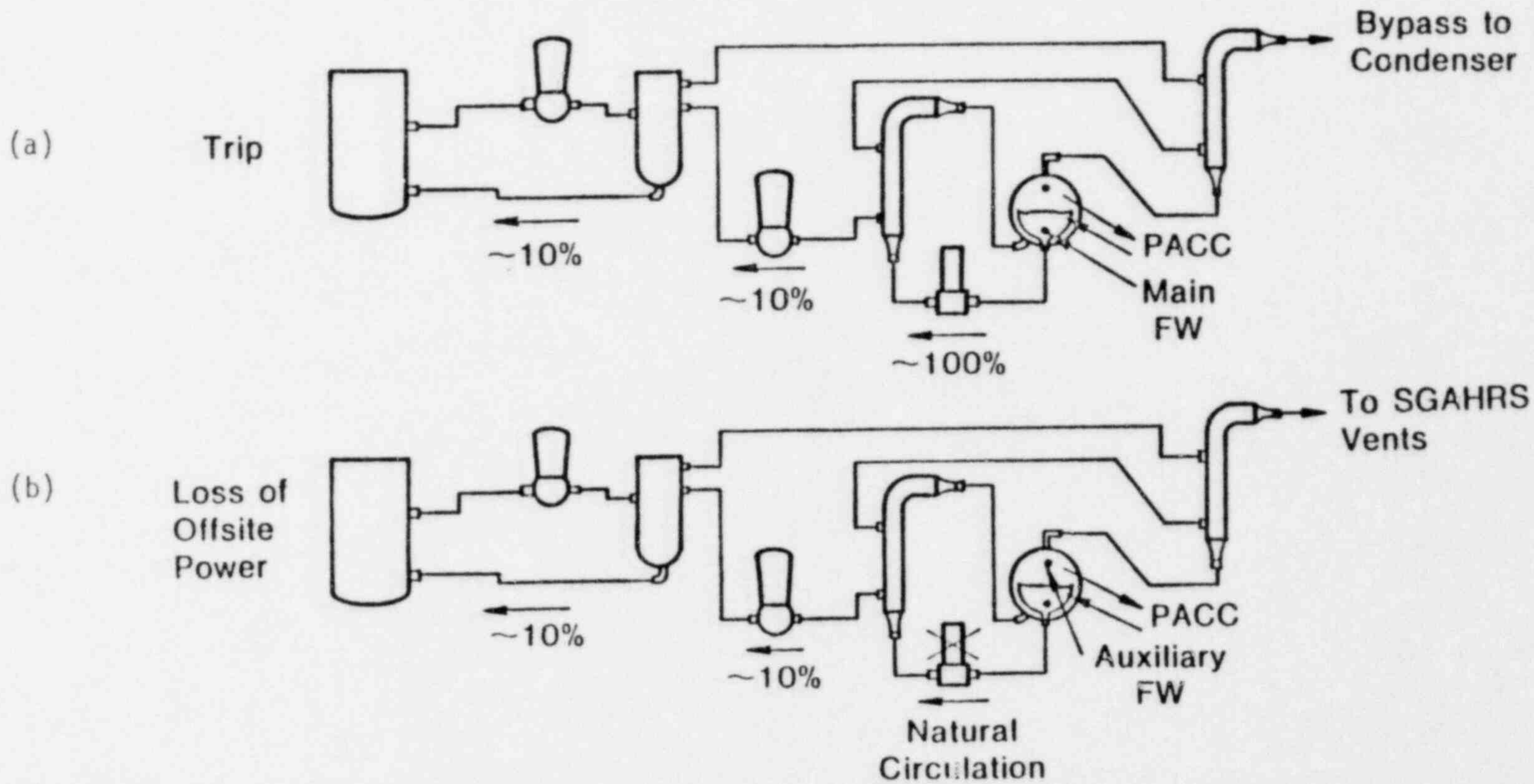

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Figure 3.3.2

SHUTDOWN HEAT REMOVAL SYSTEM SCHEMATIC (WITHOUT DHRS AND GUARD VESSELS)

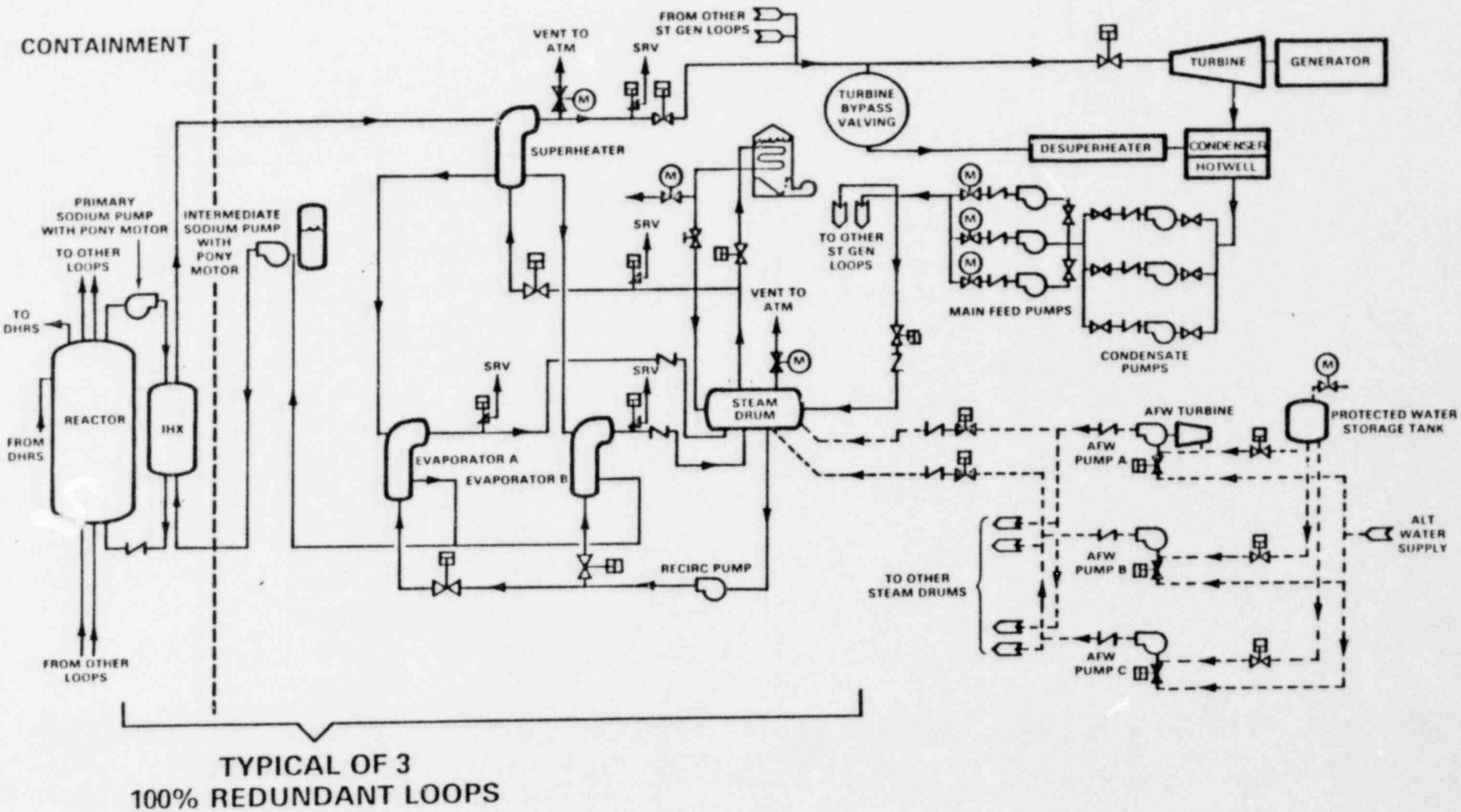
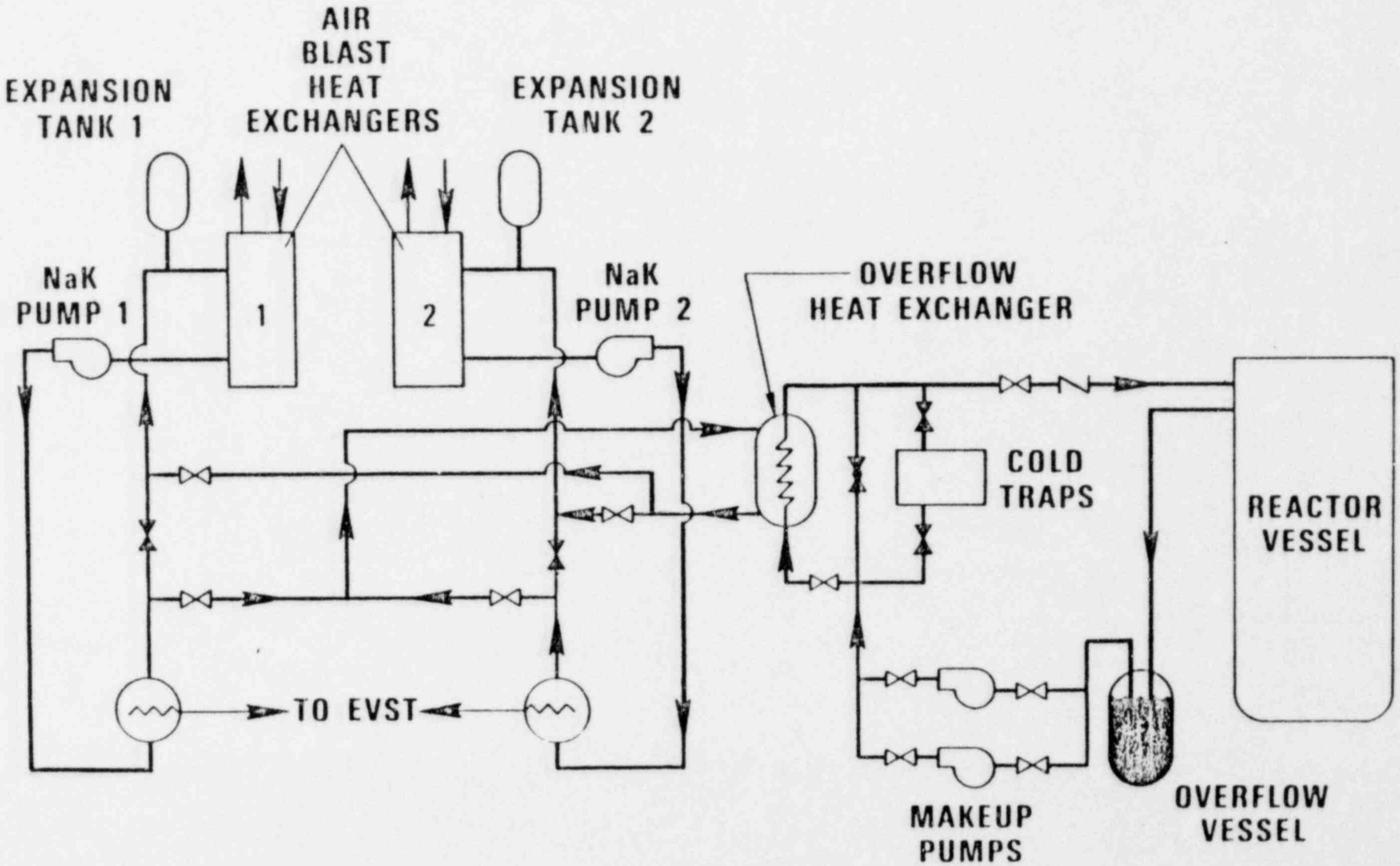


Figure 3.3.3

THE DIRECT HEAT REMOVAL SERVICE VALVE LINEUP PERMITS THE USE OF THE EVST NAK AIRBLAST HEAT EXCHANGERS AS THE SINK



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b) requires approximately one-half hour to accomplish the manual realignment, c) assumes the integrity of all PHTS, and d) requires that forced convection be continuously provided via all three PHTS pony motors.

(5) The consequences of LOHS scenarios which could ultimately lead to core fuel/cladding melt and loss of coolable geometry are addressed in a separate section (see Section 6).

3.3.3 Issues Related to Individual SHRS Components

Certain issues exist which must be addressed for the various components and subsystems of the shutdown heat removal systems to assure that they will successfully fulfill their respective functions. Those issues currently identified are discussed in the following subsections and highlighted in Table 3.3.3.

