VERIFICATION OF

NATURAL CIRCULATION IN

CLINCH RIVER BREEDER REACTOR PLANT

- AN UPDATE -

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EXECUTIVE SUMMARY

The natural circulation verification program was initiated in April 1976 with the issuance of "verification of Natural Circulation in Clinch River Breeder Reactor - A Plan". This plan was implemented to verify the findings of "A Preliminary Evaluation of the CRBRP Natural Circulation Decay Heat Removal Capability" (issued March 1976) that adequate decay heat removal capability by natural circulation existed in CRBRP and to verify the computer codes required to model the response to a natural circulation transient. The purpose of this document is to present a status report on the verification work completed to date and an update of the verification plan for the work required to complete this verification.

The emphasis for this verification effort is on testing of individual components and of their interaction in the system during the transition to and operation in the natural circulation operating regime. Plant wide natural circulation testing at FFTF and EBR-II has not only confirmed component modelling acceptability but also has demonstrated that all pertinent system/component parameters have been included in the models to adequately define the plant response to a natural circulation event. Figure 1 highlights the tests supporting the verification of the DEMO, COBRA-WC, and FORE-2M codes which are used to characterize the response of CRBRP to a natural circulation transient.

The conclusion drawn is that adequate testing is in place to characterize the pressure drop and heat transfer of components and the dynamic response of the thermal heads during a natural circulation event, and, therefore, verify the codes used to predict the CRBRP response to the event.

CRBRP NATURAL CIRCULATION VERIFICATION PROGRAM

OVERALL SYSTEM VERIFICATION

- EBR-II Tests
- FFTF N.C. Acceptance Tests (Couples DEMO/COURA-WC and FORE-2M in Pretest Predictions using same methodology to be used for CRBRP)

OVERALL SYSTEM MODELS

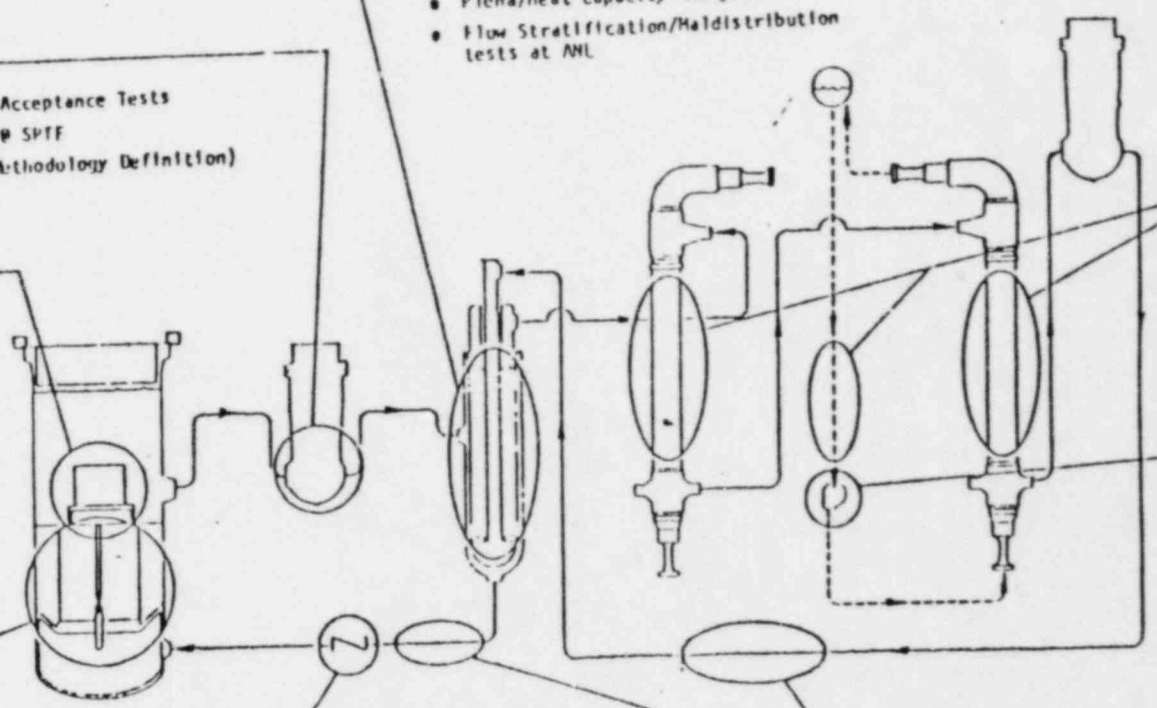
- DEMO: Plant System
- COURA-WC: Core System
- FORE-2M: Hot Rod

IHX

- Vendor Acceptance Tests
- EBR-II Tests
- FFTF Tests
- Plena/Heat Capacity Analysis
- Flow Stratification/Maldistribution tests at ANL

PUMP

- Vendor Water Acceptance Tests
- Sodium Tests @ SPTF
- FFTF Tests (Methodology Definition)



STEAM GENERATOR

- NC Tests at SCTI

OUTLET PLENUM

- ANL Scale Model Sodium/Water Mixing/Stratification Tests
- BCL Scale Model Water Mixing/Stratification Tests
- IRFM UIS ΔP Tests
- Heat Capacity Study
- Numerical Hydro Code Analyses

RECIRCULATION PUMP

- Vendor Analysis

CORE

- Decay Heat Tests (ZPPR-4, ZPPR-11, ENDF/B-IV, FFTF)
- LIM/CORE Assembly/Module ΔP Tests
- EBR-II Tests (Flow Redistribution)
- FFTF - Natural Circulation Tests
- Blanket/Fuel Bundle Heat Transfer/Mixing Tests

CHECK VALVE

- 16" (FFTF) Flow Test
- 6" (FFTF) Flow Test
- Detailed Analysis (Based on Testing)

PIPING

- ANL Stratification - ΔP Tests
- EBR-II Secondary Loop Tests
- FFTF Testing
- Heat Capacity Analysis

TABLE OF CONTENTS

1.0 INTRODUCTION

The CRBRP has been designed to remove decay heat from the reactor to the steam generator system by natural circulation in the primary and intermediate sodium loops as well as in the steam drum to evaporator recirculation loops. This has been achieved through the arrangement of components with respect to each other and through the specification of component characteristics important to natural circulation. A preliminary evaluation of the CRBRP natural circulation decay heat removal capability has been performed, which concludes that adequate circulation will be provided by the plant systems as presently specified. An overall plan to verify this natural circulation capability for the CRBRP was initially presented in 1976 (Reference 1). This update of the 1976 plan presents the results of verification efforts that have been completed to date and an updated plan to complete the verification of the conclusions drawn in the preliminary evaluation that adequate natural circulation capability exists for the CRBRP.

Three underlying assumptions to this verification are: 1) that prior to plant startup testing conclusions drawn with respect to plant wide natural circulation performance of the CRBRP will be based on analyses; 2) that a sound basis for this analysis is the understanding and verification of models for system and component behavior under natural circulation conditions; and 3) that the evaluation of data from FFTF and EBR II natural circulation testing using CRBRP methodology will contribute significantly to the overall verification of the analytical techniques used in natural circulation analyses. On this basis, the verification plan has been developed to provide the technical support required for reasonable confidence in the systems analysis tools. The program has four parts:

A. A review and selection of acceptable analytical methods (codes) used to calculate flows and temperatures to identify and provide visibility of assumptions, capabilities and limitations which may be important to natural circulation analyses.

B. A series of sensitivity studies to provide a quantitative understanding of the influence of key parameters on flows and temperatures during natural circulation transients.

C. Component testing to determine the thermal/hydraulic characteristics which are empirical in nature and to supplement this testing with detailed component analysis where testing is not programmatically feasible. Data from these tests provide the technical basis for verifying the component modelling conservatism in the system analysis code(s).

D. Comparison of predictions (both pre-test and post-test) with data from FFTF and EBR-II natural circulation tests

with the predictions being made with the same methodology and modelling techniques employed on CRBRP in order to demonstrate the systems modelling conservatism in the system analysis code(s).

2.0 REVIEW OF ANALYSIS METHODS

The purpose of a review of analysis methods is two-fold. First, it is to focus the verification efforts on the codes that, used together, can completely define the response of the CRBRP to a natural circulation transient. Secondly, it is to evaluate the capabilities of various test facilities to provide the experimental data required by the Natural Circulation Verification Program. (NCVP).

The codes to be verified by the NCVP will be DEMO (Reference 3), FORE-2M (Reference 4), and COBRA-WC (Reference 5). Three codes were selected, because it was concluded that one code alone cannot meet, nor should a singular code be developed to meet, the program objectives. Such a code would be too large a code to be of practical use in the LMFBR industry. DEMO is a system wide code used to predict the coupled performance of the reactor, heat transport systems, and the steam generator system. COBRA-WC is a whole core code that predicts how the flow and heat is distributed between the numerous parallel flow paths in the core. FORE-2M is a detailed core code used to predict "hot channel" parameters in the core. Both COBRA-WC and FORE-2M use input from the DEMO overall plant analysis and FORE-2M uses further input from the results of the COBRA-WC analysis. Therefore, as a unit these three codes provide the detailed model required to predict natural circulation transients in the CRBRP. A brief summary of each of these codes follows:

A. DEMO

DEMO is a system-wide code which predicts the coupled performance of the reactor, heat transport systems and the steam generating system (including piping and plena heat capacity effects). To provide accurate results for the plant as a whole, yet maintain the code as an amenable tool with regard to computer storage requirements and running time, localized phenomena (which do not affect the system as a whole) are not given detailed modelling. An example of this is an item such as local flow/heat redistribution between and within core assemblies at low flow. To consider localized phenomena in the core, with the required resolution, takes a separate computer code itself (i.e., COBRA-WC was selected for this purpose). In general, DEMO provides a prediction of the overall system state variables such as net flow through the reactor and bulk temperatures entering and exiting the core.

B. COBRA-WC

The COBRA-WC code, which accounts for core inter- and intra-assembly flow and heat redistribution, predicts the boundary conditions for a peak rod or cluster of rods in fuel and blanket assemblies given the reactor boundary

conditions such as total reactor flow, pressure drop and core inlet temperature from DEMO, individual core assembly powers, and individual core assembly thermal-hydraulic characteristics. Analyses of this type phenomena under natural convection cooling requires detailed core/reactor modelling because of the strong interaction between the fuel assemblies, blanket assemblies, control assemblies, plus other core regions and bypass flows which all act as highly coupled parallel flow paths with heat transfer between them.

C. FORE 2M

Given the localized peak channel flows and heat interchange as a function of time, FORE-2M predicts the hot channel coolant temperature for natural circulation which includes considering uncertainties in the predictions. This hot channel coolant temperature is used as a the basis for determining the acceptability of natural circulation for decay heat removal of LMFBR's. To have a verified hot rod prediction, one must properly account for such effects as: a) fuel restructuring; b) fuel-cladding gap conductance; c) stored heat in the rod; d) statistical significance of physics and engineering uncertainties in power, flow, dimensions, properties, ect.; e) localized hot spots on the rod due to the wire wrap or pellet eccentricity; f) localized decay heat variations; and g) localized rod power variations in the assembly.

Since FORE-2M is used for core hot rod analysis with input from DEMO and COBRA-WC, extreme detail in nodalization can be used to assure accurate temperature predictions. This level of detail is not practical for the COBRA and DEMO predictions since the storage requirements and computer running times would make the codes impractical. (Note: In addition, the code provides a detailed nuclear physics prediction for the prompt and delayed neutron fission power as a function of time from shutdown with the capability of considering doppler, sodium density, and radial/axial core expansion feedback plus the effects of control system worth and PPS functions. This information is used to verify the DEMO prediction of core power versus time).

