Department of Energy
Washington, D.C. 20545

Docket No. 50-537
HQ:S:82:070

JUL 15 1982

Mr. Paul S. Check, Director
CRBR Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regualtory Commission
Washington, D.C. 20555

Dear Mr. Check:

RESPONSES TO REQUEST FOR ADDITIONAL DOCUMENTATION

Reference: RAPIFAX from T. King, "Documentation Desired as a Result of June 22, 1982 Meeting on CRBR PSAR Chapter 4.4," dated June 24, 1982.

This letter transmits PSAR Chapter 4 pages which have been modified and provide the information requested in the referenced RAPIFAX (Enclosed). These pages will be incorporated into Amendment 69 to the PSAR, scheduled for submittal in July.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of the
Clinch River Breeder Reactor
Plant Project
Office of Nuclear Energy

Enclosures

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D001

JM725D, 15:38 [redacted]

To: Kirk Petermann (TO)

G. Hussey (ARD)

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DOCUMENTATION DESIRED AS A RESULT OF 6/22/82 MEETING ON CRBR PSAR-CHAPTER 4.4

1) Modify PSAR or provide a separate letter clarifying the Projects plans on 2-loop operation (i.e., not planned for first core operation and not included as part of first core operating license request).

2) In paragraphs 8 and 10 of Section 4.4.1 of the PSAR, reference should be made to Section 4.2.2.1.3 of the PSAR for the conditions which must be met in orificing flow to the reactor vessel internals.

3) At the end of the 3rd paragraph of Section 4.4.3.2.1 add a statement that the uncertainties and confidence levels of the hot channel factors and the affects of a non-linear application of the hot channel factors will be evaluated in the FSAR.

4) Document the margin provided to the 30-year lifetime components regarding their steady state design temperature (i.e., what is their predicted temperature, including uncertainty, versus their design temperature?).

5) For core replaceable components document rational as to why PEOC $+2\sigma$ temperatures are used for steady state and anticipated transient analysis whereas THDV $+3\sigma$ temperatures are used for unlikely and extremely unlikely event analysis. This should address whether or not cladding failure is considered a safety issue.

6) Document the maximum flow thru primary control assemblies (assuming primary pumps are at their maximum speed) and the required flow for floatation.

7) Reference experimental data used for flow distribution calculations at low flow (Section 4.4.26).

8) Change paragraphs 5, 6 and 7 of Section 4.4.1 to state that no melting is allowed (i.e., not just no centerline melting).

9) Provide a clarification on the Project position on fuel lifetime (i.e., operating license is only for 80,000 MWD/MT burnup).

10) Clarify what parameters were used as guidelines and what parameters were used as design limits in developing the reactor vessel component flow allocations. Address such items as SELT, DELT, TELT, lm, 1550°F transient temperature limit, no boiling limit, 900°F vessel temperature, etc.

c) Requirement - Transmit the applied loads from the reactor core assemblies and the Upper Internals Structure to the Core Barrel including upward vertical loads.

Bases - Transfer loads to the Core Support Structure, the primary structural support and positional reference for the Core Former Structure.

d) Requirement - Provide a structural attachment for the Fixed Radial Shield.

Bases - Maintain lateral restraint at the upper end of the Fixed Radial Shield.

e) Requirement - Provide a temporary vertical support for the Upper Internals Structure.

Bases - React the dead weight of the Upper Internals Structure during installation of reactor components.

4.2.2.1.3 Design Loading

The loading conditions to which the reactor internal structures may be subjected are categorized into Normal, Upset, Emergency, Faulted, and Design Conditions as defined in Section III NG & NB-3000 of the ASME Boiler and Pressure Vessel Code.

Table 4.2-21 provides for the 30-year life reactor internals components the design temperatures versus the predicted steady-state temperatures (including uncertainties) at the maximum temperature point of the components.

Design loading conditions are given for the two principal groups of reactor internals components, the upper internal structure and the lower internals structure.

The only structural component of the lower internals structures is the core support structure. Thus the temperature, pressure and static loads for the lower internals which follow are stated for the core support structure.

TABLE 4.2-21

DESIGN TEMPERATURES VS PREDICTED STEADY-STATE TEMPERATURES
FOR PERMANENT REACTOR INTERNALS COMPONENTS

Component	Design Temperature (°F)	Predicted 'Maximum' S.S. Temperature (1) (°F)	Minimum Margin (°F)
Core Support Structure			
Core Plate	775	750(2)	25
Module Liner	775	750(2)	25
Core Barrel	1060	1010	50
Bypass Flow Module	775	750(2)	25
Fixed Radial Shield	950	932	18
Horizontal Baffle Assy.			
FT&SA Support Block	775	750(2)	25
HBA Base Plate	1020	1015(2)	5
Core Former Structure			
Lower Ring	928	928	-
Cylinder	937	937	-
Upper Ring	1076	1076	-
Lower Inlet Module	775	755	20
Upper Internals Structure	1220	1191	29

Notes:

(1) All values shown include a 2σ uncertainty.

(2) Coolant temperature. Actual component temperature slightly lower.

4. No fuel melting is allowed in the fuel assemblies at 115% overpower conditions(*), including design and experimental uncertainties at 3σ confidence level. Consequently, the linear power rating will not exceed the limiting power-to-melt under the aforementioned conditions.

5. No fuel melting is allowed in the blanket assemblies at 115% overpower conditions(*), including design and experimental uncertainties at 3σ confidence level. The blanket management scheme will therefore be arranged not to exceed the limiting power-to-melt under the aforementioned conditions.