Direct Heat Removal Service

(1) Effect of Sodium Level: In the current design, the inlet to the DHRS (refer to Figure 3.3.4) is located behind the thermal liner (i.e., the inlet does not penetrate through the thermal liner). Sodium into the thermal liner originates from the bottom (which will essentially be cold sodium at the inlet plenum temperature) and from the top through thermal liner ports which are located only about 2.5 inches below the nominal sodium operating level. Consequently, the only source of upper plenum sodium to the DHRS will be through these highly elevated ports. Thus, the sodium level under required DHRS operation must be closely assessed. Additionally, the DHRS will offer no heat removal capability for any serious PHTS pipe break events.

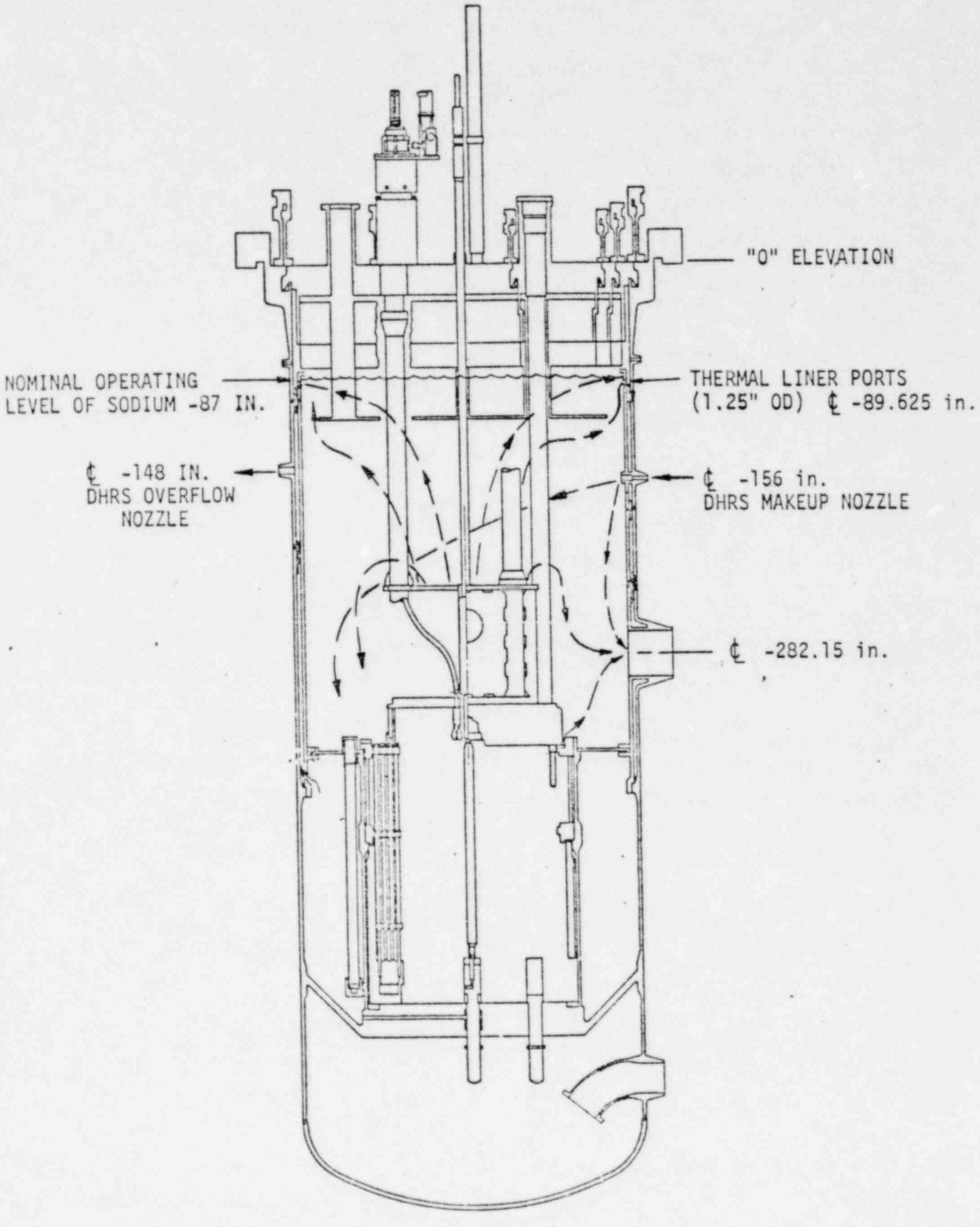
(2) Upper Plenum Effects: The applicant currently assumes that the upper plenum sodium is well mixed, given that all PHTS pony motors are providing forced convection during DHRS operations. Current evidence (see also Section 3.1) indicates that this may not be the case. Since the only source of hot sodium received by the DHRS comes from the uppermost portion of the upper plenum (refer to Figure 3.3.4 and also to previous discussion under this subsection), the impact of stratification and subsequent heat removal capability must be investigated. Additionally, the outlet from the DHRS exits directly into the upper plenum through the thermal liner (refer to Figure 3.3.4) at a location

Table 3.3.3 SHRS Issues - Component Level

SHRS Component	Identified Issues
DHRS	• Effect of sodium level • Upper Plenum effects • Thermal stress limits • Single valve failure • Capability of instrumentation and control
PACC	• Verification of natural circulation capability • Verification from prototype testing • Capability of instrumentation and control
Main Evaporator Modules	• Verification of natural circulation capability • Verification from prototypic testing

Figure 3.3.4

DHRS Flow Operation



almost opposite from the inlet. The inlet stream velocity is around 6-10 feet/second. The applicant is claiming the usage of a conservative estimate, based on experimental evidence (see also Section 3.4) to provide an upper bound on the amount of cold sodium from the DHRS outlet which may potentially communicate directly with the DHRS inlet. This direct cross flow will limit the effectiveness of the DHRS and must be verified. Also, the admissible usage of data derived from water tests as applied to a sodium system under these circumstances must be addressed.

(3) Adequacy of DHRS Design to Thermal Stress: Under the proposed operating conditions, the DHRS will be exposed to sodium temperatures in excess of 1100°F (865°K) for extended periods of time. Since the ex-vessel storage tank components, which comprise the NaK portion of the DHRS, are normally exposed to operating temperatures in the 400-600°F ($\sim$475-590°K) range, the structural integrity of the DHRS must be verified. Also, for the design basis event under which DHRS operation is necessary, the sodium entering the upper plenum from the DHRS will be at approximately 550-600°F (assuming 11 MW heat removal) while the upper plenum bulk sodium and structural temperature may be above 1100°F.

(4) Single Valve Failure: In the required manual realignment of the sodium overflow system and the ex-vessel storage tank system to form the DHRS, operation of a single, non-redundant valve is required. The reliability and unavailability of this valve must be determined.

(5) Instrumentation and Control: The capability of the instrumentation and control information available must be verified to ensure that the operator can correctly ascertain the true plant condition and implement any required DHRS action.

Protected Air-Cooled Condensor (PACC)

(1) Verification of Natural Circulation Capability: The PACC subsystem is designed to operate solely by natural circulation on the water side. On the air side, the preferred operating mode is via forced convection; however, natural circulation capability is claimed. The applicant has as yet provided no substantive basis to verify the natural circulation capability of such a unit for these applications. In view of the heightened importance placed on the PACC, particularly for loss of all on-site and off-site power events, the

assurance of natural circulation capability must be verified. Also, in those areas where operation under abnormal conditions (e.g., asymmetric transients or single failure cannot be tested (see next subsection)), verified analytical tools must provide supportive analyses.

(2) Assessment of Prototype Testing: The applicant currently plans to test a full scale prototype of the PACC subsystem. However, due to experimental test facility limitations, full power operation is currently questionable. Also, it is essential that the testing provide a sufficient basis to ensure that any analytical tool used to predict PACC response to potential transients outside the scope of the experiment can be adequately validated.

(3) Instrumentation and Control: The capability of the instrumentation and control information available must be verified to ensure that the plant operator can correctly ascertain and implement any required PACC action.

Main Steam Generator Evaporators

(1) Verification of Natural Circulation Capability: Whenever off-site power is unavailable, the evaporator modules will be required to operate under natural circulation on the water side, with no effective operator intervention possible. Short term steam venting will be accomplished through steam dump valves downstream of the superheaters. Subsequently, the superheaters will become effectively isolated and heat removal will be accomplished directly from the evaporator to the steam drum to the PACC, via natural circulation. Due to the importance of the evaporators to operate under natural circulation whenever required, it is essential that this capability be either experimentally verified or analytically calculated by validated predictive tools.

(2) Assessment of Prototype Testing: The applicant presently plans to test a full scale steam generator module (evaporator and superheater designs are identical). This testing must be closely followed to ensure that it provides a sufficient data base either to experimentally verify all modes of required normal and abnormal operating conditions or to validate analytical tools used to predict operating conditions not covered during the planned testing.

3.3.4 Issues Related to Total Integrated Plant Response During SHRS Events

The capability to remove sensible and decay heat under all postulated accident conditions involves the complex interaction of a number of main plant components and SHRS components. Two identified issues in this integrated plant

response area are general in nature, while others may be grouped under the various categories of events discussed in Section 3.3.2 and noted in Table 3.3.1. The following subsections delineate these concerns, which are also highlighted in Table 3.3.4.

1) Specification of design limits

There must be a clear definition of the design limits within which the plant response must fall for specific events. The classifications designated by the project include normal, upset, emergency and faulted events. For those events involving SHRS response, the appropriate limits for the various categories of events must be clarified. Thus, the capability of the system response can be quantitatively assessed.

2) Identification of significant SHRS events

The scenarios involving operation of the SHRS are many and varied. A consistent means must be used to identify the significant transients of interest and rank them according to their relative severity so that the appropriate design limits discussed in the previous subsection can be applied.

3) Plant response to SHRS events assuming availability of off-site and on-site power

Identified issues of concern here include:

a) Effects to individual components due to single failures such as; inadvertent operation of the turbine driven auxiliary feedwater pump; stuck open relief valve; failure in the turbine bypass system; leak in the main feedwater header common to all SG loops; inadvertent operation of a PACC.

b) Effects to the remainder of plant due to asymmetric operation resulting from failures.

c) Capability of instrumentation and control available to the operator to ascertain the true plant condition and implement corrective action if required.

d) Impact of aforementioned failures under any proposed plant operations with one loop out of service.

Table 3.3.4 SHRS Issues - Integrated System Level

- Specification of design limits.
- Identification of significant events.
- Plant response to single failures for normal, upset, emergency and faulted events.
- Effects due to asymmetric operation.
- Capability of instrumentation and control.

4) Plant response to SHRS events given loss of off-site power supplies

Here the effects due to single failures and resulting asymmetric operations are more pronounced. Identified issues of concern include:

a) Loss of a diesel generator and resulting unavailability of dependent components such as pumps, valves and controls. Delayed loss of a diesel may be potentially worse with regards to a transition to natural circulation in the affected loop(s) due to degradation of thermal head.

b) Failure of the protected water storage tank auxiliary feedwater supply which supplies all SG loops.

c) Failure of an auxiliary feedwater pump or its associated controls, particularly the full-sized turbine driven pump.

d) Failure in the turbine bypass valve system.

e) Capability of instrumentation and control available to the operator to ascertain the true plant condition and implement corrective action if required.

f) Impact of aforementioned failures under any proposed plant operations with one loop out of service.

5) Plant response to SHRS events given loss of all off-site and on-site power supplies

The plant is required to operate with natural circulation under blackout conditions as discussed in Sections 3.1 and 3.2. Identified issues of concern include:

a) Failure of turbine driven auxiliary feedwater pump.

b) Failure of either superheater or steam drum vent valve in loop 3.

c) Failure of protected water storage tank system.

d) Effect of upper plenum stratification. Stratification may be severe under these very low flow conditions (see also Section 3.1).

e) Capability of instrumentation and control available to the operator to ascertain the true plant condition and implement corrective action if required.

6) Plant response to LOHS events when the DHRS is operating

The applicant position for these scenarios appears to be that all PHTS must be intact and power to all primary loop pony motors must be available. Under these circumstances, a single additional failure could lead to a loss of coolable geometry. Such single failures and additional outstanding issues include:

a) Failure of one pony motor (instantaneously or delayed)

b) Failure of the single non-redundant valve in the DHRS (see Section 3.3.3)

c) Failure of any DHRS pump

d) Failure of Na/NaK heat exchanger

e) Failures of either Nak/air heat exchanger

f) In-vessel sodium level drops below thermal liner ports (see Section 3.3.3)

g) Stratification in upper plenum (see Section 3.3.3)

3.3.5 Approach to Resolution of Issues

The general approach necessary to resolve the issues and questions regarding the capability of the shutdown heat removal systems can be summarized by grouping the previous discussions into four main areas:

1) Define clearly the appropriate safety and design limits for the various types of SHRS events.