During the transition to and operation in the natural convection cooling mode, the effect of an increasing power-to-flow ratio approaching or exceeding that of steady state can be experienced. Consequently, core temperatures increase and natural convection phenomena such as inter- and intra-assembly flow redistribution become significant once low flow conditions are reached. In the CRBRP or FFTF, the core thermal head becomes increasingly significant relative to the form and friction losses across the core for flows below 5% of full flow. Coupled with the flow redistribution, significant heat redistribution on an inter- and intra-assembly basis occurs throughout the core due to radial

temperature differentials and an increased flow transport time. Both of these effects (i.e., natural convection flow and heat redistribution) have been found to significantly reduce maximum core temperatures.

Figure 2-1 shows the interaction between the three codes used in the CRBRP natural circulation analysis procedure. The first code (DEMO) provides core delayed neutron powers, dynamic average region temperatures and inlet flow, and average core pressure drops. The last two codes (COBRA-WC and FORE-2M) provide the two-stage calculational approach for the core analyses under natural circulation conditions. Utilizing DEMO boundary conditions, detailed whole-core flow and heat redistribution analyses of all the parallel core assemblies and bypass regions is performed by the COBRA-WC code. These data are then used for a detailed analysis on a single, "hot rod" using the FORE-2M code. These latter analyses include effects of localized phenomena and uncertainties in nuclear/thermal- hydraulic/ mechanical data.

A linkage between the COBRA-WC and FORE-2M codes has been developed and verified to incorporate the inter- and intra-assembly phenomena into the localized hot rod natural circulation analyses. For each axial node of the hot rod modelled in FORE-2M, a heat balance is performed using the expression for the heat transferred to the coolant at that section, Qc(X,tau) as:

$$Qc\ (X,tau)=Qr(X,tau)+Qex(X,tau)$$

where:

Qr(X,tau) = heat transferred from the rod surface at axial location X and time tau

Qex(X,tau) = coolant heat input or loss due to radial conduction and mixing heat transfer and flow redistribution to adjacent coolant channels; directly input from COBRA-WC.

Coupled with this, the axial mass flow rate for each axial node is also input from COBRA-WC analyses. Boundary conditions for the COBRA-WC (e.g., plenum-to-plenum pressure drop and coolant inlet temperature) are furnished by the plant-wide code, DEMO for several cases: the "best estimate" or nominal case as well as cases with uncertainties. Likewise, a corresponding modelling of the core parallel flow network, with regard to pressure drop and decay heat uncertainties, can be used in the COBRA-WC analyses for input to the FORE-2M hot rod temperature predictions.

To evaluate the effects of core uncertainties, hot channel factors are used. Similar to those applied in steady state

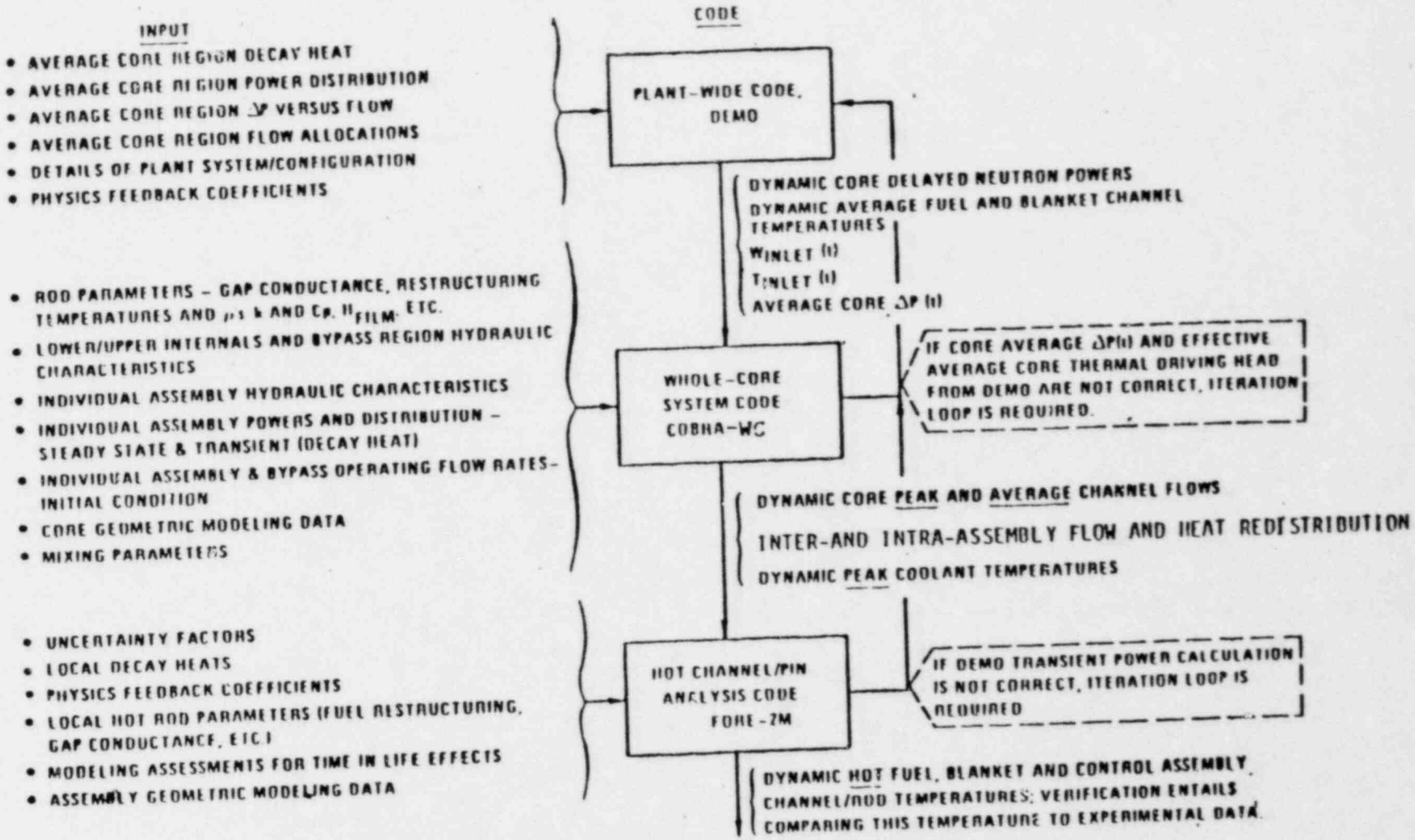


FIGURE 2-1: NCVP ANALYSIS PROCEDURE

calculations, the direct and statistical type factors are conservatively combined by the semi-statistical method.

To support this analytical method, an initial review of the testing requirements was performed. The results of this study of testing needs to support code verification are presented in the "Review of Experimental Facilities and Testing Requirements" (Reference 2). Based on this review of experimental facilities for testing and verifying the above models for natural circulation predictions, a testing program has been outlined for code verification. The specifics of this testing will be outlined in the verification discussions that follow.

3.0 SENSITIVITY ANALYSES

The objective of the sensitivity analyses is to identify those system and component parameters which have the greatest effect on the temperatures and flows during a natural circulation transient. This aids in indicating which components require the closest attention during the design process, aids in defining testing requirements, and aids in indicating where detailed modelling is required.

Sensitivity analyses in most cases are based on the outer bounds of parameter uncertainties. These analyses give early indication and visibility to the degree of conservatism in base case calculations and in nominal expected predictions of plant natural circulation capabilities. The base case for these studies is the three loop design natural circulation case from rated plant conditions presented in Reference 11. The parameters for which the temperature and flow sensitivity is determined, are based on previous CRBRP calculations, FFTF experience, and component/ system testing capabilities.

Results of initial sensitivity analyses was reported in Reference 2. This report defined the areas which required verification by testing and areas which required further study to determine the magnitude of their impact and therefore establish the need for the continuing sensitivity studies and for the testing required for component and system verification.

3.1 UPPER PLENUM MODELLING

Upper plenum characterization takes both upper internal structure modelling and outlet plenum mixing modeling into account. The base case analysis of natural circulation in Reference 11 was based on a stratified model of the upper plenum without the upper internal structure (UIS) included. Later sensitivity analysis showed the importance of including the UIS model for natural circulation analyses.

Following a reactor trip and during the transition to natural circulation, the degree of sodium mixing in the upper plenum determines the sodium density profile along the flow path in the plenum and primary hot leg piping. The integrated density around the whole primary loop flow path determines the natural circulation driving head. It follows that plenum mixing assumptions influence the plenum and primary hot leg contribution to the thermal driving head and, therefore, the natural circulation flows.

Results of an initial sensitivity study on the outlet plenum mixing (UIS excluded) impact on hot channel temperatures was reported in Reference 2. The range of impacts for full stratification to full mixing in the outlet plenum was shown to be relatively minor. Blanket assembly temperatures varied from $+12^{o}F$ to $-0^{o}F$ while fuel assembly temperatures varied from $+0^{o}F$

to -15°F.

The addition of the UIS to the model for natural circulation analyses was found to have a significant impact on the natural circulation transient. The major impact of this addition was on the transport time for the transient to reach the plenum through the chimneys. This resulted in a negative impact on thermal heads causing the primary flow to drop further and recover slower than predicted in the base case analysis. The impact on maximum sodium temperature in the core was an increase of approximately 70°F in the calculated value.

A sensitivity study has recently been performed which evaluated the impact that the upper plenum modeling assumptions have on plant transients. The basis for this comparison was the reactor outlet temperature calculated for a normal plant scram (duty cycle event U-1B). These evaluations centered on two central characteristics, the upper plenum sodium mixing efficiency and the upper internals structure (UIS) impact. The models used in this comparison were:

A. A one node fully mixed model.

B. A stratified model.

C. A detailed two region mixing model based on the ANL-PLENUM-2A code but including mass accumulation in the plenum.

D. A detailed three region mixing model which is based on the ANL-PLENUM-3 code and includes the UIS.

E. A modified version of the two region model above that includes the UIS and a more accurate geometric representation of the flows and volumes in the outlet plenum.

These models were compared against the results of a two dimensional plenum calculation with the VARR-II code (Reference 19) and the results of this comparison are shown in Figure 3-1. Based on this study, the modified two-region model has been used to predict outlet plenum performance for FFTF natural circulation test pretest predictions and is planned for use in predicting CRBRP outlet plenum performance.