6. No absorber melting is allowed in the control assemblies at 115% overpower conditions(*), including design and experimental uncertainties at 3σ confidence level.

7. The sodium temperature exiting the core assemblies will be consistent with the limitations reported in Section 4.2.2.1.3.2 to assure the structural integrity of the upper internals structure during its prescribed lifetime.

8. Mixing in the inlet and outlet plena will mitigate the effects of thermal transients on the internal structures, such that the components structural requirements are met.

9. Adequate cooling shall be provided to the shielding, core barrel and core former components to yield a thermal environment capable of assuring their structural integrity as specified in Section 4.2.2.1.3. Sufficient flow shall be provided to the reactor vessel thermal liner to limit the vessel wall temperature below 900°F during normal operation. Adequate cooling shall be provided for the Fuel Transfer and Storage Assembly to preserve the structural integrity of stored fuel assemblies.

10. Adequate heat removal by forced and free convection from heat producing reactor components shall be assured for all operating conditions.

11. During operating conditions, fuel, blanket and control assemblies total pressure drop along with the rest of the primary system pressure drop will be within the primary pump head capability at the corresponding flow.

12. Coolant velocities shall be less (unless test data support higher acceptable velocities) than the following limits dictated by cavitation and/or corrosion/erosion considerations: 30 ft/sec for non-replaceable components; 40 ft/sec for replaceable components in the high coolant temperature region (exit); 50 ft/sec for replaceable components in the low coolant temperature region (inlet).

(*)This definition means a power equal to 115% of rated power conditions.

13. The control assemblies flow rate will be such as to assure adequate margin against flotation in case the driveline becomes accidentally disconnected (see Section 4.2.3.1.3).

14. Assemblies orificing will be designed to be consistent with the requirement that the lower shield in the fuel, blanket and control assemblies will have sufficient solid volume fraction to limit radiation damage to the core support structure and to assure its prescribed lifetime.

15. The thermal-hydraulic design of the control assemblies will be such as to satisfy the scram insertion requirements during the reactor lifetime (see Section 4.2.3.1.3).

16. The sodium temperature shall be less than its boiling point during normal operation and anticipated and unlikely transient conditions.

17. The reactor will meet the aforementioned design bases operating over a range of power and flow rates, including power ranges and flow variations, from 0 to 100% of nominal conditions.

18. Adequate design margins (see Section 4.4.3.2) will be provided to account for design, fabrication, operational uncertainties and tolerances to ensure meeting the aforementioned limitations. The semi-statistical hot channel factors approach will be adopted in combining individual fuel, blanket and control assembly uncertainties.

19. As explained in Section 4.4.3.3.1, plant T&H design conditions are considered in performance evaluations of permanent plant components(+), e.g., vessel, internals, heat exchangers. Therefore, these conditions shall be considered in evaluation of items 7 through 10, 16 through 18. On the other hand, plant expected operating conditions are adopted in steady state performance and design evaluations of replaceable components such as the reactor assemblies. Therefore, plant expected operating conditions shall be considered in evaluation of items 1 through 6, 11 through 15, 17 and 18.

4.4.2 Description

4.4.2.1 Summary Comparison

This section presents a comparison of general and core assemblies design parameters for the CRBRP and FFTF reactors.

I. CRBRP AND FFTF GENERAL PARAMETER COMPARISON

	Units	CRBRP**	FFTF*
Design Life	Yrs.	30	20
Reactor Power (Thermal)	MWt	975	400
Primary Coolant	-	Sodium	Sodium
Primary Coolant Design Flow Rate	10^6 lbm/hr	41.45	17.28
Coolant Temperature:			
Reactor Vessel Inlet	°F	730	600
Reactor Vessel Outlet	°F	995	858
Reactor Vessel Temperature Rise	°F	265	258
Pressure Drop:			
Reactor Inlet Nozzle-to-Outlet Nozzle (ϕ)	psi	123	110
Lower Inlet Module to Assembly Outlet Nozzle	psi	116	101
Primary Pump Design (static)	psi	160.3	182.5
Number of Primary Loops	-	3	3
Suppressor Plate	-	Yes	Yes
Cover Gas	-	Argon	Argon
Cover Gas Pressure (nominal)	psig	0.36	0.36
Allowable Overpower	percent	15	15

(+)Permanent plant components are those components which: 1) will be designed for 30-year life; and 2) cannot be easily replaced.

*FFTF Initial Condition

**CRBRP T&H Design Values

ϕ CRBRP value includes uncertainties; FFTF value is nominal.

Outlet Plenum

All fuel, blanket, control, and a portion of the radial shield assembly flow discharges into the upper internals structure. The coolant first enters a mixing chamber before entering the chimneys (Figure 4.4-8). The chimneys duct the flow vertically upward and discharge the flow into the upper region of the vessel outlet plenum. The flow is directed into the upper region of the plenum to minimize flow stratification in this region during a reactor trip transient.

The flow from some of the removable radial shields which are located outside of the peripheral skirt of the upper internal structure discharge directly into the outlet plenum. Also, 14% of total reactor flow from the fuel, blanket, control and radial shield assemblies bypasses the chimneys through the gap between the top of the core assemblies and the skirt of the upper internals structure and discharges directly into the outlet plenum.

The coolant leaves the reactor vessel outlet plenum through three 36-inch diameter outlet nozzles.