2) Use a consistent methodology to identify all potentially significant SHRS transients. The CRBRP design must be carefully reviewed, with particular emphasis on recent changes, to ensure that all important transients are identified. Here a combined effort involving probabilistic and realjability analysis, engineering judgment and supportive parametric-type deterministic analysis will be required.

3) Conduct a close assessment of all completed and proposed experiments which provide substantiation or expansion of the SHRS data base or which lead to the reduction of uncertainties.

4) Apply validated predictive tools to analyze potential SHRS events and to provide any required independent assessment of results supplied by the applicant.

REFERENCES

3.1 D. Majumdar and R. J. Cerbone, "Survey of Available LMFBR Facilities for Testing Safety-Related Thermal-Hydraulic Computer System Codes", BNL-NUREG-23429 (October 1977).

3.2 G. D. Bouchey, S. L. Additon and E. A. Parziale, "Planned Natural Circulation Testing in the Fast Flux Test Facility", ANS Transactions, 28, p. 522 (June 1978).

3.3 U. S. Nuclear Regulatory Commission, "FFTF Safety Evaluation Report", USNRC Report NUREG-0358 (August 1978).

3.4 K. R. Perkins, R. A. Bari and L. C. Chen, "An Evaluation of Natural Circulation Tests in FFTF", BNL Informal Report, BNL-NUREG-29348 (March 1981).

3.5 R. L. Stover et al., "FFTF Natural Circulation Seminar, HEDL, Richland, Washington (September 30, 1981).

3.6 K. R. Perkins and R. A. Bari, "Preliminary Review and Evaluation of Natural Circulation Decay Heat Removal Capability in CRBR and FFTF", BNL Informal Report, BNL-NUREG-21913 (October 1976).

3.7 K. R. Perkins, R. A. Bari and D. C. Albright, "Uncertainties in the Calculated Response of the Clinch River Breeder Reactor During Natural Circulation Decay Heat Removal", BNL Informal Report, BNL-NUREG-22715 (April 1977).

3.8 R. R. Lowrie and W. J. Severson, "A Preliminary Evaluation of the CRBRP Natural Circulation Decay Heat Removal Capability", WARD-D-0132 (March 1976).

3.9 J. R. Longenecker et al., "Briefing on CRBRP Shutdown Heat Removal", Bethesda, Maryland (November 2, 1981).

3.10 K. R. Perkins, R. A. Bari, L. C. Chen and D. C. Albright, "Analysis of FFTF System Transients", BNL Informal Report, BNL-NUREG-25560 (January 1979).

3.11 Westinghouse ARD, "LMFBR Demo Plant Simulation Model (DEMO)", WARD-D-0005, Rev. 4 (January 1976).

3.12 C. F. Wolfe, "IANUS - Application Information", WARD-FPC-1253, Rev. 1 (May 1976) (Proprietary).

3.13 M. Khatib-Rahbar, I. K. Madni and A. K. Agrawal, "Impact of Thermal Buoyancy in LMFBR Piping Systems", Proc. of Specialists' Meeting on Decay Heat Removal and Natural Convection in FBRs, Brookhaven National Laboratory (February 28-29, 1980).

REFERENCES (CONT.)

3.14 I. K. Madni and A. K. Agrawal, "LMFBR System Analysis: Impact of Heat Transport System on Core Thermal-Hydraulics", Specialists Issue of Nuclear Engineering & Design, Nucl. Eng. Design 62, 199 (1980).

3.15 A. K. Agrawal et al., "Transient Simulation of LMFBR Systems", presented at the Mathematics and Computation Division Topical Meeting on Improved Methods for Analysis of Nuclear Systems, Tucson, Arizona (March 28-30, 1977); Nucl. Sci. Eng., 64, 480 (1977).

3.16 A. K. Agrawal, J. G. Guppy, I. K. Madni and W. L. Weaver, "Dynamic Simulation of LMFBR Plant Under Natural Circulation", ASME Paper No. 79-HT-6, presented at the 18th ASME National Heat Transfer Conf., San Diego, California (August 5-8, 1979); J. Heat Transfer, 103, 312 (1981).

3.17 J. Muraoka et al., "FLODISC - A Dynamic Core Flow Distribution Code: Evaluation of the Total Loss of Electric Power Event", HEDL-TC-874 (May 1977).

3.18 J. V. Miller and R. D. Coffield, "FORE-2M: A Modified Version of the FORE-II Computer Program for the Analysis of LMFBR Transients", WARD-D-0142 (May 1976).

3.19 K. R. Perkins and R. A. Bari, "Interassembly Flow Redistribution at Natural Circulation Conditions in the Fast Flux Test Facility", Trans. Am. Nucl. Soc., 30, 413 (November 1978).

3.20 M. L. Millburg, J. A. Hassberger and C. J. Boasso, "Natural Circulation Heat Transfer Testing with a Simulated Full-Scale LMFBR 217-Pin Electrically Heated Fuel Assembly". HEDL-TME 77-3 (June 1977).

3.21 R. M. Singer et al., "Studies Related to Emergency Decay Heat Removal in EBR-II", Proc. of the International Meeting on Fast Reactor Safety Technology, p. 1590, Seattle (1979).

3.22 C. K. Chan, T. K. Min and D. Okrent, "A Look at Alternative Core Disruption Accidents in LMFBRs", UCLA-ENG-7720 (February 1977).

3.23 K. R. Perkins, W. T. Pratt and R. A. Bari, "Evaluation of In-Vessel Natural Circulation During a Hypothetical Loss-of-Heat-Sink Accident in the Fast Flux Test Facility", BNL Informal Report, BNL-NUREG-26565 (August 1979).

3.24 M. J. Khatib-Rahbar, J. G. Guppy and A. K. Agrawal, "Hypothetical Loss-of-Heat-Sink and In-Vessel Natural Convection: Homogeneous and Heterogeneous Core Designs", Decay Heat Removal and Natural Convection in Fast Breeder Reactors, Hemisphere Publishing Co., New York, N. Y. (1981).

3.25 F. E. Dunn, "Severe FFTF Natural Circulation Transients with Boiling", ANL/RAS 76-26 (September 1976).

REFERENCES (CONT.)

3.26 K. R. Perkins and R. A. Bari, "SAS-3D Evaluation of Boiling at Decay-Heat Levels in FFTF", Trans. Am. Nucl. Soc., 33, p. 516 (1979).

3.27 P. W. Garrison, R. H. Morris and B. H. Montgomery, "Natural Convection Boiling of Sodium in a Simulated FBR Fuel Assembly Subchannel", Proc. of the International Meeting on Fast Reactor Safety Technology, Vol. IV, Seattle (August 1979).

3.28 A. Kaiser, W. Peppler and M. Straka, "Decay Heat Removal from a Pin Bundle", CONF-761001, IV, p. 1578 (October 1976).

3.29 Preliminary Safety Analysis Report, Clinch River Breeder Reactor Plant, Project Management Corporation, docketed June 1975.

3.30 A. K. Agrawal et al., "An Advanced Thermohydraulic Simulation Code for Transients in LMFBRs (SSC-L Code)", Brookhaven National Laboratory, BNL-NUREG-50773 (February 1978).

4. CONTAINMENT ANALYSIS

Technical issues related to the adequacy of the CRBR containment were summarized in Reference 4.1 at the termination of NRC licensing of the Project. In Table 4.1, we summarize these issues and provide a brief description of how much additional review was considered necessary at that time. The Project responded (Ref. 4.2) to the NRC staff review and in Table 4.2 we summarize their assessment of the resolvability of each of the items noted in Table 4.1.

In this section, we follow the format indicated in Table 4.1. Initially, we discuss large sodium releases in the Steam Generator (S/G) building in Section 4.1. We then discuss containment system design in Section 4.2. Finally, in Section 4.3 we discuss the impact of the new core design on radiological consequences associated with accidents beyond the design basis. The discussion on radiological consequences was not included in Reference 4.1 so it does not appear in Table 4.1.

4.1 Large Sodium Releases in the S/G Building

In Table 4.1, we summarize the major issues remaining (in the view of the NRC staff) at the suspension of CRBR licensing. Further evaluation of the models and assumptions used by the project in their analysis of sodium fires was considered necessary. The adequacy of the S/G building design pressure and temperature was not considered to be established. Additional information on the analyses of sodium fires was requested (Q001.703). Finally, further assessment of the adequacy of the proposed fire suppression and nitrogen flooding systems for sodium fires was also considered necessary. The CRBR Project's response to the above issues is briefly summarized in Table 4.2. The Project staff indicated their intention to update the PSAR to address the issues of model assumptions, S/G building design and the request for further information. However, with regard to the fire suppression systems the project staff considers that sufficient information was provided in the PSAR for the NRC evaluation. A supplement to the PSAR was planned by the Project staff for further clarification in this area. Clearly, a review of all of this additional information is required to assess the adequacy of the Project's analysis of large sodium releases in the S/G building. Issues discussed in Section 4.2 related to the choice of the DBA and integrity of the cell liners also impact the issues in this section.

Table 4.1 Summary of outstanding technical issues related to CRBR containment analysis*

Technical Issue	Comments
Large Sodium Releases in S/G Building	o Understand and evaluate models used in PSAR.
	o Adequacy of S/G building design pressure and temperature.
	o Additional information requested on sodium fire analysis.
	o Adequacy of proposed fire suppression and N_2 flooding systems.
Containment System Design	
Large Sodium Releases and DBAs	o DBA selection.
	o Preliminary review of containment isolation system-further review needed.
	o Control the accumulation of H_2.
	o Review of autocatalytic recombination test results needed.
	o Capability of TMBDB systems to function in hostile environments.
	o Pipe break spectrum.
	o Cell liners as ESF.
	o SOFIRE, SPRAY, and CACECO verification.
	o Analysis of spectrum of sodium spills and behavior of cell liners.

*Letter, W. P. Gammill to L. W. Caffey, dated November 9, 1978; Reference 4.1.

Table 4.1 Summary of outstanding technical issues related to CRBR containment analysis (Cont.)

Technical Issue	Comments
Containment System Design (cont.)	
Accommodation of meltdown	o Containment integrity must be provided for 24 hours following CDA.
	o Substantiation of the assumptions regarding core debris-material interactions.
	o Evolution and control of H_2 - auto-ignition claim requires further substantiation.
	o Concrete structural analyses.
	o Dose mitigation features.
	o Interaction of TMBDB features with ESF features.
	o Revised TMBDB report review.

Table 4.2 Summary of the Project's Position on the NRC staff review*

Technical Issue	Comments
Large Sodium Releases in S/G Building.	o PSAR will be updated
Containment System Design	
Large sodium releases and DBAs	o Project considers selected DBA to be conservative.
	o Features have been included to ensure protection against DBA.
	o Commitment to mitigate hypothetical accidents beyond design basis.
	o Equipment survival in hostile environments to be in TMBDB report.
Accommodation of meltdown	o NRC concerns will be addressed in TMBDB report.

*Letter to H. R. Denton, Project Evaluation of the NRC Staff Review of CRBRP," dated March 3, 1979.

4.2 Containment System Design

Issues related to Containment System Design were subdivided into Large Sodium Releases, Design Basis Accidents (DBAs), and the Accommodation of a Core-meltdown (refer to Table 4.1). There are a number of issues which are of concern under all of the above classifications. For example, the evolution and control of H_2 could be of concern in both DBA analysis and in the accommodation of accidents beyond DBA. Also the performance of the cell liners is crucial for DBAs and accidents beyond DBA. Indeed the final selection of a DBA will influence the extent to which some of the issues included in Table 4.1 need to be considered. Therefore, we have selected major issues in Table 4.1, which we consider to be still unresolved at this stage. We discuss these issues in the following sections.

4.2.1 Design Basis Accidents

At the termination of CRBR licensing, the adequacy of the DBA analysis was in question (Ref. 4.1). It remains for the NRC staff to identify an adequate DBA. A BNL analysis (Ref. 4.3) of the DBA proposed before licensing termination indicated some divergence from the CRBR Project's analysis. This was mostly due to assumed sodium-concrete reaction rates and to differences in heat transfer rates and choices of thermal models (thermal equilibrium model). These differences should be addressed and factored into the analysis of the selected DBA.