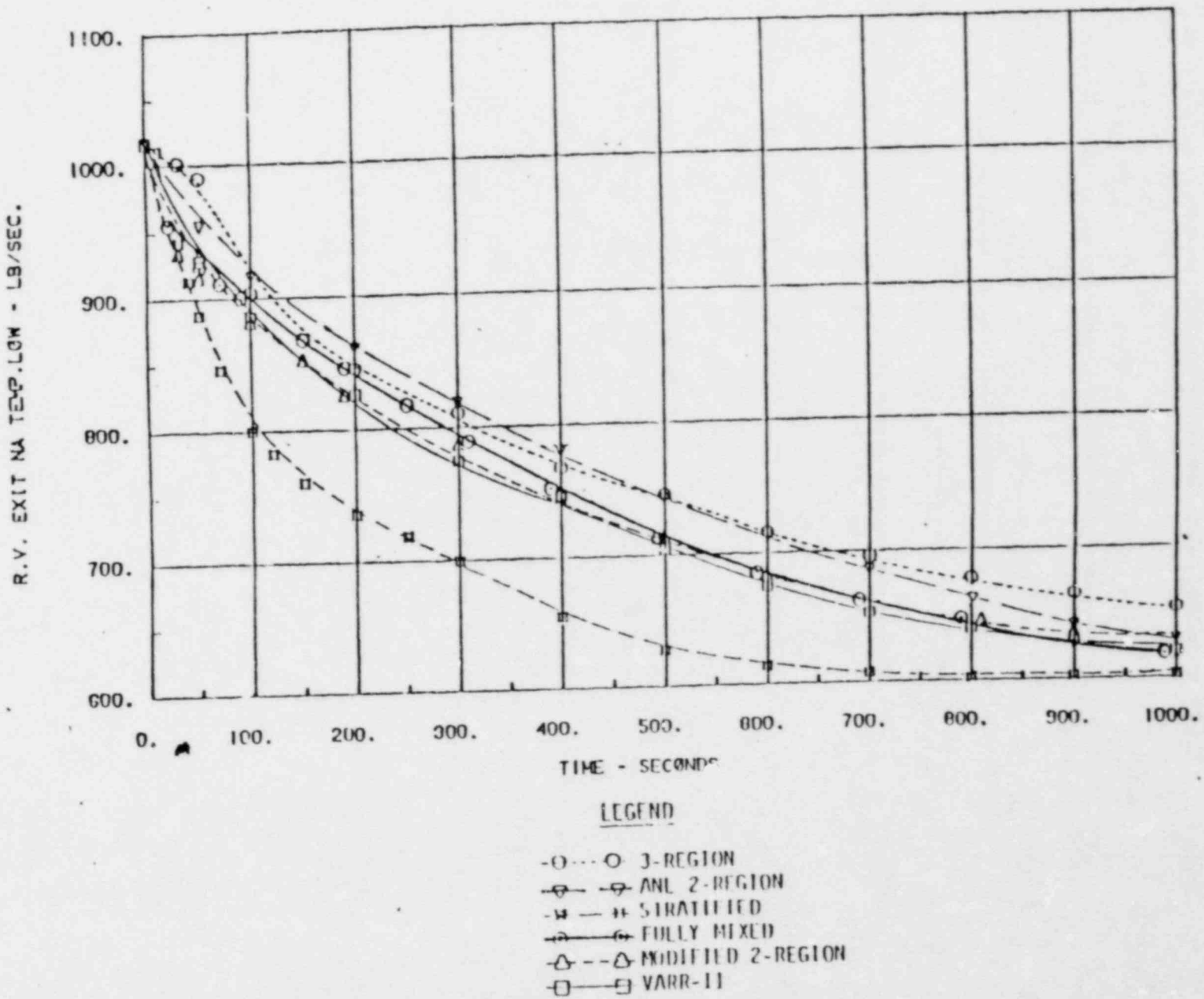
The results of these studies provide the basis for the verification testing discussed in Section 4.1.3.

3.2 DECAY POWER

The thermal load from decay power is one of the more important factors determining natural circulation performance.

As discussed in Section 4.2, values of reactor system and

FIGURE 3.1 - REACTOR OUTLET TEMPERATURE VS. TIME



individual assembly decay heat include time dependent uncertainties which are maximum at the time of scram and decrease in magnitude with increasing time after scram initiation. The decay heat uncertainties from scram to two (2) hours after scram are summarized as follows:

	Uncertainty		
	(a) Scram	(b) 2 minutes	(c) 2 hours
Reactor System	31.0%	12.3%	8.8%
Fuel Assembly	32.4%	13.7%	10.3%
Inner Blanket Assembly	28.6%	9.1%	5.6%
Radial Blanket Assembly	27.6%	10.1%	7.2%

These values are for end-of-equilibrium cycle conditions at rated power operation.

Based on an uncertainty in decay heat of +0% to -20% used in the Reference 2 sensitivity study, the effect on blanket temperatures during a natural circulation transient was shown to be from +0°F to -137°F and the effect on fuel temperatures was from +0°F to -130°F.

3.3 CORE INTER- AND INTRA-ASSEMBLY FLOW REDISTRIBUTION AND ASSOCIATED HEAT TRANSFER EFFECTS

For the core natural convection cooling mode, the effect of dynamically approaching low flow with worst case decay heat loads results in a power-to-flow ratio greater than one. Consequently, core temperatures increase and natural convection phenomena such as inter- and intra-assembly flow redistribution due to different thermal heads and hydraulic characteristics of the core assemblies become important. In general, the core thermal head becomes significant relative to the form and friction loss across the core below approximately 5% of full flow. Coupled with the flow redistribution, significant heat redistribution on an inter- and intra-assembly basis occurs throughout the core due to large temperature differentials and an increased heat transport time (low power assemblies can have a transport time of over 20 seconds). These effects (i.e., natural convection flow and heat redistribution) are found to significantly reduce maximum core temperatures as demonstrated in the EBR-II natural circulation experiments (Ref. 14).

Independent studies outside the CRBRP Project have been published which also show that reactor flow redistribution causes a significant decrease in predicted maximum core temperatures during natural circulation conditions. For example, Brookhaven National Laboratory (Agrawal, et al., in Ref. 15), using the

SSC-L code, predicted localized flow increases as large as 20% in the hot fuel assembly and 40% in the hot blanket assembly for the CRBRP during natural convection. Temperature reduction on the order of 16% and 22% and 210°F) were shown for the hot fuel and blanket assemblies, respectively, relative to the maximum temperatures predicted without flow redistribution. Similar results were found in Reference 16 using the CURL-L code. These studies do not include inter-assembly heat transfer effects, or intra-assembly flow redistribution and heat conduction effects. However, if included, these effects would further reduce the maximum predicted core temperatures.

Preliminary studies have been performed for the CRBRP to demonstrate the effect of inter-assembly flow redistribution for the heterogeneous core design. The effects of inter-assembly heat transfer and intra-assembly flow and heat redistribution were neglected in this case. Figure 3.2 shows the results of these analyses for the peak fuel, peak inner blanket and peak radial blanket assemblies. Figure 3.3 shows results for a typical orificing zone for the fuel, inner blanket and radial blanket assemblies. Consistent with other natural circulation studies, the flow increase to the hotter core regions is apparent. This effect, along with the other natural convection phenomena, will significantly decrease the maximum hot rod temperatures in the core.

As discussed in detail in Section 2.0, the effect of all natural convective cooling phenomena (i.e., inter- and intra-assembly flow redistribution and heat transfer) on the maximum transient coolant temperatures in the CRBR core will be assessed using three computer codes (DEMO, COBRA-WC, and FORE-2M).

Four cases have been analyzed to illustrate typical results predicted by this technique for a high temperature fuel rod:

- Case 1--Fuel assembly peak coolant channel transient flow and temperature calculations including both inter- and intra-assembly flow and heat redistribution, nominal conditions (no uncertainty factors applied);
- Case 2--Fuel assembly peak coolant channel transient flow and temperature calculations including both inter- and intra-assembly flow and heat redistribution; uncertainty factor conditions;
- Case 3--Fuel assembly peak coolant channel transient flow and temperature calculations only including inter-assembly flow redistribution; uncertainty factor conditions; and
- Case 4--Fuel assembly transient flow and temperature calculations without inter-assembly and intra-assembly flow and heat redistribution; uncertainty factor conditions.

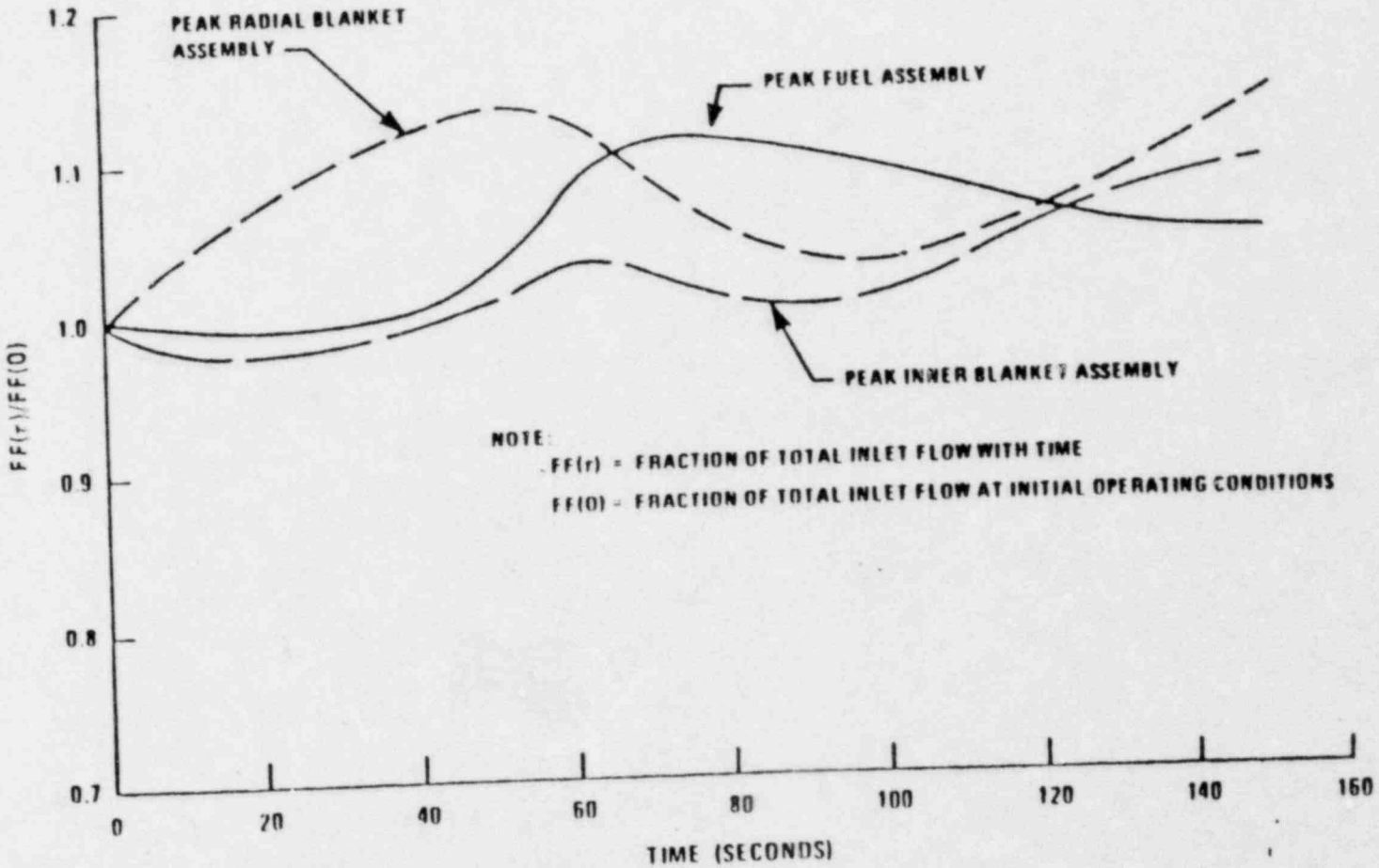


Figure 3.2 Typical Inter-Assembly Flow Redistribution for Peak Fuel, Inner Blanket and Radial Blanket Assemblies During Natural Circulation Transient