4.4.2.5 Fuel and Blanket Assemblies Orificing

4.4.2.5.1 Orificing Philosophy, Approach and Constraints

Core orificing, i.e., flow allocation to the various fuel and blanket assemblies is an important step in the core thermal-hydraulic design. Since the assembly temperatures are directly dependent on the amount of flow and since the flow allocation is the only thermal-hydraulic design parameter which can be varied, within certain limits, by the designer, it logically follows that the core T&H design and performance is only as "good" as the core orificing. Therefore, much attention in the CRBRP core T&H design has been placed on core orificing.

Orificing analyses do not provide the final design results. Following the orificing, T&H performance parameters of the core assemblies are predicted. Using these predicted performance parameters, actual design calculations are conducted to assess the adequacy of the design. If all the design constraints were already factored in the orificing, no further iteration would be necessary. Although exact prognostication and correct representation of all the constraints is not always possible, a priori consideration of the design constraints as orificing guidelines nevertheless serves as a useful means in enhancing the efficiency of the analysis process. This was the approach adopted in CRBRP core T&H analyses where a systematic orificing analysis was developed, which accounted for lifetime/burnup, transient, upper internals temperature constraints. This new approach represented a change in philosophy and a significant improvement over the previous maximum temperature equalization method. Characteristic features of this approach are determination of the limiting temperatures (see Section 4.4.2.5.2) for all types of assemblies and simultaneous orificing of the fuel and blanket assemblies. Finally, both first and second core conditions were investigated in determining the orificing constraints and the most restrictive in either core was used in deriving the orificing configuration. This guaranteed, a priori, that the thermal-hydraulic performance would satisfy the constraints considered in both cores.

The following orificing constraints (Reference 1) are satisfied in selecting the flow orificing for the CRBRP fuel, inner blanket and radial blanket assemblies:

- Maximum cladding temperature must be compatible with lifetime and burnup objectives, which can be expressed in terms of maximum allowable inelastic cladding strain and cladding cumulative damage function (CDF);

- Maximum coolant temperature conditions must be such as to assure, with adequate margin, that no boiling occurs during the worst emergency transient (e.g., the three-loop natural circulation event), accounting for uncertainties at the 3 level confidence;

- Maximum assemblies mixed mean outlet temperature and radial temperature gradient at the assemblies exit must be compatible with upper internals structure (UIS) limitations;

- Maximum of eight discrimination zones (fuel plus inner blanket) are allowed;

- Flow allocation to fuel, inner blanket and radial blanket assemblies must not exceed 94.0% of the total reactor flow to account for cooling requirements of other reactor components.

Since the heterogeneous core contains a single fuel enrichment zone and because the number of required discriminators depends on the unique combinations of flow orificing and fuel enrichment zones, the maximum number of fuel plus inner blanket assembly orificing zones is equal to the total allowable number of discriminators (i.e., 8). Inner blanket and fuel assemblies employ identical inlet nozzles. Therefore, both must be considered in determining the total number of discriminator zones. The outer blanket assemblies employ a unique inlet nozzle and, therefore, are not considered in determining the total number of discriminators. The two 6 corner positions(*) which alternate between inner blanket and fuel assemblies during successive cycles, form a separate discriminator zone which is included among the eight.

To put the lifetime/burnup and transient temperature constraints on the same basis and to provide quantitative, comparable orificing guidelines, the concept of equivalent limiting temperature is employed. The equivalent limiting temperature is defined as that cladding temperature at a specified radial position (cladding ID in these analyses) and time in life (end-of-life) which must not be exceeded in order to satisfy the considered constraint.

(*) A map of the 60° core symmetry sector analyzed in the thermal-hydraulic studies and assemblies numbering scheme are shown in Figure 4.4-9.

Three equivalent limiting temperatures were defined to represent the lifetime/burnup and transient constraints, i.e., SELT, DELT and TELT. They are defined as the end-of-life maximum cladding ID temperatures for Plant Expected Operating conditions (see Section 4.4.3.3.1), considering uncertainty factors at the 2 level of confidence, such that accounting for the assembly temperature/pressure lifetime history, the limiting value of the inelastic cladding strain (SELT), or cumulative damage function (DELT), or worst time-in-life transient coolant temperature (TELT) is not exceeded. As it appears from the above definition, the equivalent limiting temperatures are calculated for each assembly. In fact, all the various assemblies have individually different lifetime histories of cladding temperature and fission gas pressure, and therefore, the limiting equivalent temperatures are necessarily different from assembly to assembly to stay within a constraint common to all assemblies. Calculations are performed for plant expected operating conditions, which are the conditions where the CRBRP is expected to operate on a probabilistic basis and the conditions used in the design of replaceable components such as the core assemblies (see Section 4.4.3.3.1).

As previously mentioned, both first and second core conditions have been considered in defining the core orificing, therefore, the SELT, DELT and TELT have been calculated for both cores. In the case of the radial blanket assemblies, where the lifetime spans both cores, obviously only one set of limiting temperatures was calculated. Using the ØCTØPUS code, the assemblies minimum flow in the first and second core necessary to satisfy the most restrictive of the limiting conditions was calculated for each assembly. Subsequently, the various assemblies were grouped in zones and the orificing arrangement was selected such that the flow allocated to each assembly was at least equal to the larger of the flow requirements in first and second core. This assured meeting all constraints for both cores. Finally, the excess flow, if any, is allocated among the fuel assemblies to minimize and equalize the assemblies exit temperature and temperature gradients.