4.2.2 Accommodation of a Meltdown

All of the outstanding issues (refer to Table 4.1) associated with accommodation of a meltdown are addressed by the CRBR Project in their TMBDB report (Ref. 4.4). In order to establish the adequacy of thermal margins beyond the design base, a careful review of Reference 4.4 is necessary. At the suspension of the CRBR licensing effort, an assessment had been performed and a report (Ref. 4.3) written by BNL regarding the DBA analysis in CRBR. An effort was initiated to analyze the containment response to a core meltdown accident but the work was terminated. It is, therefore, recommended that this effort be resumed.

We have had only a limited review of Reference 4.4 at BNL, but it appears that issues still remain, which require further resolution. In the following

sections we discuss some of these issues, which we found during our preliminary review of Reference 4.4. More issues may be found as part of the detailed evaluation of Reference 4.4

4.2.3 Sodium-Concrete Reaction Rates

This issue has not been resolved in a consistent or satisfactory manner. The question of penetration rate as well as the ultimate depth of penetration are still open. Sandia experimental results tend to favor more rapid penetration rates (i.e., 4 to 6 in/hr) with (under certain conditions) no reaction product inhibition or termination of the reaction front propagation. These results differ from those obtained in the HEDL experiments which imply a reaction rate of 1/2-1 in/hr and a maximum penetration of 12 inches. We consider that the parametric study performed in the TMBDB report (Ref. 4.4) should as a minimum be consistent with the penetration rates used for the analysis of FFTF core meltdown accidents (Ref. 5) (6 in/hr for 12 in was used for FFTF HEDL-TC-1175).

The sodium concrete reaction rate strongly impacts the time of incipient sodium boiling in the reactor cavity and this, in turn, is the primary containment pressurization mechanism. Since the energy supplied to the sodium pool by the sodium concrete reaction rate may be comparable to the decay heat, it is a crucial parameter for predicting whether or not the 24 hour non-venting criterion can be met.

4.2.4 Availability of Containment Passive Heat Sinks

The CRBR Project's analysis of the TMBDB (Ref. 4.4) relies heavily upon the availability of the passive heat sinks in the containment to meet the 24 hour venting criterion. Transfer of heat to these existing structures is critical if full credit is to be taken for their heat absorbing capacity. The presence, in the containment building atmosphere of enormous quantities of aerosols (mostly Na_2O and NaOH) subsequent to incipient sodium pool boiling, and the accompanying plate-out of these aerosols on all the available surfaces in the RCB will certainly inhibit the flow of heat by the normal transfer mechanism to the existing passive heat sinks. Thick layers of plated materials having both high porosity and low heat transfer properties will effectively insulate and delay the transfer of energy to the available structures. This mechanism is apparently not considered in the TMBDB analysis (Ref. 4.4). An assessment should

therefore be made of the degree to which this mechanism may negatively effect the capacity to contain the post-melt-through environment for the prescribed period.

4.2.5 Debris Bed Coolability

The redesigned CRBR core is sufficiently different to warrant not only a recalculation of the coolability of sodium flooded debris beds using an average decay heat level, but one which accounts for the heterogeneous distribution of the enrichment zones. A scenario can be visualized in which selective melting of the highly enriched zones of the core occurs preferentially, resulting in a concentration of decay heat in a smaller quantity of debris and perhaps in a more rapid penetration of the vessel head.

New estimates should also be made of the amount of upward (primary piping and upper plenum) and downward fuel relocation in terms of the new heterogeneous core enrichment configuration. Possible stratification of enrichment zones in the debris may affect the coolability of debris beds both in-vessel and ex-vessel, but the largest effect would probably be in-vessel.

In a BNL assessment (Reference 4.10) of thermal margins beyond DBA for FFTF we assumed that the core debris did not form a sodium flooded debris bed. We therefore assumed that the core debris could directly attack the reactor cavity concrete even with a large quantity of sodium also in the cavity. This assumption was an important consideration for FFTF because of the presence of the unlined subcavity beneath the reactor cavity. We found that the integrity of the floor between the cavity and the subcavity was crucial to our analysis of containment pressurization. The noncoolable debris bed assumption may not be such an important consideration for CRBR (no unlined subcavity) but the consequences of such an assumption should be addressed. No such assumption is at present made in the TMBDB report (Reference 4.4), which is again not consistent with the analysis (Reference 4.5) that has already been performed for FFTF.

4.2.6 Hydrogen Generation and Auto Ignition

At the point when CRBR licensing was suspended, the applicability of the experimental data (Ref. 4.11) and extrapolation of that data had been called into question (Ref. 4.1). In particular, at lower temperatures, entrainment of large quantities of sodium droplet aerosols were required to assure a continuous

hydrogen combustion and avoid an accumulation of H_2 in the containment building. Whether this can be guaranteed has not been established.

4.2.7 Mitigating Features

Independent review of the containment response to mitigating features such as vented-filtered systems has not yet been performed. In view of the possibility of more rapid sodium-concrete penetration rates and the subsequent increased density of aerosols in the RCB, an independent assessment should be made of the design and capability of the proposed scrubbing system to adequately remove the particulate and volatile radioactive materials. Also, an assessment is required to determine the efficiency of the annular cooling system in the presence of inhibiting aerosol plate-out. In addition, the question of the feasibility of an ex-vessel core catcher and/or Na K cooling system may again emerge.

4.2.8 Potential for Basemat Penetration

The analysis performed in the TMBDB report (Reference 4.4) does not reflect the latest experimental and analytical models available (Reference 4.12) to predict the potential for basemat penetration by thermal attack of the core debris. In Section 4.2.5 we note the potential for a noncoolable debris bed to form and concrete penetration to occur even in the presence of sodium. The assumption in the TMBDB report (Reference 4.4) that no penetration of the core debris into the concrete can occur until all the sodium is boiled dry underestimates the potential threat to basemat integrity. Also, the model used (TRUMP) cannot model the potential for a solid core debris to penetrate concrete. SANDIA experiments have shown that molten core debris rapidly forms crusts, which prevent dilution of the core debris (and hence dilution of the volumetric heat source) but allows concrete penetration. The analysis of basemat penetration and the TMBDB report (Reference 4.4) is inadequate and should be improved to include currently available models.

4.2.9 Model Validation

Although the CRBR Project claims that the CACECO code has been fully validated, we suggested that the code be reviewed by an independent (from the writers of the code) organization. A code with the degree of complexity contained in CACECO requires more than experimental validation of individual models. An overall evaluation of the code's energy and mass balances and the

way in which the individual models mesh with the containment response models are extremely important. In addition to the obvious checks against simply hand calculations, the code results should be compared to those obtained by alternate codes such as the BNL developed CONAN code (Ref. 4.6). Care must also be taken in accepting, as experimentally validated, models which were obtained by extrapolation of non-prototypical experiments. These types of issues should be considered in any "independent" assessment.

A number of model deficiencies were detected and reported by BNL (Ref. 4.7) before the CRBR licensing review was suspended. Some of these problems may still remain, e.g., the assumption of thermal equilibrium between the containment atmosphere and the sodium pool in the reactor cavity. A careful inquiry should be made into the areas that have previously been identified as problematical as well as a thorough re-evaluation of the latest version of the code.

Validation of the sodium fire codes has been undertaken at BNL. The NACOM code (Reference 4.8) was developed at BNL to analyze sodium spray fires. NACOM was compared with the SPRAY code (Reference 4.9) (used by the Project in their analysis of sodium spray fires) and reasonable agreement found. However, extrapolation of both of the above models, which are based on single droplet burning models, to the turbulent conditions that occur within the spray core still require resolution. Comparison of the predictions of the above models with the more advanced spray model (SOMIX) is needed.

4.3 Radiological Consequences

The radiological consequences analysis for the revised CRBR is based on the homogeneous core configuration. The claim is made in this analysis that, even with the higher plutonium inventory in the heterogeneous core, its isotopic makeup is such that the radiological impact due to plutonium aerosols is lower. In view of this implied sensitivity to plutonium isotopic content, a comparatively more severe situation might arise if the heterogeneous core was fueled with homogeneous core plutonium. Table 4-6 of the CRBRP-3 (Ref. 4.4) seems to indicate that the homogeneous core is fueled with LWR grade plutonium, while the heterogeneous core is fueled with FFTF grade plutonium. Since both grades were considered for the homogeneous core, it is not unreasonable to consider both grades for the heterogeneous core. A further point, of lesser importance, but which should be considered is the different burnup expected for the two cores.

At equilibrium the burnup in the heterogeneous core is expected to be approximately 25% lower than in the homogeneous core. This change in burnup will affect the fission product inventory which in turn will have an effect on the radiological consequence.

REFERENCES

4.1. Letter, W. P. Gammill to L. W. Caffey, dated November 9, 1978.

4.2. Letter to H. R. Denton, "Project Evaluation of the NRC Staff Review of CRBRP," dated March 5, 1979.

4.3. R. D. Gasser, S. S. Tsai, D. C. Albright and W. T. Pratt, "Containment Design Basis Accident Analysis for the Clinch River Breeder Reactor," BNL-NUREG-25561, January 1979.

4.4. "CRBRP-3, Hypothetical Core Disruptive Accident Considerations in CRBRP, Volume 2, Assessment of Thermal Margin Beyond the Design Base," Issued by the CRBRP Project Office, March 1980.

4.5. R. J. Jensen et al., "Further Studies of Containment Margins in FFTF for Postulated Failure of In-Vessel Post-Accident Heat Removal," HEDL-TC-1175, July 1978.

4.6. R. D. Gasser, "CONAN: An LMFBR Containment Response Computer Code," BNL-NUREG-CR-1355, November 1979.

4.7. S. S. Tsai, R. D. Gasser and W. T. Pratt, "Containment Design Basis Accident for LMFBRs: Review of Methods," BNL-NUREG-23221 September 1976.

4.8. S. S. Tsai, "The NACOM Code for Analysis of Postulated Sodium Spray Fires in LMFBRs," BNL-NUREG-51180, NUREG-CR-1405, March 1980.

4.9. P. R. Shire, "SPRAY Code User's Report," HEDL-TME 76-94, March 1977.

4.10. R. D. Gasser and W. T. Pratt, "Containment Response to Postulated Core Meltdown Accidents in the FFTF," BNL-NUREG-24141-R, August 1978.

4.11. R. W. Wierman and R. K. Hilliard, "Experimental Study of Hydrogen Formation and Recombination under Postulated LMFBR Accident Conditions," Hanford Engineering Development Laboratory, December 1976.

4.12. W. T. Pratt, R. D. Gasser and R. A. Bari, "Potential Influence of Core/Concrete Interactions on PWR Containment Pressurization," Trans. Am. Nucl. Soc. 39, 609 (1981).

5. RELIABILITY ANALYSIS

This section reports on issues related to reliability analysis of the CRBR. Section 5.1 provides some comments on the risk study conducted by the CRBR Project. Sections 5.2 and 5.3 contain updates of the issues related to the reliability of the shutdown system and shutdown heat removal system, respectively.

5.1 Risk Assessment

In March 1977, the CRBRP risk assessment report was issued (Ref. 5.1). The report states that its objective was to provide a realistic evaluation of the risk to the public from the CRBR plant. A methodology similar to the one used in the Reactor Safety Study (Ref. 5.2) was used. Since this risk study was issued near the end of CRBR licensing activities, it has not been reviewed and evaluated in depth by BNL. However, BNL has performed extensive reliability analyses on the systems supporting the two main safety functions, namely, the reactor shutdown function and the shutdown heat removal function. The major issues that these studies have identified are presented in the corresponding sections of this report (See Sections 5.2 and 5.3). This section contains general comments on the overall risk assessment methodology and results.

The methodology for the assessment of the CRBRP risk can be divided into the following steps.

1. Accident Sequence Definition and Quantification.
2. Accident Analysis and Evaluation.
3. Consequence Modeling.
4. Risk Evaluation.

The following issues are pertinent to the completeness and validity of the analysis.

- o The risk study has been performed for the 1976 CRBRP design. Before the results of this study can be extrapolated to the latest CRBRP design, a careful examination of the potential impact of the design changes in the various parts of the analysis should be performed.