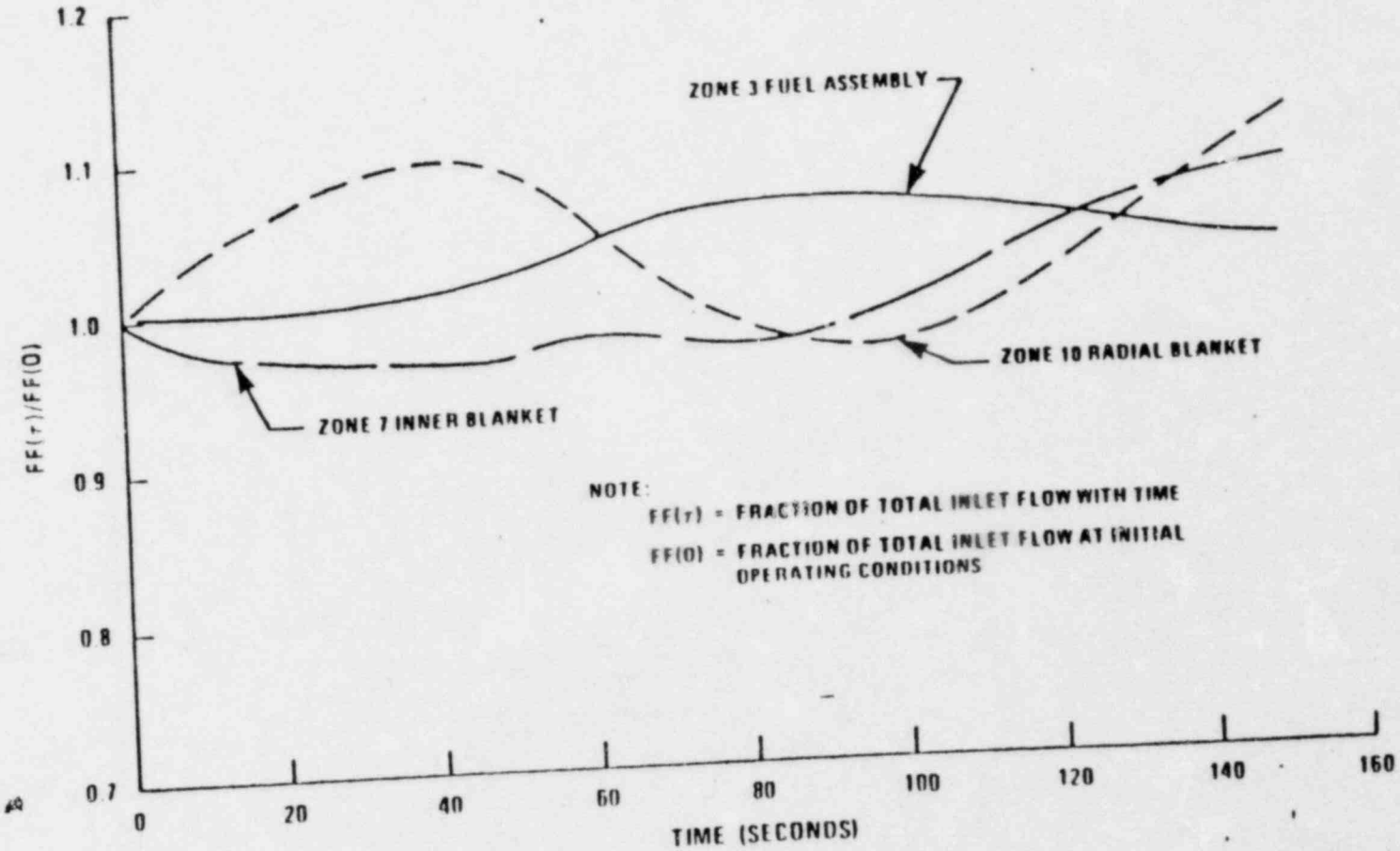


Figure 3.3 Typical Inter-Assembly Flow Redistribution for Several Core Orificing Zones During Natural Circulation Transient.

Results for these four cases are presented in Figure 3.4 as normalized temperature differences relative to the steady state temperature difference in Case 1, i.e.,

$$\emptyset = \frac{[Tc(tau)-TM(tau)]i}{[Tc(0) - Tm(0)]\ i} \qquad i=\text{case } 1,2,3 \text{ or } 4$$

where:

Tc(tau) = maximum hot channel coolant temperature;

Tin(tau)= inlet temperature.

As can be noted by comparing Cases 2, 3 and 4, accounting for inter- and intra-assembly flow and heat redistribution effects significantly decreases the predicted transient coolant temperatures in the hot channel. It can also be seen that the uncertainty factors cause a significant increase in the predicted hot channel coolant temperatures (i.e., Case 1 versus Case 2).

In summary, natural convection cooling of the core exhibits one of the few CRBRP core design transients where low power/high temperature conditions exist. Due to the long coolant transport time and low pressure drop for the core while descending into and operating in this mode, core inter-assembly and intra-assembly flow/heat redistribution; a) becomes significant with regard to accurately predicting temperatures; and b) significantly decreases the maximum hot rod temperatures in all core regions, when compared to analyses which neglect these effects.

3.4 LOOP FLOW RESISTANCES

The primary and intermediate loop flow resistances have a direct effect on the quasi-steady state flows and hot to cold leg during natural circulation.

During the transition from forced to natural circulation following a pump trip, the coastdown rate is determined by the system's initial kinetic energy and its rate of energy dissipation in the pump and system. The hydraulic dissipation rate is directly proportional to the system flow resistance. The plant operating conditions prior to the transient are also directly influenced by the primary system resistance. Resistances lower than that used for the base case analysis (Reference 11) would result in higher flows and, therefore, lower initial primary and intermediate system temperatures.

The results of the base case analysis show that the primary

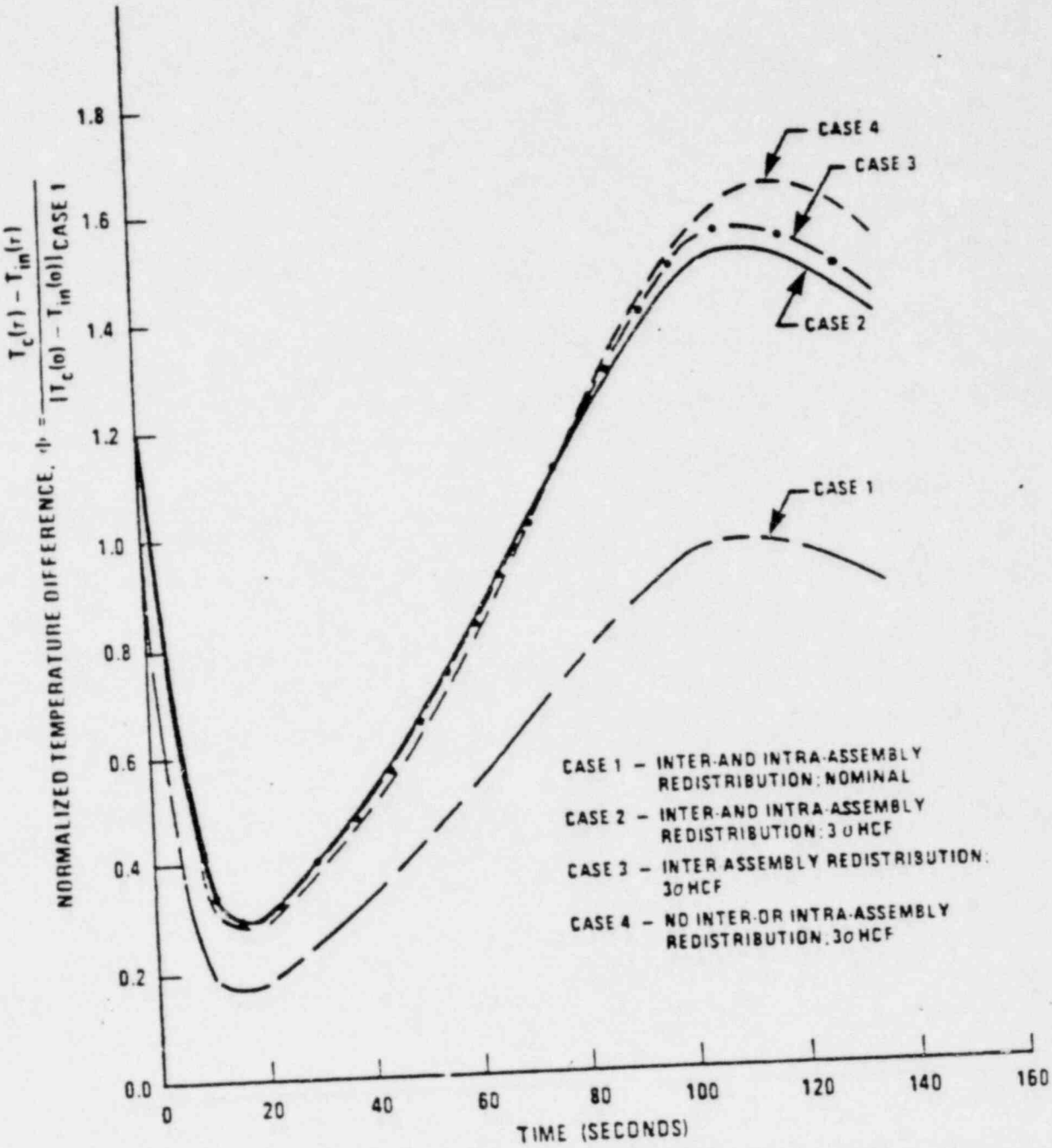


Figure 3.4 Typical Effect of Uncertainties and Inter- Intra-Assembly Flow Heat Redistribution on F A Peak Coolant Temperatures During Natural Convection Cooling

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system flow has the highest affect on both initial conditions and natural circulation conditions. As long as the IHTS resistances are such that the IHTS flow rate at natural circulation conditions is higher than the PHTS flow rate, the actual IHTS pressure drop uncertainties have less of an impact on the transient (primary flows) than do the PHTS pressure drop uncertainties. Therefore, sensitivity studies have focused on the primary system. The overall primary loop system pressure drop is composed of that in the reactor (45%), the pump with an assumed stationary impeller (25%), the IHX (5%), the check valve (20%), and the piping (5%). The percentages are based on worst case specifications at natural circulation flows for the respective components and are referenced to the base 3-loop natural circulation case. For natural circulation on less than 3 loops, the percentage drop from the loop components would increase. These pressure drop fractions, which show the relative impact of the components, plus the results of the sensitivity study, which defines the relative importance of the overall pressure drop, have been used to define component testing which will verify the pressure drop correlations and have been used as a basis for the sensitivity evaluation of the primary system resistance reported in Reference 2. The impact of a +17% to -32% primary loop flow resistance uncertainty range at 3% flow was found to be +50°F to -70°F in blanket assembly hot channel temperature and +66°F to -100°F in fuel assembly hot channel temperature. The sensitivity of temperatures and flows during a natural circulation transient to the flow coastdown rate and to the initial conditions are specifically addressed in the sensitivity studies discussed in Sections 3.5 and 3.7.