4.4.2.5.2 Calculation of Equivalent Limiting Temperatures

Assemblies lifetime/burnup goals are achieved when both the cladding inelastic strain and cladding CDF are within the established limits during steady-state operation. The ductility strain guideline was set at 0.2% and the CDF guideline for orificing analyses was set at 0.7 in the fuel assemblies and 0.5 in the blanket assemblies. Since the CDF limit for steady-state plus transient operation is by definition 1.0, the margin for CDF transient accumulation was 0.3 in the fuel assemblies and 0.5 in the blanket. Both cumulative cladding strain and CDF depend on the rod cladding temperature/pressure history. Thus, using a preliminary estimate of the assembly flow (but using the proper physics data), the hot rod (*) in each assembly at end-of-life was identified

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using the subchannel analysis code CØTEC. Subsequently, the hot rod was followed throughout lifetime and the lifetime temperature/pressure history was calculated with the NICER code. Uncertainty factors (see Section 4.4.3.2) at the 2σ level of confidence were used in the cladding temperature/pressure calculations. Based on the above lifetime histories, a strain equivalent limiting temperature (SELT) was calculated for each assembly. The SELT serves as an analytical expression of an orificing guideline. It represents the end-of-life temperature which, if maintained constant throughout lifetime, would cause an end-of-life cumulative strain of 0.2% for the particular assembly relative behavior of cladding temperature and pressure through lifetime. Accordingly, the SELT does not depend on any guessed value of assembly flow, but rather on the relative behavior through lifetime, which is only a function of the power generation changes during life.

Since the DELT is the equivalent end-of-life temperature corresponding to a CDF of 0.7 or 0.5, the method employed in its determination was to extract it from a curve correlating the cladding ID temperature at EOL with the corresponding CDF. Thus, at least three (in some instances more were necessary) lifetime temperature/pressure histories were generated for each assembly by varying the flow and the corresponding CDF was calculated. Typical curves are reported in Figures 4.4-10 through 4.4-14 for the fuel and inner blanket assemblies (first and second cores) and radial blanket assemblies. By interpolation, the DELT corresponding to the CDF constraint was then determined.

Regarding the transient constraint, the design guideline is to provide adequate margin-to-sodium boiling throughout the assembly lifetime during the worst transient. This was quantitatively translated into an orificing guideline of 1550°F which was conservatively defined as the maximum coolant temperature allowable during a natural circulation transient in any assembly at any time in life accounting for uncertainty factors at the 3 level of confidence. This guideline also assumes plant THDV conditions and a 750°F reactor inlet temperature.

(*) Each assembly is characterized by its hot rod at end-of-life, which is obviously the one with the highest strain and CDF.

the first and second cores, minimum flows must be put on the same basis. Cycle 4 was chosen as the standard basis since it will require the higher core flow fraction (fuel assemblies are in alternating row 6 positions). When flow requirements for cycle 2 are translated to cycle 4 equivalent values, second core requirements are found to be slightly more restrictive in some outer fuel assemblies, as shown in Figure 4.4-17. Cycle 5 flows are reported for the transient limited second row radial blanket assemblies, since their TELT's are maximum at EOL.

Using the required minimum flows as guidelines, the ØCTØPUS code selected, for a given number of orificing zones, that combination of assemblies grouping into orificing zones which among all the various possible combinations, yielded the minimum value of total core flow and was therefore the most effective. As mentioned in Section 4.4.2.5.1, a maximum number of eight discriminators (and orificing zones) is allowed for the fuel and inner blanket assemblies. Four orificing zones in the radial blanket assemblies were chosen, thus, the total number of core orificing zones resulted equal to 12. The selected arrangement is reported in Figure 4.4-18, where the starred assemblies are the ones which determine the amount of flow allocated to the orificing zones (they are called zone driver assemblies, or drivers). Also indicated are the limiting assemblies in each orifice zone for first and second core; obviously the driver is the one with the more restrictive flow requirement (compare with Figure 4.4-17).

As shown in Figure 4.4-18, the orificing arrangement does not have a 30° symmetry because the control rod location and insertion pattern, hence the power generation, does not have a 30° symmetry. For example, considering the assemblies around the row 7 corner control assemblies (see Figure 4.4-16), first core conditions are limiting for the fuel assemblies around the control assembly at the right of the figure, while second core conditions are prevalently limiting for the fuel assemblies surrounding the control assembly at the left.

The minimum amount of core flow necessary to satisfy the various constraints and the grouping of the core assemblies into 12 orificing zones was equal to 93.07% of the total reactor flow of cycle 4 conditions. Since 94% of the total reactor flow is allocated to the fuel and blanket assemblies and since 93.07% is the minimum required to meet the conservatively selected constraints, it follows that slightly less than 1% of the total reactor flow is available to be allocated as deemed desirable by the designer. Usually, if a significant amount of excess flow is available, this is distributed among the fuel assemblies to minimize/equalize the assemblies mixed mean temperature and temperature gradient. This was not, however, the procedure adopted in these studies since the amount of available excess flow is not enough to significantly influence the value of the outlet temperatures. Additionally, the relative assemblies power generation and the sophisticated orificing, which are characteristics of this heterogeneous design, yielded maximum differences in exit temperature (see Sections 4.4.3.3.3 and 4.4.3.3.5) between two adjacent assemblies (which generally occur at the fuel/inner blanket interface in rows 6 through 8) within the UIS capability. Therefore, the excess flow was distributed roughly evenly among the various core orificing zones. The final core flow allocation is reported in Table 4.4-4, which shows the cycle-by-cycle variation of flow in the various orificing zones. Both thermal-hydraulic design value (THDV) and plant expected operating condition (PEOC) flows are reported in Table 4.4-4.