- o The handling of the potential dependences among systems should be reviewed and evaluated. Of particular interest are interactions between key systems that can contribute (through their failure) to the frequency of accident initiators or increase the failure probability of a safety function. Both a qualitative analysis should be performed (having as its objective the identification of existing dependences) and a quantitative assessment of the impact of the potential dependences on the probabilities of the various accident sequences.
- o The coupling of the reactor shutdown and the shutdown heat removal event trees with the containment event tree through the core response and pressure vessel failure model is highly subjective. A more detailed modeling of the dependences between the failures of systems, the response of the core, the pressure vessel failure modes and the containment failure modes (Ref. 5.3) is required.
- o The numerical results are point estimates obtained by inputting the median values of the various parameters. The use of median values in the input does not yield the median value of the final result. Yet, the results of this study are compared with the median results of WASH-1400. Furthermore, even if the exact median of the CRBRP risk were presented, the comparison of two median is not adequate since the median does not contain any measure of the spread of the results. Hence, an uncertainty analysis should be performed and the total range of the CRBRP risk should be estimated and presented.

5.2 Reliability Analyses of the Shutdown Systems

5.2.1 Background

BNL has performed an extensive analysis of the reliability of the shutdown systems of the CRBRP, as well as, their contribution to the probability of Loss of Core Coolable Geometry (Ref. 5.3).

The BNL study has employed the most advanced methodology for analyzing the reliability of large systems, namely, the modeling of the stochastic behavior of the systems by Markov processes. This approach allows for a more realistic modeling of the system that avoids unnecessary conservative assumptions.

The BNL analysis includes dependences between the components of the electrical subsystem of the primary and secondary systems as well as between the components of the mechanical subsystems. Interdependences between the unavailability of the systems and the occurrence of the transients have been included, resulting in the removal of unnecessary conservatisms. Thus, whether the shutdown system responds successfully to a challenge depends on its state and on the type of transient, e.g., for limited response transients only the mechanical subsystem is required to respond. Successful challenges reveal partial failures of the system (e.g., one channel down out of three) and result in system renewal. Finally, the time-dependent nature of the model accurately calculates the probability of loss of core coolable geometry by avoiding the double counting introduced by the approximation:

failure probability = (frequency of challenge) x (system unavailability)

The BNL model allows for inspection and maintenance procedures that depend on the state of the system and include the possibility of human errors. Inspection of the protective function networks of the primary and secondary systems are staggered to minimize the effect of human errors. Two inspection policies are possible depending on whether there are tripped channels in one or both shutdown systems.

Uncertainties are expressed by assuming the failure rates, the repair rates and all other input variables as random variables, distributed according to given probability density functions. The distributions are such that the upper 90% percentile of the input parameter is one order of magnitude higher than the lower 90% percentile.

To assess the sensitivity of the probability of loss of core coolable geometry to interdependences and human errors, four special cases were examined: (a) the failure rates of the components are completely independent and the inspection is perfect, i.e., every four weeks the electrical subsystems are completely renewed; (b) the failure rates and the states of the components are completely independent, and the inspection is perfect; (c) the failure rates of the components are completely independent, but inspection is imperfect, i.e., human errors are possible; and (d) interdependences exist, and the inspection is imperfect. For these cases, medians, 90% probability bands, and point estimates of the RSS failure probability per year are listed in Table 5.1. Table 5.1

Table 5.1 Point Estimate, Median, and 90% Probability Band of the Failure Probability per Year under Various Assumptions for the Shutdown System of the CRBR

(Point estimate is the value of the failure probability obtained when the input variables are assumed to be fixed at their means.)

Case	Assumption	Failure Probability			
		Point Estimate	Median	0.05 Percentile	0.95 Percentile
a	No dependences, perfect inspection	8×10^{-10}	1×10^{-9}	4×10^{-11}	2×10^{-8}
b	Dependences, perfect inspection	2×10^{-7}	8×10^{-8}	1×10^{-9}	6×10^{-6}
c	No dependences, imperfect inspection	4×10^{-7}	2×10^{-7}	1×10^{-8}	5×10^{-6}
d	Dependences, imperfect inspection	2×10^{-6}	2×10^{-6}	2×10^{-7}	2×10^{-5}

5.3 Shutdown Heat Removal System Reliability

Work on the reliability of the Shutdown Heat Removal System (SHRS) was performed at BNL prior to cessation of licensing activities for the CRBR. This work, as pertains to the most recent design of the SHRS in existence at the time of the cessation of licensing activities, is summarized in Reference 5.4. The design considered was capable (under certain circumstances) of using the Direct Heat Removal Service (DHRS) to successfully remove decay heat even if heat transport through the main heat transport system is cut off at the intermediate heat exchangers. Consideration was given only to the so-called baseline design case in which operation on less than three loops was not permitted. Pony motors in one primary loop and its associated intermediate loop were assumed to be capable of being powered by a battery with a two hour capacity. The analysis given in Reference 5.4 confined itself to the initiator considered most important, the loss of offsite power initiator. It was assumed in Reference 5.4 that natural circulation in the steam generator loop is sufficient for SHRS mission success, or else that the recirculation pump in the steam generator loop 2 may be operated by a battery.

It is really not clear whether one requires both motor-driven pumps in the auxiliary feedwater system (AFWS) or only one, for AFWS mission success. The CRBRP risk assessment report (see Ref. 5.1, p. 10-24, section 10.3.5) assumed one motor-driven pump was sufficient. However, according to the CRBR PSAR (Ref. 5.5, p. 5.6-1C, Amendment 43 dated January 1978) "each motor-driven pump is half-size, such that the combination can supply the required flow to all three loops." This statement could be interpreted as meaning that both motor-driven pumps are required for AFWS mission success. This point must be clarified. Reference 5.4 presents results for both cases.

Results were presented a Reference 5.4 for two different sets of diesel generator failure probabilities. One of these was the Reactor Safety Study (Ref. 5.2) diesel generator data, where the probability of one diesel generator failing to start (and take on load) was .03 per demand, and the probability that both fail to start (or take up load) is .01. The second set of diesel generator failures probabilities had a .01 per demand probability of failure for one diesel generator, and a .001 per demand probability for the common mode failure of

both diesel generators. The Reactor Safety Study data for common mode failure of the diesel generators is probably somewhat pessimistic, but the second set of diesel generator failure probabilities are optimistic, when compared to present experience.

Reference 5.4 considered the SHRS reliability for two different probabilities for restoration of offsite power within two hours: .1 and .01. The significance of the two hours is that the batteries on the pump motors will be drained in two hours. (Consideration should also be given to the effects of loss of D.C. power for control and instrumentation for an extended loss of all AC electric power). The probability of .1 for restoration of offsite power in two hours was taken from the Reactor Safety Study (see Figure III.6.4 in Appendix III of Reference 5.2) and came from Bonnsville power grid data. Actually, even the probability of .1 for restoration of offsite power in two hours may be optimistic. The reason is that the Bonnsville power data was for restoration of single line outages, and the distribution function for the time to repair of a single line may depend on whether the outages considered are those which cause loss of only a single line or those which cause total loss of offsite power. There are indications in the data collected by Ray Scholl (Ref. 5.6) that this is the case. The frequency of loss of offsite power used in Reference 5.4 was .2/year; TVA grid specific values could be better.

The results obtained in Reference 5.4 are summarized in Table 5.1 of that reference and are reproduced here in Table 5.2.

Since the work performed in Reference 5.4, additional work has been performed on the SHRS system reliability by Jamali, as part of his thesis. This work is summarized in Reference 5.7, by Jamali and Kerr. The results obtained by Jamali (see Table 5.3) give much higher unavailabilities for the SHRS than were obtained in Reference 5.4. Some of the reasons for this are:

1. Reference 5.7 uses, for a loss of main feedwater frequency 2/year, which is reasonable. However, no credit is given for timely restoration of main feedwater. The Reactor Safety Study assumed only 1% of all loss of main feedwater transients could not be repaired in 1/2 to 1 hour (see Appendix V, p. 37, of Reference 5.2).

2. Reference 5.7 gave no credit for the DHRS as an alternative path for decay heat removal, independent of the AFWS.

3. Reference 5.7 assumed an AC dependency of the steam-turbine driven pumps in the AFWS, because of its dependence on the ventilation system. The ability of the steam-turbine pump in the AFWS to operate in the absence of AC power should be confirmed.

4. Reference 5.7 included many types of common mode failures by a variant of the β-factor method (Ref. 5.8) where factors $\beta_{2/1}$ for the probability of failure of a second component given failure of the first, and $\beta_{3/2}$ for the probability of failure of the third component given failure of the first two, are obtained. However, many maintenance-caused failures of, e.g., valves are assumed coupled, not only valves performing similar functions in redundant trains of a system. This very likely introduces a conservative bias in the results, since the probability a second valve fails given the first loss depends on how similar the valves are in function and in location, and in the maintenance they are subjected to. The precise manner of incorporating the somewhat weaker dependence of valves performing dissimilar functions is of interest, however.

Jamali has commented on some nonconservative aspects of his analysis, including neglect of certain types of electrical common mode failures, neglect of some coupled human failures, and fault-tree truncation.

Table 5.2 Failure probability, per year, of the SHRS due to LOSP shutdown initiating event.

	Failure Probability, Per Year, of SHRS (Due to LOSP)			
	Case $\delta_1 \alpha_1$	Case $\delta_1 \alpha_2$	Case $\delta_2 \alpha_1$	Case $\delta_2 \alpha_2$
(a) No credit for natural circulation in sodium loops after two hours from shutdown; $p_2 = .1$	3×10^{-4}/yr	2×10^{-4}/yr	2×10^{-5}/yr	2×10^{-5}/yr
(b) Like (a), but $p_2 = .01$	9×10^{-5}/yr	5×10^{-5}/yr	6×10^{-6}/yr	6×10^{-6}/yr
(c) Credit for natural circulation in sodium loops after two hours from shutdown, or negligible probability that off-site power will not be restored in two hours	7×10^{-5}/yr	3×10^{-5}/yr	4×10^{-6}/yr	4×10^{-6}/yr

p_2 = Probability off-site power is not restored by two hours after shutdown.

Key to Cases:

δ_1 : Reactor Safety Study[7] diesel generator failure data. The probability one diesel generator fails to start is .03; the probability both fail to start is .01.

δ_2 : The probability one diesel generator fails to start is .01; the probability both fail to start is .001.

α_1 : One motor-driven pump loop in the AFWS is insufficient for AFWS mission success.

α_2 : One motor-driven pump loop in the AFWS is sufficient for AFWS mission success.

Table 5.3 Summary of results for failure of short-term forced circulation.

Initiator	Median Unavailability	Failure Probability per year	Major Sequences	Critical Component Failures
Normal Shutdown	2.1×10^{-4}	3×10^{-3}	• Dependent Failures of Pony Motors • Loss of all AC Power • Dependent Pony Motor Failures Combined with a Maintenance Fault • Dependent Failures of Valves	• Pony Motors • Faults in Hydrogen Detection System • Turbine Driven Auxiliary Feed-water Pump
Loss of Offsite Power (LOSP)	4.4×10^{-2}	9×10^{-3}	• Dependent Failures of Diesel Generators • Loss of Chilled Water Sys. • Dependent Failures of Valves	• Turbine Driven Auxiliary Feed-water pump • Diesel Generators • Normal and Emer-gency Chilled Water Systems
Rupture of Main Feedwater Header (Loss of Main Feedwater System)	7.2×10^{-3}	1.4×10^{-2}	• Dependent Failures of Valves in Auxiliary Feedwater System • Dependent Failures of Diesel Generators Combined with Failure of Normal Chilled Water System • Loss of DC Power System, A or B Combined with Failure of Normal Chilled Water System	• Turbine Driven Auxiliary Feed-water pump • Maintenance Fault on Circuit Breakers and Gas Coolers • Normal Chilled Water System

REFERENCES

5.1 CRBRP Risk Assessment Report, CRBRP-1, March (1977).

5.2 Reactor Safety Study, "An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants," WASH-1400 (NUREG-75/014), U.S. Nuclear Regulatory Commission, October (1975).*

5.3 I. A. Papazoglou and S.E.P. Gyftopoulos, "Markovian Reliability Analysis Under Uncertainty with an Application on the Shutdown System of the Clinch River Breeder Reactor," NUREG/CR-0405, BNL-NUREG-50804, September (1978).-

5.4 A. J. Buslik, I. A. Papazoglou, and R. A. Bari, "Reliability of the CRBRP Shutdown Heat Removal System," Paper V.2 in the Proc. of the ANS/ENS/OECD Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, May 8-10, 1978, Los Angeles, California.