3.5 FLOW COASTDOWN

The sensitivity of the maximum cladding temperature for the fuel and blanket assembly hot rods to various pump and pump drive system stored energy values were performed during the first half of 1975. This study was used to provide a basis for the specification of the pump and its drive system. Increasing stored energy in the pump results in lower fuel assembly peak temperatures and times after scram at which these peak temperatures occur. As reported in Reference 1, the coastdown characteristics of the pump are a significant factor in determining the flow integral and, therefore, the peak fuel and blanket cladding temperatures. Within the narrow design window allocated for pump inertia, the impact of using the maximum pump inertia in natural circulation analyses, which is 11% higher than that used in the base case (minimum inertia) analysis, was reported by Reference 2 to reduce the blanket assembly hot channel temperature by 15°F and the fuel assembly hot channel temperature by 20°F. Therefore, the pump coastdown characteristic which depends on the effective inertia, pumping torques and friction torques, will be tested, as discussed in Section 4.3.1, to determine the actual values for use in natural circulation analyses.

3.6 IHX HEAT TRANSFER PERFORMANCE

The IHX is designed to promote mixing on the primary shell side. However, during a natural circulation transient with low primary flows, one may speculate that buoyancy induced flow maldistributions may exist. Such maldistribution could alter the unit's heat transfer effectiveness, change the frictional pressure drop, and also change the unit's contribution to the thermal driving head. Assuming a range in IHX performance of +33% to -73% (heat transfer performance) the Reference 2 sensitivity study predicts a blanket assembly hot channel temperature variation of +0°F to -7°F and a variation of +7°F to -13°F in fuel assembly hot channel temperature. This study was made by assuming that only a specific fraction of the tubes are active and the remainder are thermally and hydraulically isolated. To implement this analysis, the effective heat transfer area was varied in direct proportion to the assumed effective flow fraction. Thus, extremes in IHX flow maldistribution show insignificant impact on natural circulation transients.

3.7 TRANSIENT INITIAL CONDITIONS: FLOWS AND TEMPERATURES

The preliminary evaluation of the CRBRP natural circulation capability was based on a set of plant initial conditions for the primary, intermediate, and steam generator systems that are believed to result in a conservative set of temperatures and associated flows. The primary flow used corresponds to the minimum flow which would be expected at 100% pump speed since it is related to the intersection of the minimum pump head/flow characteristic with the maximum system resistance curve. The intermediate loop flows and temperatures were based on the thermal hydraulic design flow with an arbitrary upward adjustment of 20°F on the hot and cold leg temperatures.

The actual plant conditions, however, can vary significantly from the conditions discussed above. The primary flow depends on the "as built" loop impedance characteristics and pump head/flow characteristics. The IHTS conditions are controlled by the heat transfer conditions between it and the SGS. It is important to note that conditions of fouling and plugging for the steam generator modules will vary with plant lifetime. From a performance prediction standpoint, uncertainties in heat transfer coefficients and the transition from nucleate to film boiling in the evaporator can require the IHTS conditions to vary significantly. At rated power, for example, the required intermediate loop flow may be at any value between 11.5 and 13.5 million lb/hr. The primary loop flows can vary between 13.8 and 15.9 million lb/hr. The corresponding variation in hot and cold leg temperatures along with the possible variations in system resistances have an affect on the flow decay curves in the PHTS and IHTS and, therefore, the behavior of the IHX and ultimately have a contribution to variations in thermal heads.

Sets of temperature and flow conditions will be selected to bracket the possible variations in these conditions and the sensitivity of the natural circulation controlling parameters will be determined for each case. This sensitivity study of plant initial conditions will be performed using the DEMO computer code and the results will be presented as a part of the natural circulation assessment supporting the final system designs.

4.0 VERIFICATION OF SUBSYSTEM CHARACTERISTICS

4.1 REACTOR THERMAL HYDRAULIC ASPECTS

4.1.1 REACTOR STRUCTURES AND ASSEMBLIES OVERALL REACTOR DELTA P

The reactor vessel dynamic pressure drop correlations used in DEMO are as outlined below:

$$\text{Delta P Dynamic}_{\text{Demo}} = \text{Delta P Total}_{\text{Cobra}} = \text{Delta P Static}_{\text{Demo}} \quad (4\text{-}1)$$

where:

Delta P Total = Cobra	total reactor pressure drop calculated by COBRA for given reactor flow;
Delta P Static = Demo	static reactor pressure drop as calculated by DEMO for the same flow conditions; and
Delta P Dynamic = Demo	dynamic pressure drop to be used for determining the pressure drop correlations.

Using experimental data on pressure drop under low flow conditions through each of the reactor components, the total pressure drop for a series of steady state calculations at various flows with the power-to-flow ratio near one are made. The dynamic pressure drop is then calculated from Equation 4-1.

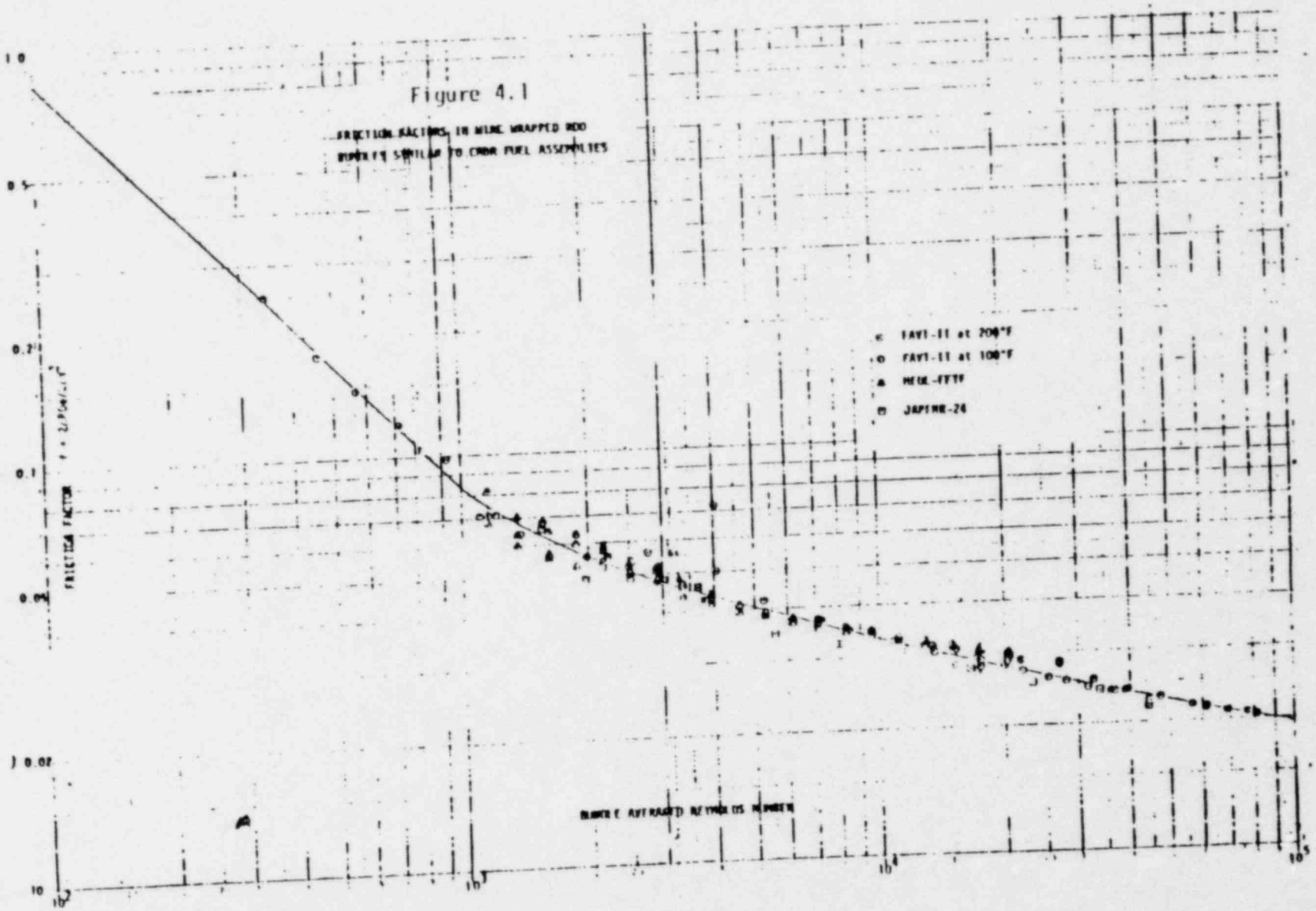
Test programs are either underway or completed to experimentally determine the hydraulic characteristics of all reactor components over the range of expected flow rates. Table 4.1 lists the major reactor pressure drop components and the applicable test programs. These test programs will confirm that the flow rate in each CRBRP assembly will be achieved with an error no greater than the value assumed for design analyses[18]. This test data will determine the reactor pressure drop characteristics down to approximately 2% of rated flow to cover natural circulation operation. Figures 4.1 and 4.2 show typical data for fuel and blanket rod bundles, respectively. As can be noted, a smooth transition between turbulent and laminar flow occurs.

4.1.2 CORE FLOW REDISTRIBUTION AND HEAT TRANSFER

Flow and heat redistribution verification data are required at high and low steady state operating conditions and during transients. The tests listed in Table 4.1 will provide the necessary data to verify the reactor assembly hydraulic characteristics. Verification of temperature distributions at

Figure 4.1

FRICTION FACTORS IN WIRE WRAPPED ROD BUNDLES SIMILAR TO CRBR FUEL ASSEMBLIES



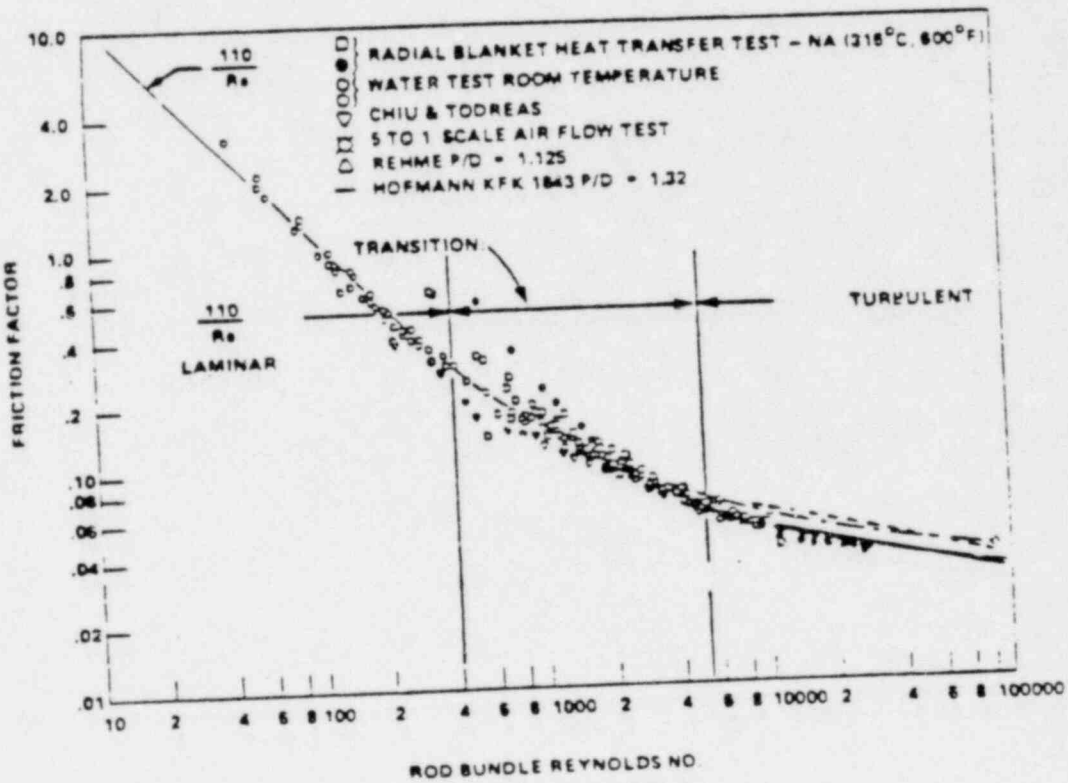


Figure 4.2 Average Friction Factor Data for a Wire-Wrapped Rod Bundle: P/D = 1.08, 4" Wire-Wrap Pitch Similar to CRBR Blanket Assemblies