Subsequent performance predictions and design calculations reported in Sections 4.4.3.3 and 4.2.1 demonstrated that the core orificing so determined was adequate and that design constraints and objectives were met.

4.4.2.6 Reactor Coolant Flow Distribution at Low Reactor Flows

The normal mode of CRBRP core heat removal upon reactor shutdown is by forced circulation from AC powered pony motors (which have emergency backup power from diesel generators) driving the primary pumps. However, the CRBRP has been designed to have the added capability of adequate cooling by means of natural circulation. This inherent emergency coolant flow is provided by the thermal driving head developed by the thermal center of the IHX being elevated above that of the core (plus the respective elevation differences in the intermediate loops and steam generator system).

At the ~10% pony motor flow level after shutdown, insignificant flow redistribution occurs between the parallel flow core assemblies. However, 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 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. This has been demonstrated in the EBR-II and FFTF natural circulation experiments (Ref. 68 and 79).

In addition to the in-pile data, a large out-of-pile data base exists to characterize the flow behavior of the various components over a wide range of operation, including low flow conditions. A listing of the experimental data references for flow distribution calculations is provided in Table 4.4-36.

Independent studies outside the CRBRP Project have been published which show a significant decrease in predicted maximum core temperatures due to reactor flow redistribution during natural circulation conditions. For example, Brookhaven National Laboratory (Agrawal, et.al., in Ref. 69), using the SSC-L code, predicted localized flow increases as large as 20% in the hot fuel assembly and 40% in hot blanket assembly for the CRBRP during natural convection cooling. Corresponding reductions in the predicted maximum transient coolant temperature on the order of 16 and 22% (130°F 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 70 using the CURL-L code. For these studies, inter-assembly heat transfer as well as intra-assembly flow redistribution and heat conduction effects were neglected. Inclusion of these effects would further reduce the maximum core temperatures.

Preliminary studies with CØRINTH have been performed 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 which were neglected are discussed later. Figure 4.4-66 shows the results of these analyses for the peak fuel, peak inner blanket and peak radial blanket assemblies. Figure 4.4-67 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.

To assess 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 CRBRP core, the following system of three computer codes is used:

1) DEMØ - predicts the overall plant-wide, dynamic natural circulation performance and defines the core boundary conditions;

2) CØBRA-WC - predicts the detailed dynamic, core-wide performance including all inter- and intra-assembly flow and heat redistribution effects;

3) FØRE-2M - predicts the localized hot rod dynamic temperatures including effects of localized rod phenomena and uncertainties in nuclear/thermal-hydraulic/mechanical data.

A linkage between the CØBRA-WC and FØRE-2M codes has been developed to incorporate the inter- and intra-assembly phenomena into the localized hot rod transient analyses by using the expression for the heat transported to the coolant for each axial node of the hot element modeled in FØRE-2M. Coupled with this, the axial mass flow rate for each axial node is also input from CØBRA-WC analyses. The heat and axial mass flow rate for each axial node are based on nominal conditions in the CØBRA-WC code. This is a conservative approach because these values are lower than those calculated for the hot channel temperature conditions and thus, result in a conservatively higher predicted hot channel temperature.

Based on a 90%/10% Pu (239 + 241)/U-238 fission rate split, the weighted average fission gas yield value may be calculated directly from the data presented in Table 4.4-15. The value of the Xe + Kr fission yield in fuel rods resulted equal to 0.249.

For the blanket case, the fuel isotopic composition, and hence, the isotopic fission rate, changes significantly with burnup (plutonium accumulation). For a fresh assembly in either the inner or outer blanket, about 90% of the fissions occur in U-238, and the remaining 10% occur in U-235. Therefore, the beginning-of-life fission gas yield is equal to 0.240.

At end-of-life, just prior to discharge, the breakdown of fissions is as follows: In U-238, 33% for inner and 16% for outer blanket; in U-235, 2% for both inner and outer blankets; in Pu-239, 65% for inner and 82% for outer blanket. Thus, the fission gas yield calculated from data in Table 4.4-15 is 0.247 in inner blanket assemblies at EOL and 0.249 in outer blanket assemblies at EOL.

Conservatively, a nominal fission yield of 0.249 constant throughout life for both fuel and blanket assemblies was adopted.

The isotopic uncertainty in the ENDF/B-IV fission yields results in a $\pm 3.5\%(1\sigma)$ uncertainty in the rare gas (Xe+Kr) yield from U-235, U-238 and Pu-239 fissions. Therefore, the 2σ fission yield adopted in plenum pressure calculations was equal to 0.266.

The substantial conservatism in calculating plenum pressures is discussed in Section 4.4.3.2.4, together with a quantitative evaluation of the over-estimation of plenum pressure for two typical blanket rods.

4.4.2.9 Thermal Effects of Operational Transients

Current design practice is that LMFBR components must meet the required conditions of ASME Code Section III (Ref. 43) and RDT Standard C-16-1T (Ref. 44). Transient reactor design events are divided into categories of normal, upset, emergency and faulted according to their likelihood of occurrence. Table 4.4-16 gives: a) the definitions for the various incidents; and b) the allowable severity with respect to structural consequences. Note that the RDT Standard respective terminology for the events are: normal operation, anticipated fault, unlikely fault and extremely unlikely fault.