5.5 Clinch River Breeder Reactor Plant, Preliminary Safety Analysis Report, Project Management Corporation.

5.6 "Loss of Offsite Power, Survey Status Report, Revision 3," prepared by Raymond F. Scholl, Jr., enclosure to a Memo from D. Crutchfield, Chief, Operating Reactors Branch #5, Division of Licensing, USNRC to D. Crutchfield, Acting Chief, Systematic Evaluation Program Branch, Division of Licensing, dated September 25, (1980).

5.7 K. Jamali and W. Kerr, "Probabilistic Methods for Accident Progression Analysis," paper presented at the International ANS/ENS Topical Meeting on Probabilistic Risk Assessment, September 20-24, 1981, at Port Chester, New York (proceedings in print).

5.8 K. N. Fleming and P. H. Raabe, "A Comparison of Three Methods for the Quantitative Analysis of Common Cause Failures," Paper X.3 in the Proc. of the ANS/ENS/OECD Topical Meeting on Probabilistic Analysis of Nuclear Reactor Safety, May 8-10, 1978, Los Angeles, California.

5.9 WARD-D-0118, "Reliability Assessment of CRBR Reactor Shutdown System," Westinghouse Electric Corporation, November (1975).

6. LOSS OF HEAT SINK

This accident scenario differs from other core disruptive accident scenarios in that it is assumed that the reactor is scrammed and thus, initially the only heat source is the decay heat of the fuel. Furthermore, it is assumed that no heat sinks exist and thus eventially the coolant boils, cladding and subassembly walls melt and relocate forming blockages and eventually the fuel compacts. This fuel compaction may lead to a critical configuration in a fast reactor. At this point the heating is not only due to decay heat but also due to fission heating. In an analysis of such a scenario for the homogeneous core-CRBR it was found (Ref. 6.1) that all relocation patterns of the fuel and control material led to the formation of a critical configuration. Figure 6.1 shows a contour plot of active core height and control subassembly height for multiplication factor (k_e)=1. It is seen that a large area of the map is excluded due to compaction limits. Furthermore, it is seen that no compaction could reach the compaction limit without first passing through the k_e=1 contour.

In the analysis mentioned above, the natural circulation of the coolant was not determined in great detail. It is clear from an analysis carried out subsequently that considerable recirculation flow loops are set up (Ref. 6.2) (up hot subassemblies and down cooler subassemblies) in cores experiencing such an accident scenario. This phenomena is illustrated on Figures 6.2, 6.3 and 6.4 which show the channel assignments, flow fraction and maximum sodium temperatures of a heterogeneous core following a LOHS immediately after scram (power is assumed to be 6%). From Figure 6.3 it can be seen that there is substantial reverse flow in channel 19 (control subassembly) and a low flow in channels 1, 2 and 9. Figure 6.4 shows that the sodium reaches boiling temperatures in channels 1 and 8 approximately 90 seconds after the accident is initiated. These channels do not represent the hottest or coolest subassemblies, but some intermediate subassembly, whose flow rate is not great enough to provide sufficient cooling. The spatial incoherency in the onset of coolant boiling implies an incoherency in the location of core damage, (steel and fuel melting and relocating) which affects the progression of the accident in both space and time. If, for example, a highly incoherent situation arises, the possibility of a recriticality could be avoided if the fuel has a chance to drain away in selected position before

the remainder of the core compacts. Furthermore, the amount and timing of internal blanket material relocating could affect the progression of the accident, these being mostly composed of ^{238}U and would thus tend to retard the formation of a critical mass.

In the event that the fuel drains out of the core region, thus reducing the probability of assembling a critical mass in this region, the possibility of a critical mass forming in the inlet plenum region has to be addressed. Studies (Ref. 6.3) of this problem for the homogeneous core have indicated that if the fuel and steel are separated and collect in the bottom of the inlet plenum a critical mass results. In addition, most of the values of k_e determined for various configuration in this study have been determined to be close to .9 or larger. In view of the uncertainty (Ref. 6.4) connected with the determination of k_e for compacted cores, a value of $k_e \geq .9$ should be studied with great care.

In determining the formation of the blockages due to steel relocation, care should be taken to allow for the smaller pitch/diameter of 1.072 in the internal blankets as compared to a value of 1.24 in the driver fuel region. Clearly the formation of blockages in the internal blankets will be enhanced relative to the fuel regions. Finally, due to the incoherent nature of the boiling of coolant, melting and relocation of steel and fuel the neutronic calculations would require a space-time kinetic model with appropriate feedback coefficients (Doppler and volumetric change). Finally, the mechanical work that such an accident might imply would have to be determined based on some terminating scenario, i.e., upper axial blanket collapse, pressure driven compaction etc.

REFERENCES

6.1 R. A. Bari, H. Ludewig, W. T. Pratt, Y. H. Sun, "Accident Progression for a Loss-of-Heat-Sink with Scram in a Liquid Metal Cooled Fast Breeder Reactor," Nucl. Tech., 44, 357 (1979).

6.2 M. Khatib-Rahbar, J. G. Guppy, A. K. Agrawal, "Hypothetical Loss-of-Heat Sink and In-Vessel Natural Convection: Homogeneous and Heterogeneous Core Designs, Proc. Decay Heat Removal and Natural Convection in Fast Breeder Reactors," Hemisphere Publishing Corp. (1981).

6.3 A. Buslik, "Recriticality Potential at the CRBR Reactor Vessel Bottom," BNL/FRS 75-7, (1975).

6.4 D. Wade, "Critical Experiments on Severely Damaged Cores," Presentation to ACRS, September 27, (1978).

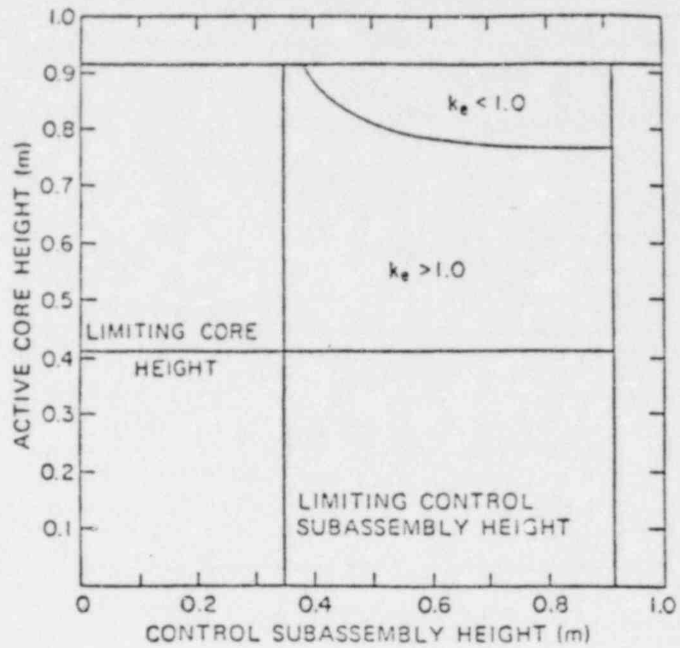


Figure 6.1 Contour of k_e=1.0.

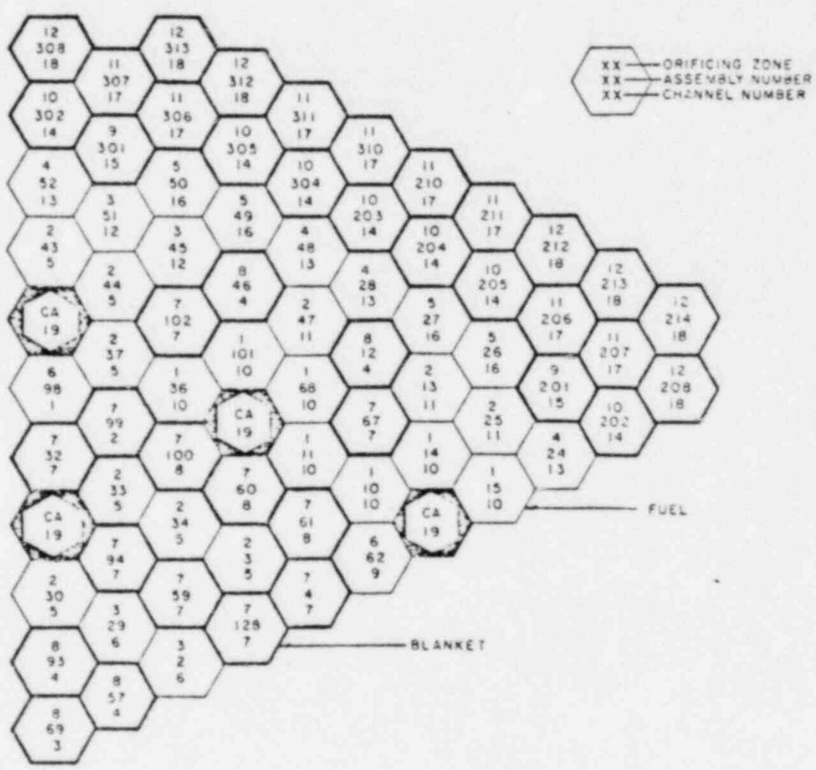


Figure 6.2 Channel assignments for thermal-hydraulic calculation.

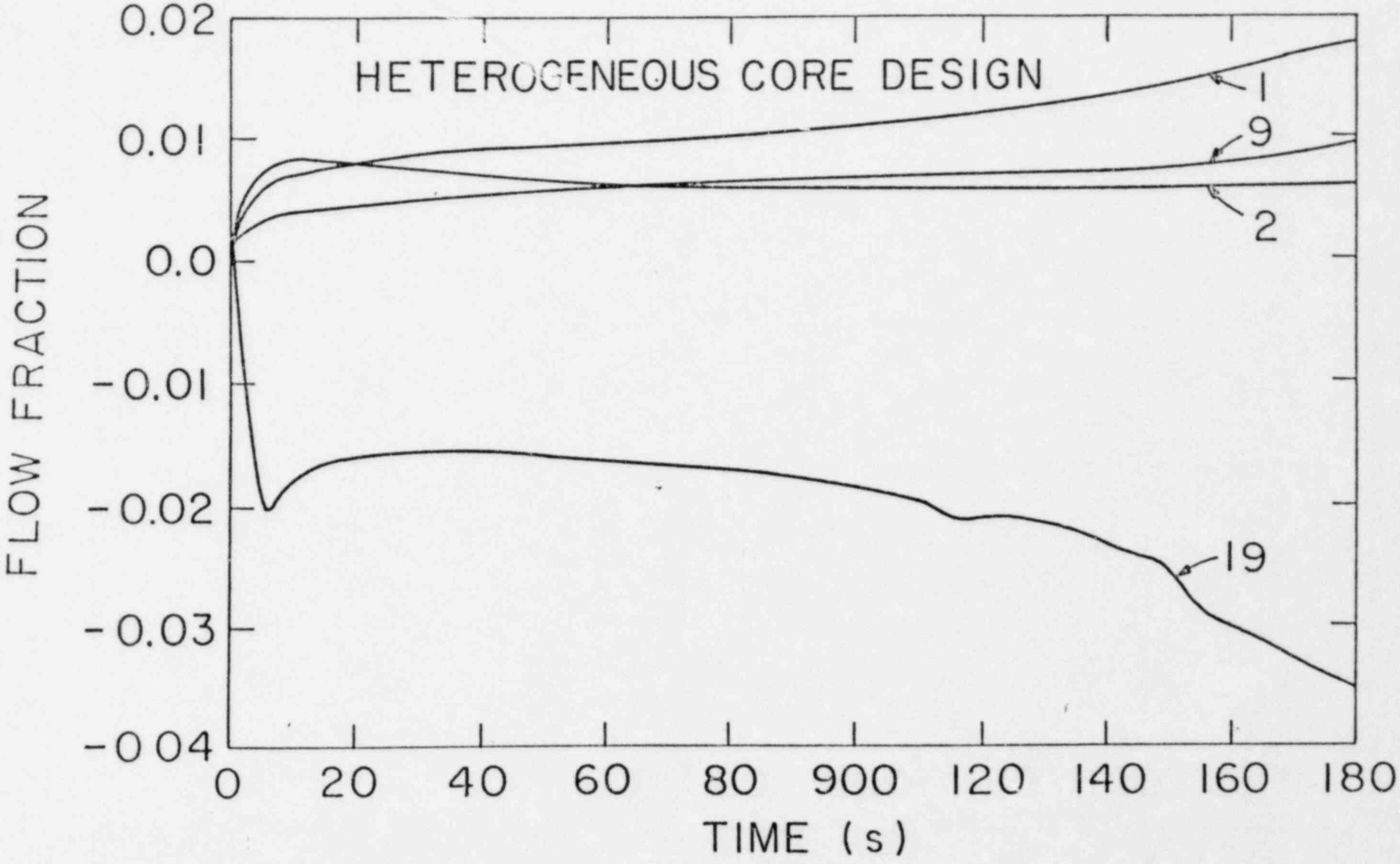


Figure 6.3 Flow fraction.