0369-1

TABLE 4.1
REACTOR COMPONENT PRESSURE DROP VERIFICATION TESTS

Pressure Drop Component	Test	Status
Inlet Plenum & Modules	Inlet Plenum Feature Model	Hydraulic testing completed 1976 (1/4scale)
[illegible] Plenum and Upper [illegible]als	Integral Reactor Flow Model (IRFM)	Delta P data completed 1977 (1/4 scale)
	IRFM Core and Permanent Reactor Structure Thermal Striping Test	Complete testing in 1982. Report to be issued 1983.
Radial Blanket Orificing	Radial Blanket Flow Orificing	Orifice Calibration completed 1979, complete testing LIM in 1982. Report to be issued 1982.
Fuel Assemblies:		
Inlet Nozzle, Shield and Orifice	F/A Inlet Nozzle Flow	Completed full scale water test 1979
	Cavitation and Orifice Calibration Test	Testing completed in 1982. Report to be issued in 1982.
Rod Bundle Inlet and Bundle Outlet Nozzle	F/A Flow and Vibration Test	Full scale water test completed 1980
Outlet Nozzle	F/A Outlet Nozzle Flow Test	Full scale water test test completed 1979
Blanket Assemblies:		
Rod Bundle, Rod Bundle Inlet and Assembly Outlet	Radial Blanket Assembly Flow and Vibration Test	Full scale water test completed in 1982. Report to be issued in 1982.
Control Assemblies:		
	Control Assembly Hydraulic Tests	Full scale water test completed 1982. Report to be issued in 1982.
Removable Radial Shield Assemblies:		
RRSA Orifice Pressure Drop Test	RRSA Orifice Pressure Drop Tests	Full scale water test completed 1977

high reactor flow rates in fuel assemblies will draw upon the large body of sub-channel mixing and heat transfer test data collected in support of both FFTF and CRBRP. Calibration and verification of temperature distributions in blanket assemblies will be provided by the full scale blanket heat transfer test program with sodium coolant. This blanket testing by W-RM provides detailed data on intra- and inter-assembly heat transfer for a wide range of representative blanket power distributions at low, intermediate and high flow rates. Transient forced and natural circulation data in the natural circulation flow range are also available. Fuel assembly low flow intra-assembly flow and heat redistribution data are available from the Fuel Assembly Natural Circulation Sodium Heat Transfer Test at HEDL. Detailed steady state and limited transient heat transfer data in sodium on fuel rod bundles of 61 pins are also available from the heat transfer test program conducted at ORNL for a wide range of flow and heat input conditions. An extensive wire wrap bundle mixing test data base is available which has been utilized to improve the understanding of the thermal/hydraulic phenomena within wire wrapped rod bundles for calculation of subchannel analysis computer codes. References 12 and 13 describe many of these tests.

Aside from the single assembly data described above, valuable core systems data for confirming the verification of the core inter- and intra-assembly flow and heat redistribution analyses is available from the EBR-II and FFTF natural circulation tests. In EBR-II detailed flows and temperatures have been measured in the instrumented XX07 and XX08 assemblies which were prototype EBR-II fuel assemblies located in the core during the testing. Similar FFTF data is available from Open Test Assemblies (OTA's). A high flow/high power zone and a low flow/low power zone prototypic fuel assemblies were located in two OTA's during the FFTF testing. Both the EBR-II XX07 and XX08 assemblies and the FFTF OTA's had flow meters and numerous thermocouples at various axial and radial locations throughout the rod bundles. These data sources are extremely valuable for validating the local, as well as the core wide, temperature calculations made by the COBRA-WC and FORE-2M computer codes.

Many irradiation experiments have been performed in EBR-II to verify the fuel restructuring characteristics under prototypic fast reactor conditions. This includes fuel and blanket rod type irradiation data. Similar data will be available from FFTF to support the CRBRP FSAR. This information is being used to verify the steady state fuel behavior code LIFE (Reference 17) which in-turn is used to validate the restructuring model in the transient core hot rod analysis code FORE-2M. The restructuring validation is necessary to assure that phenomena which offset the rod stored heat release/inertia are satisfactorily modelled.

Table 4.2 summarizes the verification approach for the COBRA-WC and FORE-2M analyses described above.

TABLE 4.2

COBRA-WC/FORE-2M CODE VERIFICATION APPROACH

- o Apply steady state single assembly data taken at conditions indicative of natural circulation (High Power-to Flow Ratios, Low Flow and High Power Skews)
  - -- HEDL 217-Rod F/A low flow data in sodium (includes data with one side of bundle heated and other side unheated)
  - -- ARD 61-Rod B/A data in sodium (Steady State and Transient Data includes bypass channels to typify inter-assembly heat transfer)
  - -- ORNL 61-Rod F/A data in sodium (Steady State and Transient Data)
  - -- Fuel, Blanket and Control Assembly hydraulic characterization from 100% to low flows
- o FFTF and EBR-II Quasi-Steady State and Dynamic Data for Coupled Core System
  - -- Low power and flow quasi-steady state tests
  - -- Dynamic total loss of power tests from initial high power/flow operation
    - a) Single XX07 and XX08 instrumented assembly for accurate flow and rod bundle temperature data in EBR-II tests
    - b) Multiple Open Test Assembly (OTA) data for accurate flow and rod bundle temperature data in FFTF tests
- o Irradiation data from EBR-II and FFTF to confirm fuel restructuring, fuel/cladding gap conductance, stored heat in rod, etc., for fuel and blanket type hot rods (by direct comparisons and by LIFE computer code calibration)

4.1.3 REACTOR UPPER PLENUM MODEL

Sensitivity studies have shown the necessity to verify the outlet plenum thermal transient behavior during a natural circulation event. This temperature response at the outlet nozzles can directly influence the overall reactor plant performance during the event.

Although sensitivity studies showed that mixing/stratification in the outlet plenum had a minor impact on overall natural circulation in CRBRP, a significant number of tests to assess the outlet plenum characteristics have been performed. These tests include Argonne National Laboratory (ANL) 1/10-scale model water tests (Reference 21), ANL 1/15-scale model water and sodium tests (Reference 22), and Battelle-Columbus Laboratory (BCL) 0.55 scale model water tests (Reference 23). It should be noted that water testing in small scale models is adequate to simulate full scale CRBRP sodium conditions. The ANL experiments indicate that no significant difference exists between water and sodium testing (1/15 scale models). Additionally, comparison of the BCL 0.55 scale, ANL 1/10 scale, and ANL 1/15 scale model test results indicate little difference in transient behavior.

The data obtained in the water natural circulation experiments were compared to predictions obtained from VARR-II[19], a numerical hydro code. This code has been verified and used to predict outlet plenum transient responses at the outlet nozzle for all natural circulation events that will be considered for evaluation in CRBRP. The outlet nozzle responses predicted by the numerical hydro code will be used in conjunction with the overall plant performance code such that accurate feedback, due to plenum transient behavior, will be included in the system model.

The tests discussed above also showed that the UIS chimneys mitigate flow stratification and improves fluid mixing. However, as discussed in Section 3.1, the major impact that the UIS has on the natural circulation event is due to its impact on natural heads/loop pressure drop and therefore flows. The thermal head's retarding of flows in the UIS makes the value of the hydraulic resistance across the gap between the bottom of the UIS skirt and the removable radial shield (RRS) nozzles important. Therefore, 1/4 scale model IRFM water tests have recently been conducted to determine the pressure drop characteristic of this gap. The results of these tests will be used to set the range of gap resistances in system analyses.

Confirmation testing for CRBRP models is not possible from FFTF natural circulation tests. This lack of verification is due to the differences in the upper internal structures. The FFTF instrument tree and CRBRP UIS chimneys have significantly different impacts on both the outlet plenum mixing and the UIS impedence/thermal head affects, however, sufficient tests are in place without FFTF test input to demonstrate that the outlet

plenum models that are important to natural circulation analyses in CRBRP.

4.2 DECAY POWER

The post-shutdown reactor power generation is comprised of the fission power and decay power. The fission power is calculated in DEMO and FORE-2M with a point kinetics model. COBRA-WC uses the DEMO calculated fission power. The decay power, on the other hand, uses data from separate decay power calculations and is input in tabular form as a function of time, with separate tables for the fuel and non-fuel assemblies. Thus, the codes (analysis methodology) and input data used to generate decay powers for DEMO, COBRA-WC and FORE-2M analyses are included in the verification of these system codes.

Decay power predictions of the reactor core include contributions from fission product, transuranic, actinide, and neutron activation isotope decay energy release. Isotopic fission and capture rates in the CRBRP fuel, inner blanket, and radial blanket assemblies are used in the prediction of the reactor system decay power. Verification of the data for fission product isotope decay energy release due to fast neutron fission in Pu^{239}, U^{235}, and U^{238} has been provided by detailed comparison to experimental data. The transuranium isotope decay energy release is provided by decay data from Nuclear Data Sheets or Table of Isotopes and exists as reference data. Verification of isotopic neutron fission rates and neutron capture rates is provided through an extensive critical experiment program.

Time dependent uncertainties associated with fuel and blanket assembly decay power result in approximately a 28-33% increase in predicted decay heat at shutdown and approximately a 6-10% increase at 2 hours after scram. Actual uncertainties are dependent on the type of assembly, i.e., fuel, inner blanket, or radial blanket. The fission rate data uncertainty is included directly in hot channel factors used for the analysis of the hot rod in an assembly. U^{238} capture rate uncertainties are verified values and uncertainties are included in transuranium isotope decay energy release data.

ENDF/B-IV nuclear data (fission product isotope decay constants, energy yields, and branching ratios) for 824 fission product isotopes and fast fission yields for 6 fissionable isotopes for application to CRBRP were provided upon completion of a contributing base development program. This data bank was used to provide decay energy release data for each fissionable isotope for use in a design methodology that considers the detailed spatial and time variation of isotopic fission rates for full power, partial power, or load follow conditions. Verification of the ENDF/B-IV data base was achieved by prediction of decay heat experiments where individual fissionable isotopes were irradiated and decay energy release was measured by spectrometric or calorimetric methods. Use of the design methodology method (S-4M

or RIBD-II computer program) to predict experiments as a function of time of irradiation and decay time was performed. Uncertainty assessments for U^{235} and Pu^{239} have been completed and documented in References 24-35. This information, as well as analysis of current experiments, provides verification of decay energy release data.

Verification of the codes and input data used to generate decay heat values for DEMO, COBRA-WC, and FORE-2M analyses will be complete by July, 1983.

4.3 PLANT COMPONENT THERMAL HYDRAULIC CHARACTERISTICS

4.3.1 PRIMARY AND INTERMEDIATE PUMPS

The pump characteristics important to the analysis of natural circulation events are those which influence the flow coastdown and the resistance to flow after the pump has stopped turning.