Table 4.4-17 presents a summary of preliminary design criteria (Limits and Guidelines) for emergency and faulted events to assure that the core operates safely over its design lifetime and meets the requirements of the ASME Code and RDT Standard. The frequency of occurrence and classification of events is established by the designer based on industrial and nuclear experience and also the special characteristics and differences in LMFBR design (as compared with an LWR for example).

Under normal steady state operating conditions, the cladding is loaded due to the internal gas pressure. Fission gases are released from the fuel with burnup, and thus, the internal pressure continually increases over the rod's

The primary pump head flow characteristics and reference operating points are presented in Section 5.3.2.3.1 and 5.3.3.3 and Figures 5.3-19, 5.3-20 and 5.3-21. The primary pump flow coastdown is presented in Figure 5.3-22. The intermediate pumps are identical to the primary pumps with the exceptions noted in Section 5.4.2.3.1 with operating characteristics shown in Figure 5.4-3.

4.4.3 Evaluation

4.4.3.1 Reactor Hydraulics

The total reactor flow rate is one of the primary parameters that affect the thermal performance of the CRBRP. The hydraulic analyses include the effects of uncertainties such as: instrumentation errors, correlation uncertainties, experimental accuracy, manufacturing tolerances and primary loop temperature and flow uncertainties.

The method used to perform the steady-state hydraulic analysis consists essentially of identifying all possible flow paths in the reactor, establishing a hydraulic network and solving the network by use of such codes as CATFISH and HAFMAT. Solution of the network will provide reactor flow rate and flow distribution within the reactor for certain specified plant operating conditions, which in the case of the CATFISH code are the pump head/flow characteristics curve. The CATFISH code includes pressure drop analytical correlations obtained from the results of the out of file tests reported in Tables 4.4-36.

The coolant flow distribution is determined by the geometry of the regions through which sodium flows. Their hydraulic impedance establishes the reactor pressure drop and pressure distribution. These paths include inlet and outlet nozzles, inlet and outlet plena, core support structure modules, annulus between radial shielding and core barrel, annulus between vessel and core barrel, annulus between vessel and vessel liner and the core assemblies upper internal structures region. Because of their importance, the resistance and hydraulic characteristics of the main flow paths are determined by scale model tests. The tests conducted for CRBRP are discussed in Section 4.4.4, Testing and Verification. Prior to the availability of data from these tests, the results from similar tests in the FFTF Development Program are used where applicable. Also see Section 4.4.2.7 for a discussion on hydraulic impedance correlations.

In addition to the main flow path, leakage flow paths exist in the CRBRP; these are taken into account in the flow distribution studies, but no credit is taken for leakage flow when satisfying cooling requirements. Seals between the core support structure and the core inlet module liner, between various parts of the hydraulic balance system, etc., form flow paths for leakage. The design objective of the seals is to minimize leakage. Where possible, the piston ring type seal developed in FFTF will be used and others of different design will be evaluated experimentally with the intent to minimize leakage.

4.4.3.2 Uncertainties Analysis (*)

4.4.3.2.1 Introduction

The impact of theoretical and experimental analyses uncertainties, instrumentation accuracy, manufacturing tolerances, physical properties and correlations uncertainties must be considered in predicting the reactor thermal-hydraulic performance to ensure the safe and reliable operation of the CRBRP core and to guarantee that proper margins are provided so as not to exceed the design limits and requirements.

Hot channel/spot factors for all core assemblies have been determined to account quantitatively for the above uncertainties. Consistent with previous studies, the semi-statistical hot spot analysis method is used for the CRBRP core assemblies; i.e., random variables are combined statistically and together with the direct bias uncertainties they characterize a hot channel/spot as the one affected by the simultaneous occurrence of all uncertainties. Predicted hot channel/spot temperatures are the ones to be compared with the required limits.

The preliminary uncertainties analysis made certain simplifying assumptions, such as the overall temperature difference is a linear function of individual variables, statistical uncertainties are normally distributed, and a large number of samples are implicit in the data base. The effect of these assumptions have been investigated in a detailed study (Ref. 19) which showed that the overall uncertainty analysis approach adopted in these analyses is conservative. A full evaluation of the adopted uncertainties, of the confidence levels of the hot channel factors and of the effects of non-linear application of the hot channel factors, will be performed for the FSAR.

Use of the semi-statistical method requires the separation of the variables which cause the hot spot temperatures into two principal groups, one of statistical origin and the other non-statistical. The two categories are defined below.

A non-statistical (or direct) uncertainty is defined as a variable, the exact value of which cannot be predicted in advance, but which

(*) The information specified in the Standard Format and Content for Section 4.4.3.2 "Influence of Power Distribution", is included in Section 4.4.3.3 to enable the inclusion of this major area of T&H analysis as Section 4.4.3.2.

4.4.3.3 Steady-State Performance Predictions

Reported in this section are the analyses performed to characterize the steady-state thermal behavior of the CRBRP core together with highlights of the results. For a much more detailed report of the results, see Sections 4 and 5 of Reference 3.