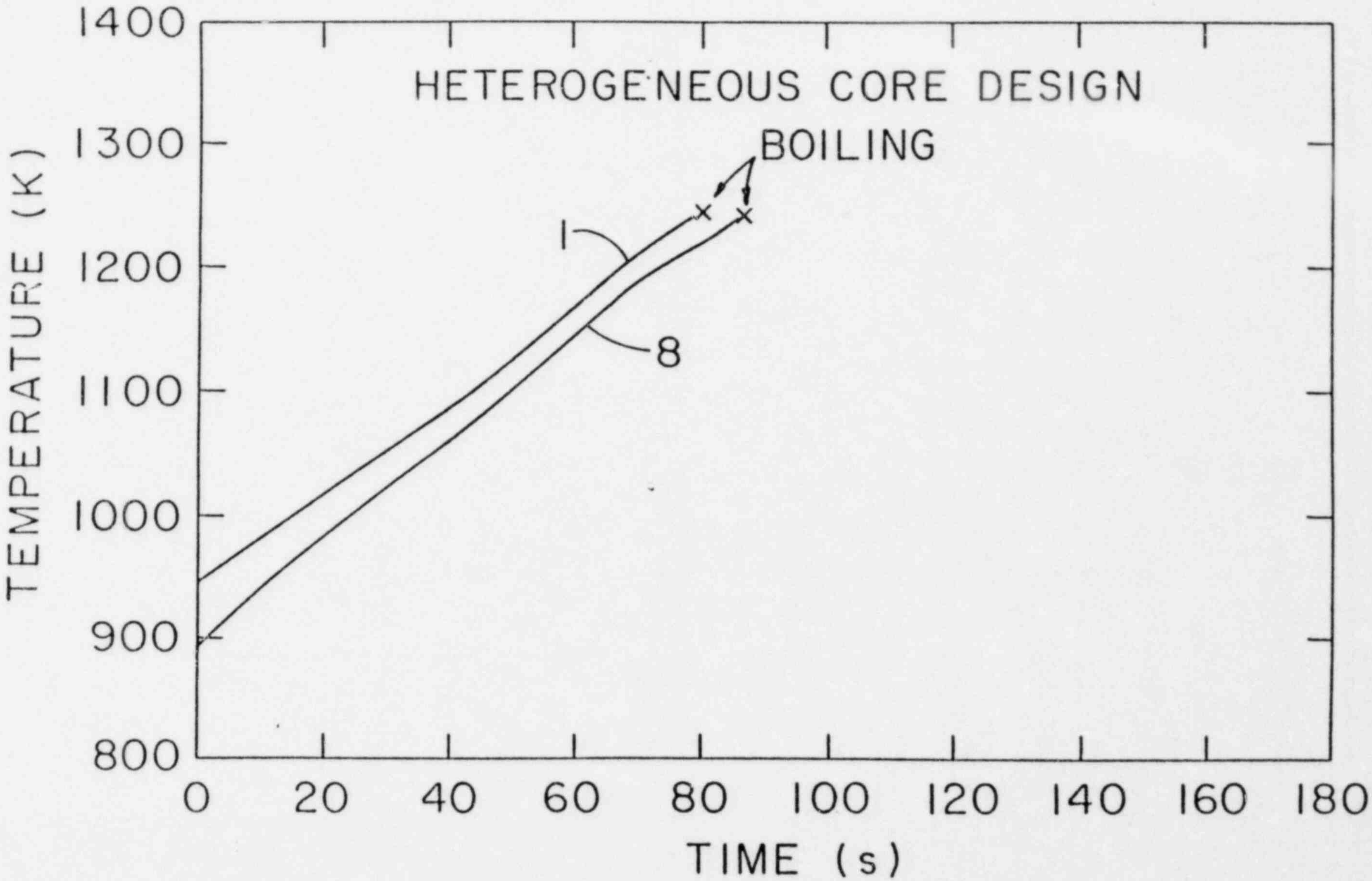


Figure 6.4 Sodium temperature.

7. REACTOR PHYSICS

The CRBR core design has been revised and the new layout is outlined in Amendment 51. Briefly, the core layout has been changed from a homogeneous core with two enrichment zones (inner core ~ 17%, outer core ~ 27%) to a heterogeneous core with a single driver fuel enrichment (32.8%) and approximately two internal blanket zones. These two designs are shown as Figures 7.1 and 7.2, which illustrate the two enrichment zones of the homogeneous core and internal blanket zones of the heterogeneous core, respectively. From a safety point of view the revised core design implies different reactivity feedback coefficients and thus different response to operational transients and behavior under accident conditions.

Table 7.1 compares the Doppler coefficient for the two cores, ignoring the radial blanket and the upper and lower axial blankets at BOC. From the table it can be seen that the major contribution to the Doppler coefficient in the heterogeneous core comes from the internal blanket while in the homogeneous core the major contribution comes from the inner core zone. Thus, although the total coefficient is higher in the case of the heterogeneous core the fuel region has a much lower coefficient than the homogeneous core which is all fuel. At BOC, this difference could be significant, since the internal blanket zones generate a relatively small fraction of the power and any power rise due to a transient will first occur in the fuel. At EOC the situation is not as clear since the power fraction generated by the internal blanket has increased and at this time some combination of the coefficients (fuel and internal blanket) will play a role. Table 2 shows the sodium void reactivity for the two cores at BOC1 and EOC4. This table shows that the total sodium void reactivity is higher for the heterogeneous core compared to the homogeneous core. However, in the case of the heterogeneous core the fuel and internal blankets contribute approximately equally to the total, while in the case of the homogeneous core the inner core plays a dominant role. It is seen that the sodium void reactivity associated with the fuel region in the heterogeneous core is much lower than that associated with the inner core region of the homogeneous core. In the event of an accident leading to sodium voiding, the heterogeneous core is expected to start voiding in the fuel zones at BOC, since the blankets are overcooled at this time. In the homogeneous core, the void generally starts in the inner core.

Thus, at BOC the initially added reactivity due to sodium voiding is much larger in the homogeneous core than in the heterogeneous core. At EOC the situation is not as clear as voiding could start in the internal blankets of the heterogeneous core (depending on the orificing) and then move to the fuel region thus changing the space-time dynamics of the power shape. The spatial voiding pattern is very important in determining the accident sequence.

The above discussion on feedback coefficients implies that a redetermination of these together with steady state power shapes, and a value of β_{eff} characteristic of this core is required. Furthermore, a determination of the steady state condition of the core would be required as a starting point for the analysis of transients and accident scenarios (scrammed or unscrammed). The possibility of a re-criticality during an accident in a fast reactor requires a steady state configuration even in a scrammed reactor.

The calculational capability to carry out such an analysis is in place at BNL. Table 7.3 is a tabulation of fast reactor physics codes available at BNL and their status as of November 1980. However, in order to have confidence in the adequacy of the analysis methods, they should be benchmarked against selected heterogeneous core critical experiments (ZPPR-7 and ZPPR-8).

Table 7.1 Doppler coefficient for BOC1 ($-T \frac{dk}{dT} \times 10^{-4}$).

Heterogeneous Core

Fuel	Internal Blanket	Total	
25.8	44.0	69.8	Flooded
16.6	35.4	52.0	Voided

Homogeneous Core

Inner Core	Outer Core	Total	
44.3	13.7	58.0	Flooded
23.1	8.1	31.2	Voided

Table 7.2 Sodium void reactivity ($).

Heterogeneous Core

	Fuel	Internal Blanket	Total
BOC1	1.51	1.4	2.9
EOC4	2.31	1.64	3.95

Homogeneous Core

	Inner Core	Outer Core	Total
BOC1	2.71	- 0.83	1.88
EOC4	3.55	- 0.21	3.34

Table 7.3 Fast reactor physics codes available at BNL.

Multigroup Cell Codes

MC^2-II	- Operational - Multigroup cross-section preparation code.
SDX	- Operational - Multigroup fast reactor cell code (slabs and cylinders)
RABBLE	- Operational - One-dimensional cell calculations for resonance range
NJOY	- Operational - Cross-section preparation code

Transport Theory

ANISN	- Operational - One-dimensional S_N code
TWOTRAN	- Operational - Two-dimensional S_N code
DOT 4.2 & 3.5	- Operational - Two-dimensional S_N code

Monte Carlo

MORSE	- Operational (needs modifications) - Monte Carlo code for reactor analysis, using multigroup library

Diffusion Theory

1-DX	- Operational - One-dimensional code used for cross-section preparation
SPHINX	- Operational - Similar to 1-DX, more flexible
SIZZLE	- Operational - One-dimensional burnup code (limited to 18 groups)
2DB	- Operational - Two-dimensional code with burnup option; several revisions available allowing for upscattering and multi-chain fission product models.

Table 7.3 (cont'd)

DIF-3D	-	Operational - Three-dimensional code
3DB	-	Operational - Three-dimensional code
PERT-V	-	Operational - Perturbation theory code, interfaces with 2-DB

Core Dynamics with Feedback

FX-2	-	Running - needs checking; two-dimensional space-time kinetics code

Fission Product Codes

CINDER	-	Operational - Study fission product models
TOAFEW	-	Operational - Collapses multigroup fission product library for use in CINDER, 1-DX or 2-DB

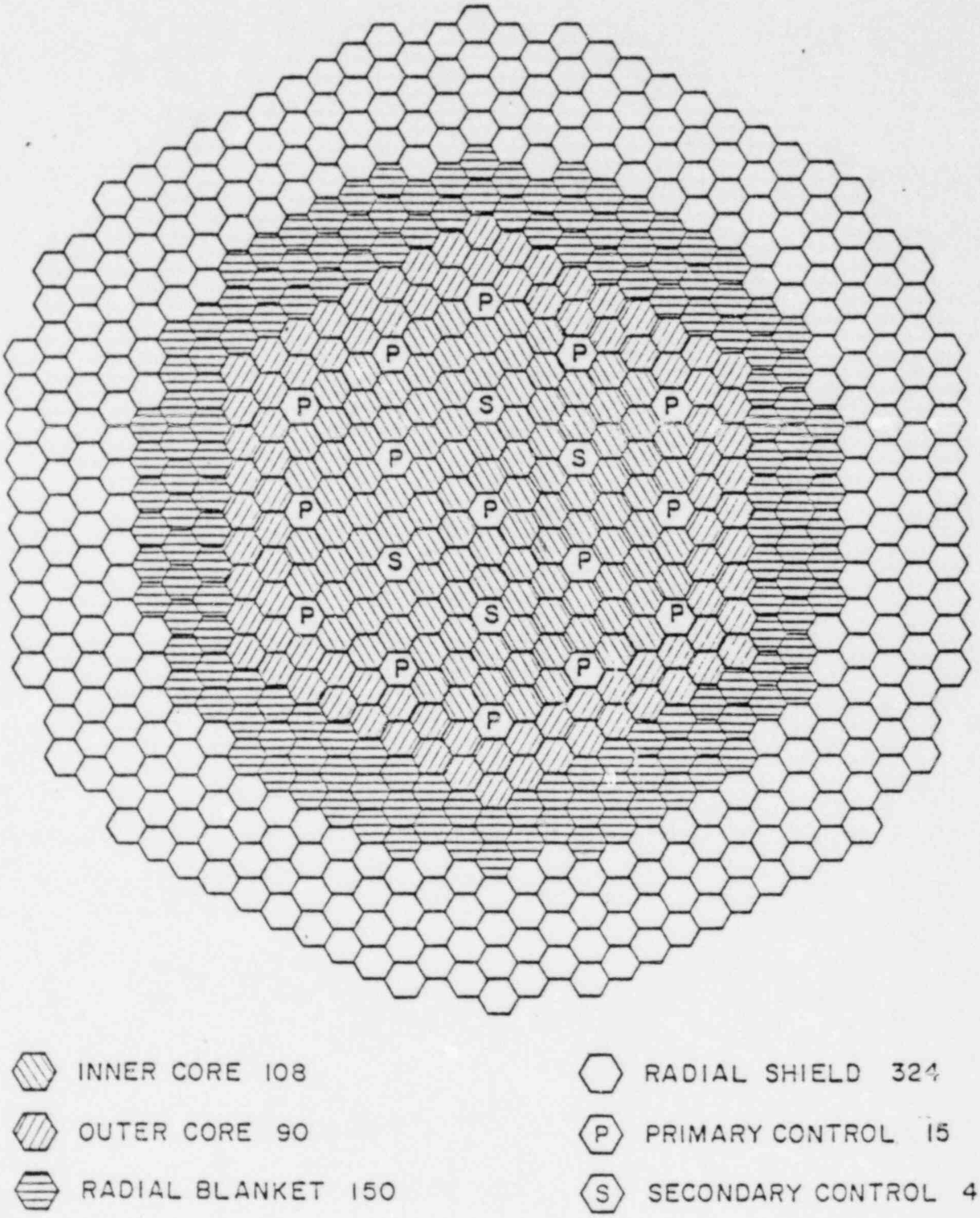


Figure 7.1 Homogeneous core layout.