The pump characteristics are second only to the reactor decay power in their effects on peak temperatures. The pump coastdown affects the peak core temperatures in three ways. First, the integral of the flow coastdown determines the amount of sensible heat that has been taken out of the reactor. Secondly, the time at which the pump stops determines the reactor decay power level at which transition to natural circulation occurs. Thirdly, the stopped pump impedance represents approximately 25% of the loop pressure drop at natural circulation flows.

The initial primary flow coastdown (down through 10%) is largely determined by the pump and pump drive system stored energy and this follows the system pressure drop curve down to low flow. This coastdown (initial flow coastdown and subsequent pump tailoff) affects the power to flow history and hence the temperature transients in the core. The preliminary natural circulation calculations have been based on minimum specified flow coastdowns with conservative assumptions on the pump tailoff (high values for the loss torques at low speed).

The pump pressure drop, when the shaft stops rotating, has been calculated based on an equation which treats the pressure drop as fully turbulent (flow exponent of 2.0) and the maximum specified coefficient on the normalized flow has been used. The relationship implies a pressure drop of 54.6% of design head, should design flow be forced through the stopped pump in the normal flow direction. Based on natural circulation experiments at EBR-II, analysts have revised estimated pump pressure drops to significantly higher values. A flow exponent of 1.8 was recommended together with transition to laminar at high Reynold's number approximately (50,000) which resulted in a turbulent to laminar transition at about 6% flow, which is well above the 3% flow range expected for CRBRP primary flow in the natural circulation mode. Experiments to measure the locked rotor pressure drop of the FFTF pump at LMEC indicate, however, that

for the FFTF pump at least, no such transition to laminar flow was evident for flows down to approximately 2%.

Preliminary performance data has been developed, based on vendor water test data. This data has been used to make a determination of the coastdown characteristics and stopped rotor impedance for use as interim data until sodium test data is available.

During prototype pump testing at the Sodium Pump Test Facility (SPTF), a series of coastdown tests will be conducted to verify that the equations modelling the decay in flow are, in fact, conservative. By running a series of tests, data will be generated to determine a mean and variance for the pump coastdown and locked rotor resistance. These statistical values can then be used to verify the conservatism used in the system simulation. Prototype pump tests at SPTF began in 1982. Therefore, both the calculated values and preliminary water test data used in the model will be verified during sodium testing at SPTF. Water tests of production pumps will be conducted at the vendor's facility. These tests are expected to confirm the repeatability of the prototype pump characteristics particurlarly with respect to the pump "tailoff" and the locked rotor resistance. The method used to develop the pump characteristics from test data has been verified (Reference 9). Excellent comparison of DEMO calculations and measured speed and flow coastdowns at FFTF were obtained.

4.3.2 IHX

Verification of the IHX model in the DEMO code is important for two reasons. The thermal performance of the IHX is important to assure that the overall plant response is correct and secondly, to assure that the structural design transient applied to the IHX in the equipment specification is adequate for structural analyses.

Both of these considerations are dependent on the assumption that a one dimensional model adequately describes the hydraulic characteristics of the IHX in the natural circulation flow range (1% to 4% flow). Verification of the one dimensional flow model can be accomplished most efficiently as an integral part of the whole plant performance analysis verification which is discussed in Section 5. It should be noted that the existance of localized phenomena such as buoyancy effects or flow maldistribution are of no concern unless they impact the natural heads and loop flow rates. Therefore, system testing is adequate to verify the IHX model used in DEMO.

The adequacy of the one dimensional flow model for the secondary (tube) side of the EBR-II IHX has been demonstrated by comparisons to test data (Reference 6). The tests at EBR-II centered on determining the acceptability of the heat transfer model and the one dimensional flow model. The results of this

testing was close agreement between model predictions and measured EBR-II data, verifying both the product of the overall heat transfer coefficient times the effective surface area (UA) and tube side one dimensional flow assumption. Post-test analysis of FFTF natural circulation tests is also expected to contribute to the primary side one dimensional flow model verification. Pretest predictions for natural circulation testing at the FFTF facility have been issued in Reference 7.

4.3.3 COLD LEG CHECK VALVES

The hydraulic design of the CRBRP 24" Cold Leg Check Valve (CLCV) is based on hydraulic similitude with the 16" FFTF Check Valve. The flow testing performed on that program, 6" model size and full flow tests on a 16" prototype has provided the data for the 24" valve design. Due to this similarity in design between the FFTF tested valve and the CRBRP valve, no testing of the CRBRP valve was performed. A detailed report (Reference 8) has been prepared which justifies the applicability of existing test data and quantifies the uncertainties in the extrapolation of test data to the CRBRP valves.

The cold leg check valve characteristic is an important part of the verification of natural circulation because it represents approximately 20% of the primary loop resistance at natural circulation flows. Based on the results of the detailed evaluation of the CLCV, the model used in the preliminary evaluation of the natural circulation capability of CRBRP, was found to be overly conservative. Therefore, flows predicted were lower than expected.

4.3.4 PIPING STRATIFICATION AND MIXING, AND HEAT TRANSFER

In DEMO calculations, thermal driving head is calculated as a summation of the sodium densities around the respective primary and intermediate flow loops. Fluid stratification and fluid to structure heat transfer can potentially have a significant impact on natural circulation transients. This impact would be not only on the transient rate, but on the thermal driving head. The total primary system thermal driving head, with the design hot to cold leg temperature difference, is only approximately 1 ft.

Fluid stratification in piping during the transition to and operation at natural circulation conditions has been investigated by test at ANL. The tests demonstrated that for temperature and flow transients during natural circulation events, one dimensional models are adequate for use in codes (such as DEMO) to make predictions of natural circulation performance of piping systems in LMFBRs. The significance of the test results and fluid stratification in LMFBRs during natural circulation events is discussed in a report on the effects of natural circulation induced thermal stratification on plant system simulation (reference 36).

As discussed in Section 4.3.2, tests at EBR-II and FFTF have also shown that a one dimensional flow model is adequate to define the natural circulation flow phenomena. On the local scale there may be some stratification, however, these tests have shown that a one dimensional model adequately defines the response of the system as a whole.

A detailed evaluation of the importance of sodium to structure heat transfer was performed. This study (Reference 10) determined that it is important to account for piping/IHX plena/ pump metal to fluid heat transfer in the analysis of low flow transients where thermal heads are important. Therefore, the "bucket brigade" delay model previously used in DEMO analyses has been replaced. Verification of the sodium to structure models has been accomplished through comparisons with EBR-II data (Reference 6) and will be further verified by data from the FFTF natural circulation tests.

4.3.5 WATER/STEAM SYSTEM NATURAL CIRCULATION

The natural circulation performance of the steam generator system (SGS) does not have a significant impact in the peak core temperatures which occur immediately (approximately 10 minutes) after shutdown. This is due to the long (approximately 10 minutes) loop transport times from the SGS back to the reactor resulting from the low sodium flow rates inherent in the natural circulation mode. However, the SGS does affect the rate of heat transfer from the IHTS later in the transient and, consequently, influences the thermal head and resulting flow in the IHTS. The IHTS flow determines the elevation of the effective thermal center in the IHX and, therefore, is one of the important factors which determine the PHTS thermal head and resulting flow. Therefore, the analytical models for the SGS must be verified to insure that the system will provide adequate natural circulation flow to remove decay heat.

Literature surveys have been conducted to determine the appropriate heat transfer and pressure drop correlations to use in the analytical model of the SGS. As a result of these surveys it has been determined that well established correlations are available for the SGS loop components (piping, valves, and pump) so that additional verification is not necessary. However, data in the literature for the thermal hydraulic performance of heat transferring components at low flowrates is not as well developed. Therefore, it has been concluded that additional data are required to adequately verify the analytical techniques for the steam generator modules to be used in the DEMO computer code. Continuing verification efforts are concentrating on

(1) follow of ongoing industry thermal hydraulic studies

(2) comparisions of calculations with test data from EBR-II

(3) evaluation of tests to be run on the CRBRP prototype steam generator in the Sodium Components Test Installation (SCTI) at ETEC

- Steam Generator Modules

A literature survey has been performed of available information relevant to waterside heat transfer coefficients and heat transfer coefficients for natural circulation conditions. The literature reviewed included published information on natural circulation studies for light water reactors, including foreign facilities, as well as forced circulation studies.

The correlations used in DEMO were also compared to correlations that have been used in other computer programs, including the TRAC computer program (a State of the Art two phase steam and water thermal hydraulic code developed by Los Alamos National Laboratory for NRC). The review covered two phase flow pressure drop correlations, heat transfer correlations for subcooled, nucleate boiling, film boiling and super heat regimes, and departure from nuclear boiling correlations. These correlations govern the heat transfer and flow through the evaporator and are among the most important features of the steam generator system that influence sodium side natural curculation flow. The conclusion of this detailed review was that the correlations used in DEMO are appropriate for the thermal hydraulic conditions during the transition to and during natural circulation. Additional confidence that these correlations are fully applicable to CRBRP will be gained from the SCTI tests of the prototype steam generator.

- Steam Drum

The preliminary analysis (Reference 11) assumed no mixing of auxiliary feedwater (AFW) with drum water during the natural circulation transient. This resulted in a significant cold sodium transient at the exit of the evaporator. The design of the steam drum has since been changed to include an auxiliary feedwater sparger which sprays the AFW into the steam space. This design significantly reduces sodium temperature transients. Additionally uncertainties in modeling the mixing of cold AFW water with the liquid in the steam drum are eliminated since the AFW spray droplets are heated to saturation temperature as they fall to the liquid surface in the drum. Therefore, with the revised design, the need for verifying AFW/drum water mixing models is not required.

- Recirculation Pump

The SGS recirculation pump design, manufactured and testing has been completed. Vendor data for pump operating

performance, coastdown characteristics and stalled rotor hydraulic impendance is being incorporated into the DEMO analytical model of the recirculation pump during the transient to natural circulation.

- Piping and Valves

Modeling of the SGS piping is being updated to the final arrangement. Vendor data for the various valves in the system will be incorporated into DEMO as the data becomes available from the vendor's design efforts.

- EBR-II Tests

The DEMO steam generator model has been used to calculate EBR-II transients for a scram with loss of flow. As discussed in section 5.2, this comparision demonstrates that the sodium side and steam generator models are properly coupled and that they reliably predict overall system dynamic response.

- SCTI Tests

Tests on a full-scale prototype steam generator module are planned at the ETEC SCTI facility. Both steady state and transient tests will provide data which can be used in verifying the steam generator performance and the validity of the DEMO analytical model of the SG module.

Steady-state thermal hydraulic performance will be obtained at low levels representative of steam generator operation under plant emergency decay heat removal conditions to evaluate steam side two phase flow stability and sodium side temperature stratification characteristics. This testing will employ low flow sodium (shell) side forced circulation with water (tube) side natural circulation.

A set of transient tests will be performed which will envelope the fastest and slowest sodium temperature decrease rates anticipated for initiation of natural circulation in the Clinch River plant. These transients have been developed to be within the design capabilities of the prototype steam generator. They will employ programmed changes in sodium flows and temperatures with water (tube) side natural circulation.

Analyses of the steady state and transient tests will be conducted to verify the models used to predict CRBRP natural circulation.

Review of pertinent literature to date and the ongoing DEMO verification efforts indicates that natural circulation in the CRBRP SGS is a viable method of removing decay heat. The remaining efforts are focused on increasing the confidence in the accuracy of the calculations specific to the CRBRP SGS and its components, particularly the steam generator modules. The approach being taken is to perform steady state and transient thermal hydraulic performance tests on the CRBRP prototype steam generator water side operating in the natural circulation mode. The resulting data will be used to further verify the correlations and models of the steam generator to be used in the DEMO code. The correlations used in the model and a reference or justification for their use will be provided in an updated Natural Circulation Report.