4.4.3.3.1 Plant Conditions

Two sets of plant conditions are used in the thermal-hydraulic design, i.e., plant thermal-hydraulic design value (THDV) conditions and plant expected operating conditions (PEOC). The THDV conditions (730°F inlet/995°F outlet temperature; total reactor flow 41.446×10^6 lb/hr) are the Clinch River rated plant conditions and are used in: a) analyzing permanent components which have the same 30-year lifetime as the plant; b) transient and safety analyses, since they are more conservative than the plant expected conditions and represent the "worst bound" of plant conditions. The plant expected operating conditions represent the plant conditions at which the CRBR is expected to operate accounting for the operating conditions of the heat transport systems, such as pump characteristics, reactor and primary loop pressure drop uncertainties, fouling and plugging of heat exchangers, etc. During actual reactor operation, the long-term damage accumulated by the fuel and blanket assembly components is expected to correspond to the damage which would be calculated using time averaged nominal temperatures. However, in assessing the effects of steady-state operation and anticipated faults (normal and upset conditions), fuel and blanket assembly component temperatures are based on maximum expected plant operating conditions (PEOC) and upper 2σ levels. At this level, there is a 97.5% probability that the corresponding temperatures are not exceeded. This is conservative since the calculated damage accumulation generally increases with temperature. For the unlikely and extremely unlikely events (emergency and faulted conditions) an upper limit on plant conditions (Thermal Hydraulic Design Values - THDV) and the upper 3σ uncertainty level is used, simply to add additional conservatism for the safety analyses. At this level, the probability of exceeding the calculated temperature is ~0.1%.

The above designated use of plant conditions and uncertainties derives from the premise that stochastic failures are not a safety issue and the plant is capable of operation with limited fuel rod cladding failures. To support safe operation with failed fuel, all the safety analyses described in Chapter 15 of this PSAR are based on continued and extended plant operation with 1% failed fuel.

The primary heat transport system principal parameters (inlet, outlet temperature and ΔT) are evaluated, together with the associated uncertainties. The results of this study for the heterogeneous core, which comprised a Monte Carlo type analysis, are reported in Table 4.4-28. some significant features are: 1) the consideration of the progressive fouling of the heat exchangers during the plant 30-year lifetime, which affects the predicted values of the plant operating conditions (rather than conservatively assuming end-of-life fouling, i.e., after thirty years operation); and 2) a comprehensive accounting of all uncertainties affecting plant operation. Plant expected operating conditions are adopted in core thermofluids analyses of replaceable components, such as the core assemblies, chiefly in determining the fuel rod parameters (cladding temperature, fission gas pressure) which are the basis for evaluating the structural behavior and for assessing whether lifetime/burnup objectives are actually met.

Plant expected operating conditions and associated uncertainties adopted in the thermal performance analyses are reported in Table 4.4-29. Following is a brief discussion of the rationale in determining the values reported in Table 4.4-29 from the ones in Table 4.4-28.

First, the mean values of Table 4.4-28 are chosen as the nominal values of Table 4.4-29, thus, conservatively including the bias factor directly into the nominal values. Since the most critical time for core

For the blanket assemblies, inner blanket assembly 99 was investigated as the blanket assembly having the highest power in the first five years of CRBRP operation. Assembly 99 reaches its maximum power in the second core, at end-of-cycle 4 (see Figure 4.4-33). Inner blanket assemblies envelope with respect to power-to-melt conditions the longer residence time radial blanket assemblies.

Both the hot and the peak rods were investigated, since the peak pin has the highest linear power, while the hot pin has the highest cladding temperature. The cladding temperature has, in fact, a very significant effect on cladding swelling, hence on fuel/cladding gap size, hence gap conductance, fuel temperature and finally on power-to-melt. Thus, both the hot pin and the peak pin need to be investigated. Analysis of the hot pin was obviously not necessary for the fuel assemblies, since their critical time in life is at beginning-of-life, rather than end-of-life as for the blanket assemblies. Finally because the maximum power in blanket assemblies occur at end-of-life, the programmed start-up cannot affect the power-to-melt in the blanket.

The axial positions where the cladding temperature and the linear power rating are maximum were investigated in addition to intermediate positions between the two above. Also considered were: a) when the blanket pins go through a full overpower factor of 1.15 at EOL; and b) when the reactor power is increased to 115% of rated power from the top of the allowed variation, i.e., with an overpower factor of 1.15/1.03.

It was found that the no-melting criterion is fully satisfied in the worst case. The peak pin has 0.4% less margin than the hot pin. When the overpower excursion is a full 15% the margin is 0.4% less than for the case when the reactor power is ramped from 1.03% of the rated power. Substantial conservatism was implicit in the analyses (e.g., in cladding swelling evaluation, adopting a direct combination of nuclear uncertainties), thus, removal of the implicit conservatism and factoring of experimental data when available, would substantially improve the power-to-melt margin.

4.4.3.3.7 Control Assemblies Thermal-Hydraulic Performance

The CRBRP has two control systems: primary and secondary control rod system (PCRS and SCRS) with nine (9) and six (6) control assemblies, respectively. Detailed design features of the systems are provided in Section 4.2.3 (Reactivity Control Systems).

The bases and methodology of the thermal-hydraulic analysis of the primary control assemblies followed that used in the homogeneous core design, reported in Reference 13. A summary of the principal operating parameter for the primary and secondary control assemblies are presented in Tables 4.4-32a and 4.4-32b, respectively. Values reported in Table 4.4-32 are for the row 7 corner assembly, which is the thermally limiting PCA.

Key hydraulic performance assessments relate to the assembly flow margin to control rod flotation and control rod scram dynamics. The PCA E-Spec. requires that the control assembly design shall assure that the control rod cannot be lifted (or floated) from the fully inserted position, under maximum assembly flowrate (and pressure drop) conditions, more than the distance causing a reduction in shutdown reactivity margin equal to the stuck rod margin. This requirement shall apply to all 9 rods, either with the driveline connected to the control rod or to refueling conditions for which the driveline is disconnected and withdrawn to its refueling position.