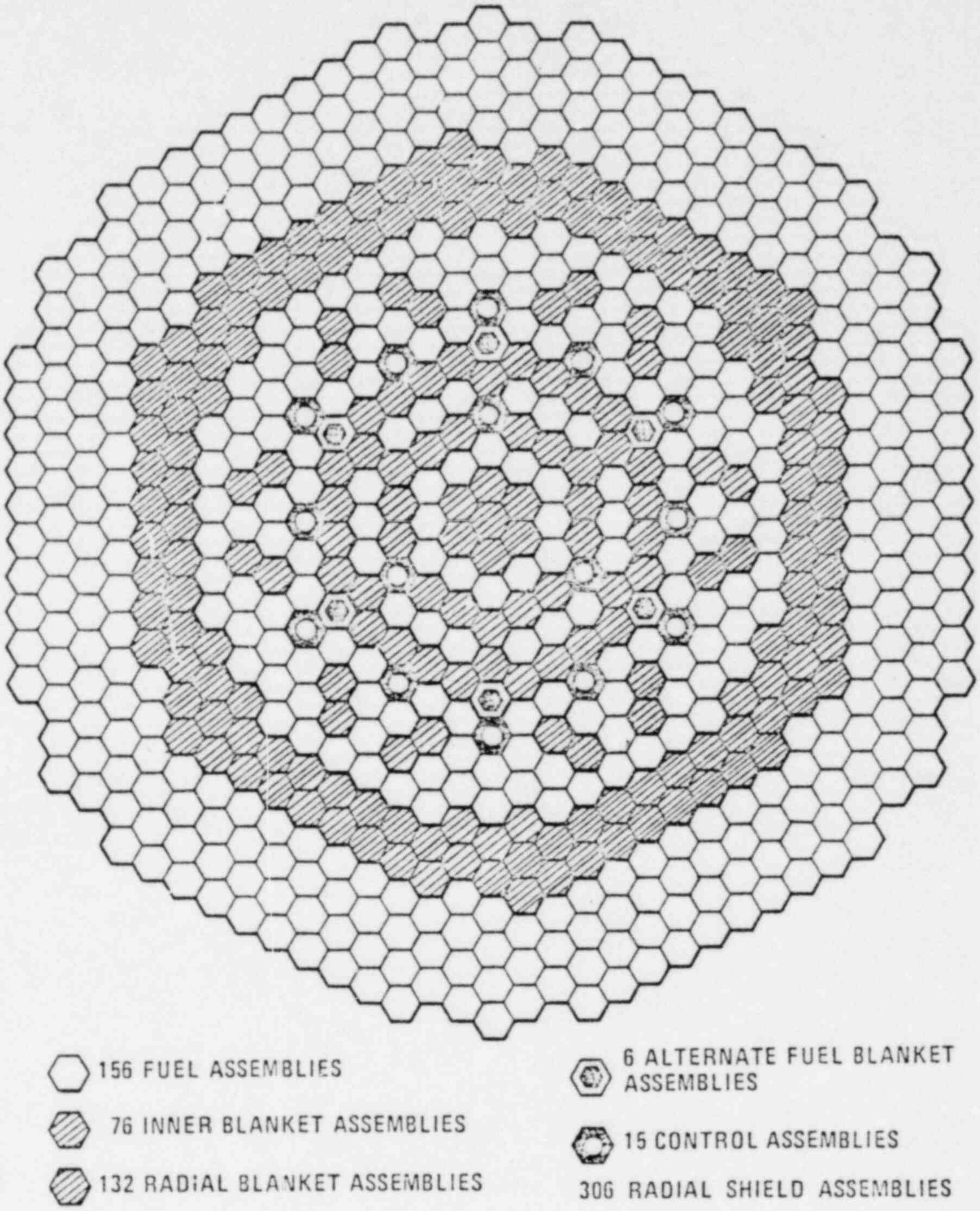


Figure 7.2 Heterogeneous core layout.

8. CDA ENERGETICS: TRANSITION PHASE LICENSING ISSUES

This section provides a discussion of issues related to the transition phase of core disruptive accidents. BNL has an NRC-sponsored experimental program which has produced key information related to phenomenon involved in the transition phase.

8.1 NRC Staff Position

The NRC staff position with respect to the transition phase was outlined in NUREG-0122 (Ref. 8.1). The core meltdown phase of the CDA was created by attempting to identify recriticality scenarios during the meltdown sequence. The objective of this approach was to determine the potential for recriticality events. It was recognized that a mechanistic treatment of this portion of the CDA was not possible due to a lack of analytical tools. On the basis of the analysis it was concluded that "... although there is potential for large reactivity insertion during meltdown, there does not appear to be sufficient driving forces to cause a sustained prompt-critical excursion which is not enveloped by the present staff position." (The staff position specified that CRBR should accomodate a "... work-energy release of 1200 MW-sec based on fuel vapor as the working fluid and expansion to one atmosphere.")

8.2 CRBR Project Position

The current position of the CRBR project (Refs. 8.2, 8.3) is that the transition phase is the most likely progression for the LOF-CDA-core meltdown. It is argued, that this leads to a non-energetic accident termination. Recriticalities are possible but incoherencies prevent them from becoming "sustained" and energetic. Large-scale molten pool development is unlikely since fuel losses away from the core region are likely to occur prior to development of such pools.

The CRBR Project position is based upon a combination of separate neutronic and thermal-hydraulic computations. The latter computations are based upon fuel relocation and volumetric boiling molten pool models. Details of the calculational methods have not yet been supplied.

8.3 Major Changes During Past Four Years

Two major factors have been introduced since 1977 which could affect the approach to transition phase issues involved in the licensing process. First, the design has gone from the homogeneous to the heterogeneous core. This, it is claimed in CRBR-3, leads to a greater likelihood that the transition phase path would be the path to CDA termination rather than early disassembly. This argues that closer scrutiny should be placed on analysis of the transition phase se-quence for the heterogeneous CRBR. The second factor is that the SIMMER (Ref. 8.4) code is now available for analysis of transient phase sequences. This will allow a more mechanistic treatment of the transition phase than was possible earlier.

8.4 Issues to be Resolved

The technical issues with respect to transition phase analysis have not changed over the past four years. Uncertainties remain is such key areas as: mechanism of fuel disruption, molten material relocation and solidification, behavior of molten boiling masses of fuel and steel, etc. These uncertainties will not be resolved on a short time scale. While SIMMER is available, it is recognized that it is largely unverified. Some out-of-pile simulant experiments have been performed to help guide application of SIMMER in transition phase calculations. However, there have been no in-pile prototypic fuel assembly meltdown experiments performed which would provide a meaningful level of verification of the code for licensing application.

The technological uncertainties and shortcomings described above suggest that the question of accident energetics will not be resolved on the short term. Analysis of the transition phase progression for the heterogeneous CRBR is, however, required to evaluate the project position and to provide support for development of a NRC licensing position. The approach should be two-fold:

(i) A phenomenological analysis of the transition phase for the heterogeneous CRBR should be performed, which parallels the effort in the BNL transition phase assessment report (done for the homogeneous core) (Ref. 8.5). This approach applies phenomenological arguments to trace possible accident progression paths. Time scales

for important events are computed and potential paths to recriticality are identified.

(ii) Accident sequence calculations should be performed using the SIMMER code. These calculations should be coupled to initiation phase calculations, and should be guided by available experimental data relevant to transition phase phenomenon.

REFERENCES

8.1. J. F. Meyer et al., "An Analysis and Evaluation of the Clinch River Breeder Reactor Core Disruptive Accident Energetics," NUREG-0122, March 1977.

8.2. R. M. Stark, "Summary of Hypothetical Core Disruptive Accident Considerations," Presentation to NRC, Bethesda, Md., October 1981.

8.3. CRBRP, "Hypothetical Core Disruptive Accident Considerations in CRBRP," Volume 1: Energetics and Structural Margin Beyond the Design Base," CRBRP-3, January 1979.

8.4. C. Bell et al., "SIMMER-I: An S_n Implicit, Multifield, Multi-Component, Eulerian, Recriticality Code for LMFBR Disrupted Core Analysis," LA-NUREG-6467-MS, January 1977.

8.5. G. A. Greene et al., "Assessment of the Thermal-Hydraulic Technology of the Transition Phase of a Core-Disruptive Accident in a LMFBR," BNL-NUREG-27366, February 1980.

9. STRUCTURAL ANALYSIS

The primary objectives of this section relate to assessing the structural adequacy of the Primary Heat Transport System (PHTS), the reactor vessel, the closure head as well as the Direct Heat Removal Service (DHRS). Such an assessment should be made for all accident conditions including seismic loadings (the safe shutdown earthquake), and core disruptive accident loads. These loads should be combined with the loads from the duty cycle. An adequate assessment is particularly needed for the plant towards the end of its useful cycle.

The issues that need resolution are:

(a) Integrity of the PHTS piping network for a number of loadings including thermal/pressure (static) and seismic disturbances (dynamic),

(b) Integrity of the DHRS piping network with its convection to the Reactor Vessel for loadings noted above,

(c) DHRS system does <u>not</u> use the guard vessel or sleeves around the primary coolant (radioactive). Adequacy of this concept needs clear assessment.

Table 9.1 Issues in the Structural Analysis Area

Item	Issues and Comments
Primary Piping System	Piping network analysis for a combination of loads, particularly towards the end of its useful life (designed plant life-30-years)
DHRS	DHRS system does not have guard vessel or sleeves around pipings that carry radioactive sodium. Substantial quantity of radioactive sodium can leak in the containment.
	Recent CRBRP analysis for thermal margin beyond design basis (TMBDB) accident did not make any assessment of large leaks through DHRS early in the core disruptive accident scenario.
	Need to assess integrity of the DHRS inlet and outlet nozzles for CDA loadings.
Containment Performance	Need to assess performance of the reactor containment building subsequence to sodium fire and also subsequent to core-concrete-sodium interactions.

10. SUMMARY

This report has identified several technical issues that will require resolution during the course of the forthcoming licensing review of the CRBR. It is clear that the new amended PSAR will need an evaluation.

For operational transients, a careful review will be needed of the selection of initiating events, of the criteria put forth for acceptance of the consequences of events, and of the methods and tools used in the evaluation of the consequences of postulated events. Particular attention should be given to the impact of the new core design on these technical issues related to operational transients.

An evaluation of the shutdown heat removal system capability is needed. Particular attention should be given to redundancy and diversity of heat removal paths and to the capability of the systems to remove shutdown heat under a spectrum of transient conditions. The potential capability of the heat removal system to remove shutdown heat under natural circulation conditions should be reviewed. The available results from the natural circulation test program for the Fast Flux Test Facility should be evaluated and assessed for their implications for the CRBR configuration.

In the area of containment heat removal, a review must be performed of the CRBR Project's approach to design basis sodium spills. Additional analyses, if needed, should be identified as a result of this review process. The analysis and philosophy associated with Thermal Margins Beyond Design Basis (TMBDB) must be evaluated. The heat removal capability of the containment to a spectrum of severe core damage conditions should be evaluated. An assessment should be performed of the interactions of sodium with concrete to establish reaction and penetration rates.

The risk assessment report that was prepared by the CRBR Project in 1977 has not yet been evaluated. In addition, the reliability of the shutdown system and of the shutdown heat removal system have not been evaluated for the new design changes. Review activities on these topics should be commensurate with the role that reliability and risk assessment will play in NRC's licensing review of the CRBR.

The heterogenous core design will need review and evaluation for reactor physics concerns as well as for more general core disruptive accident issues.

As the licensing review of the CRBR goes forward, it is expected that some resolution will be achieved on several of the aforementioned technical issues.

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