5.0 VERIFICATION OF WHOLE PLANT ANALYSIS

Tests conducted at two separate reactor plant facilities have become an important part of the natural circulation verification program. Secondary loop and core testing conducted at EBR-II and plant testing conducted at FFTF provide a significant contribution to the verification of analysis methods used to predict CRBRP performance during a natural circulation transient. The comparison of predictions using the computer codes employed for CRBRP transient analysis against FFTF and EBR-II testing data not only provides detailed verification of specific models, but also verifies that the computer codes properly integrate the models required to adequately characterize a loop type LMFBR over the range of plant conditions sustained by the plant during a natural circulation transient.

5.1 FFTF TESTING

A series of natural circulation tests were conducted at the FFTF as a part of its Acceptance Test Program. These tests not only demonstrated the FFTF's capability to remove decay heat by natural circulation but when coupled with pretest predictions, verified the analysis methods employed for making predictions of natural circulation behavior for conditions other than those tested.

The FFTF tests which are being used to verify the codes used for CRBRP natural circulation analyses are as follows:

A. A transition to natural circulation in both primary and secondary loops from 35% power and 75% flow. One of the secondary loop pony motors remained engaged during the test.

B. A transition to natural circulation in both primary and secondary loops from 75% power and 75% flow.

C. A transition to natural circulation in both primary and secondary loops from 100% reactor power and flow.

Pretest predictions and post-test analysis including comparisons with actual data from these tests are a key element in the whole natural circulation verification program. These tests have provided a unique opportunity for demonstrating that the DEMO, COBRA-WC and FORE-2M codes used for CRBRP analysis provide adequate (conservative) predictions of the dynamic response of a similar, loop-type, sodium cooled reactor.

From the standpoint of whole-plant analysis (using the DEMO code), good agreement between pretest predictions and FFTF results will demonstrate:

- that all models necessary for simulation are included,

- that all phenomena important to natural circulation predictions have been included, and
- that component interactions (synergisms) do not produce unexpected, significant results.

This activity will also verify pump dynamic models, pressure drop models, thermal models (for piping, pumps, heat exchangers), as well as the methods to calculate thermal heads on a dynamic basis.

Since the FFTF employs instrumented assemblies, the above mentioned tests also provide the data necessary to check the intra and inter assembly heat and flow redistribution which are important modeling considerations in the COBRA-WC and FORE-2M codes. Finally, the verification tasks associated with these tests provide the means to demonstrate the practicability of the three-code concept.

There are recognized differences between the FFTF and CRBRP that are important to predictions of plant response to a natural circulation transient. The FFTF employs a dump heat exchanger (DHX) rather than a steam generator as a heat sink. Physical differences in the loops (e.g., piping lengths and elevation differences) alone result in a different effect on the secondary system cold leg temperatures and dynamic thermal heads. The basic phenomena important to making predictions of the responses of these two loops, however, remain unchanged by the above mentioned physical differences. In the reactor area, the FFTF does not contain blanket assemblies and the flow patterns in the upper plenum are different from those predicted for CRBRP. As with the secondary loop, although physical differences in the FFTF and CRBRP reactors exist the phenomina that must be modeled to characterize the response to a natural circulation transient are unchanged.

The basic plant response to a reactor scram with simultaneous sodium pump trips (with de-energized pony motors) is the same for the two plants and the problems of calculating, on a dymanic basis, the thermal heads, flows, core flow redistribution and thus core (including individual pin) temperatures is the same. Only the physical data (mainly geometric) is different.

Best estimate predictions were made, in addition to the design case predictions. The design case predictions were generated by applying uncertainties to the various parameters which can affect the temperatures and flows (such as individual pressure drop correlations, decay heats, etc). Since the predictions were made using design data (i.e.; design information as opposed to plant test data) that existed prior to the conduct of any FFTF plant tests, any predictions made for CRBRP prior to its operation will be based on data having the same level of validity (i.e.; data generated as part of the plant design effort). If when using the same level of uncertainties for these data, predictions are made

for FFTF which are shown to be conservative, when compared to natural circulation test data, the argument is made that the design case predictions for CRBRP will likewise be conservative.

As a part of the post-test analysis effort for the FFTF tests, comparisons will be made with the best estimate calculations, and any differences between these calculations and actual plant data will be resolved. The intent will be to show that the current models are adequate or that one or more need to be modified to bring the best estimate calculations in line with the test data.

The generation of the pretest predictions had to, of necessity, make assumptions with regard to the power history prior to initiation of the test, as well as the heat sink (DHX) boundary conditions. It turns out that these boundary conditions (DHX outlet temperature as a function of time as well as the power-history-influenced decay powers) were not the same for the actual tests as those used in the predictions. This, then, necessitates an upgrading of the predictions as part of the post test analysis effort.

As discussed in the component verifications (Section 4.0) assumptions in reactor outlet plenum mixing, IHX flow, and loop flow modelling will be qualitatively verified. Although components are not instrumented in enough detail to develop exact models of the component response to transients, sufficient data will be developed to determine if the modelling presently contained in DEMO is satisfactory to characterize the overall plant response to the event.

5.2 EBR-II TESTING

The results of EBR-II tests provide a significant basis for verification of parts of the DEMO, COBRA-WC, and FORE-2M models.

For this test evaluation, the DEMO models of the EBR-II IHX and secondary loop hot leg piping were coupled (i.e.; the DEMO calculated IHX secondary outlet temperature was used as an input to the piping calculation); the measured IHX primary and secondary inlet flows and temperatures provided the boundary conditions, and the calculated temperature response at the superheater inlet was then compared with measured values.

As discussed in detail in Reference 6, the close agreement between analsyes and measured data for this testing demonstrates that the numerical techniques embodied in the DEMO programming are adequate to define the IHX and secondary loop piping response to a natural circulation transient event. The importance of sodium to structure heat exchange in piping and plenum models, which was verified by closed form solution (Reference 10), was demonstrated at EBR-II as well as the adequacy of a one dimensional flow model for piping analyses. EBR-II provided significant insight into the IHX secondary side (tube side) modelling requirements. The one dimensional hydraulic model of

the IHX in DEMO adequately predicted the hydraulic response to transients. Therefore, if streaming or tube-to-tube flow oscillations exist, they are not significant enough to impact the overall response of the unit to natural circulation conditions.

The EBR-II staff at ANL has merged a DEMO model of the EBR-II secondary loop and steam generator with their reactor and IHX model (NATCON) to simulate the overall plant. This merged code is called NATDEMO. The ANL analysis of a LOF/SCRAM at EBR-II with NATDEMO (Reference 20) and comparisons with measured data serves to verify the whole-plant modeling capability of DEMO. The verification was based on the conclusion of this study, that NATDEMO simulates the system well enough to accurately predict natural circulation transients. The parallel between NATDEMO and DEMO provides further verification of the methodology of calculating pressure drops, thermal heads, and loop heat capacities that are planned for use in evaluating CRBRP natural circulation transients.

Verification of the COBRA-WC and FORE-2M models centered on the results of data collected from two instrumented core assemblies (XX07 and XX08). Several different categories of transients are experienced with regard to primary and secondary flow variations and initial power and primary flow conditions. Two sets of tests were selected based on constant secondary system flow and high initial (pre-tripped) flow and decay power conditions. These are Test F of assembly XX07 and Test 7A of assembly XX08.

Test F from the XX07 series involved a loss of primary forced flow which was being provided by the auxiliary pump while the reactor was shut down and the fission-product decay power was 1.6% of rated power (60 MWt). The secondary flow was held constant at 2% of its rated value. Among the flow and temperature measurements through the instrumented fueled driver assembly, coolant temperatures at inlet, mid-core and near-core exit were used to compare with FORE-2M code predicted values.

Test 7A from XX08 series initiated under steady state operating conditions of power at 17.1 MWt (28.5% rated power) and a flow rate of 2626 gpm (32.1% of rated flow). The transient was initiated by interrupting the electrical power supply to the motor-generator set driving the primary pumps, the primary auxiliary pump having been previously de-energized. The secondary forced flow was maintained during the initial three minutes of the transient. Comparisons with code predicted coolant temperatures were made through "near center" elements (average of four fuel elements near the assembly center). Comparisons were also made with COBRA-WC code predictions.

Selected EBR-II experimental data on the assemblies were compared with FORE-2M code calculations. Assembly averaged values are used for XX07 comparison. As such, the heat, pressure drop and flow redistribution within the assembly and the inter-assembly heat transfer were not considered. In the case of XX08 assembly

data (near center subchannels are treated) the intra-assembly flow redistribution was simulated by varying the flow maldistribution factor. In general, the FORE-2M model over-predicts the maximum core coolant temperature reached during the natural circulation transients. By considering the flow redistribution alone, the FORE-2M predictions agree very well with XX08 assembly test 7A data in "near center" subchannels. The largest differences in coolant temperature comparison cases occur when near corner element measurements were compared with FORE-2M results. Edge channel overcooling by inter-assembly heat transfer and intra-assembly flow redistribution are the causes of the deviation. When these factors are taken into account, as was done in the COBRA-WC model, the agreement was improved. A refined FORE-2M model was established for FFTF natural circulation transient test calculations to incorporate these factors.

Considering the power and flow measurement uncertainties of about +10% in these tests, it results in a band of 10-20°F variation in the measured coolant temperature. Therefore, the FORE-2M results are within the region of test data uncertainty. More importantly, in all cases, FORE-2M predictions of maximum temperatures are conservative.

6.0 Natural Circulation Verification Schedule

Component System Verification	Completed to Date	CY 1981		CY 1982				CY 1983			
		3	4	1	2	3	4	1	2	3	4
Steam Generator	1, 2							3	2		
Pump			4						5		
Check Valve	6										
IHX	7				8						
Piping	9				8						
Decay Power	10			11				12			
Upper Plenum	13				8						
Reactor Flow Redistribution	14, 7				8						
Reactor Pressure Drop	15, 16, 17, 19, 21, 24, 25, 26			24		26	23, 18				

1. Component Detailed Analysis.
2. Natural Circulation Report
3. SCTI Module Testing.
4. Prototype Pump Water testing by vendor.
5. Prototype Pump Sodium Testing at SPTF.
6. Check Value Hydraulic Analysis Report,
7. EBR-II Test Verification.
8. FFTF Post Test Analysis Complete.
9. Piping Stratification Study Complete,
10. Assessment of Fast Fission Decay Energy Release Uncertainties.
11. Isotopic Fission Capture Rate Verification Fm ZPPR Exp.
12. Implement Methodology in Design.
13. Scale Model Testing.
14. ORNL 61-Rad F/A Testing.
15. 1/4 Scale Inlet Plenum Feature Testing.
16. 1/4 Scale IRFM Reactor Testing.
17. IRFM ΔP Testing @ Low Flows.
18. LIM Testing and Radial Blanket Orificing.
19. Full Scale Inlet Nozzle Water Test.
20. Cavitation and Orificing Calibration Test.
21. F/A Flow and Vibration Water Test.
22. F/A Outlet Nozzle Flow Test.
23. R/B Assembly Flow and Vibration Water Test.
24. Full Scale Control Assembly Hydraulic Water T
25. Full Scale RRSA Orifice Pressure Drop Tests.
26. 1/4 Scale IRFM Testing (UIS-RRS Gap ΔP at low Flows)

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