Both experimental and analytical investigations were conducted to assure the PCA will not float under the worst possible conditions; the results of these investigations are summarized in Table 4.4-33. Data, analytical results and margin-to-flotation were expressed in terms of both assembly flowrate and pressure drop across the absorber bundle. Prototypic testing of the CRBR PCA provided experimental measurements of the PCA flotation characteristics; experimental uncertainties were directly superimposed over the observed values. On the other hand, the maximum flow through the PCA was calculated with the CATFISH code accounting for all the various effects causing a flow variation in the PCA. Specifically, the hydraulic resistance uncertainties in all core components were varied by their maximum value and, conservatively, the absolute variation in the PCA flow and P was taken as increasing the design value. The three leading causes for an increase in the PCA flow were found to be: primary pumps at their maximum speed resulting in a maximum reactor flow equal to 115% of the rated THDV value; PCA orifice resistance at its minimum; and LIM containing the PCA at is minimum resistance allowable.

As reported in Table 4.4-33, the individually induced variations in the PCA flowrate and P were combined at various levels of conservatism, ranging from 2σ and root sum of the squares to 3σ and absolute sum combination. Correspondingly, the flotation margin ranged from 15% to 3% in terms of ΔP and from 9.5% to 5.5% in terms of flowrate. In all cases a large amount of conservatism was included, for example: a) by comparing the minimum experimental with the maximum predicted flotation characteristics, analytical and experimental uncertainties were superimposed rather than combined statistically; b) use of absolute rather than relative values of the PCA flow (and ΔP) variations does not take into account the variations causing decrease, rather than increase of the PCA flowrate and ΔP. In spite of this conservatism, a positive margin to flotation resulted under the worst conditions, as shown in Table 4.4-33.

The secondary control rod system uses the concept of hydraulic scram-assist design with a net hydraulic force in the 150-250 lbs. range on the control rod when fully withdrawn from the core. The same magnitude of downward hydraulic force (in addition to the weight of the assembly) is also available under the abovementioned design conditions. Thus, it is concluded that the secondary control rods do not float at 100% flow (even when disconnected).

Predicted control rod scram performance of the primary control rod system is reported in Section 4.2.3 (Reactivity Control Systems).

Figures 4.4-54 and 4.4-55 show typical PCA absorber region temperature distributions under the minimum withdrawal and full withdrawal control rod conditions, respectively.

4.4.3.3.8 <u>RRS Thermal-Hydraulic Analyses</u>

The steady-state duct temperatures at PEOV conditions were calculated for a 30° sector of the RRS.

The region analyzed is partially shown in Figure 4.4-56. The model consists of all 29 RRSA's in a 30° sector, plus a corresponding section of fixed radial shielding (FRS), core barrel (CB) and core

TABLE 4.4-17

SUMMARY OF PRELIMINARY DESIGN CRITERIA

Event Classification	Severity Level	Criterion**
Emergency	Minor Incident	The total cumulative damage function is to be less than 1.0.
(Unlikely Faults)		The accumulated plastic and thermal creep strain is to be less than 0.3%.
Faulted	Major Incident	No cladding melting (temperature less than 2475°F) and
(Extremely Unlikely		*No sodium boiling (temperature less than saturation temperature at the existing pressure).

*Sodium boiling temperature is quoted as a guideline to establish that no cladding melting can occur.

**The emergency criteria are limits .
The faulted criteria are guidelines.

TABLE 4.4-32a

PRIMARY CONTROL ASSEMBLY OPERATING PARAMETERS

o Number of orificing zones	1
o PCA's total flow allocation (fraction reactor flow)	0.01
o Flowrate (PEOC, lb/hr)	49,500
o Flow split (bundle/total assembly flow)	0.62
o Maximum bundle flow velocity (ft/sec)	8
o Maximum hot rod midwall cladding temperature (PEOC, 2σ, ^{o}F)	1006
o Maximum fission gas pressure (2σ, psia)	3600 @ 275 fpd
o Maximum linear power rating (3σ + overpower, Kw/ft)	16 Bottom 1.4 Top
o Maximum absorber temperature (THDV nominal, ^{o}F)	3367
o Maximum mixed mean exit temperature (THDV nominal, ^{o}F)	853
o Maximum exit gradient (nominal, ^{o}F)	246

TABLE 4.4-32b

SECONDARY CONTROL ASSEMBLY OPERATING PARAMETERS

Flow Rate (THDV, lb/hr)	
Control Rod Flow	9,130
Bypass Flow	9,330
Total Upflow	18,450
Downflow	50,710
Total Assembly	69,170
Hydraulic Scram Assist Force at Full Flow (lbs)	148 - 248
Peak Linear Power (kw/ft)	4.0
Outlet Temperature (THDV, Nominal, ^{O}F)	
Control Rod Bundle	829
Assembly	854
Maximum Cladding Midwall Temperature (^{O}F)	
Nominal, THDV	853
Hot Spot (THDV, 2σ)	895
Maximum Absorber Temperature (^{O}F)	
Nominal, THDV	1054
Hot Spot (THDV, 3σ)	1188

TABLE 4.4-33

SUMMARY OF FLOTATION EVALUATION

	Rod Bundle Pressure Drop (Psi)	Flow Rate (Lb/Hr)
1. Test observed flotation conditions	7.5	57,500
2. Above, accounting for test uncertainties	7.2	55,000
3. PCA operation conditions, nominal, PEOC	5.94	49,600
4. Total effect of core components hydraulic resistance uncertainties:		
o r.s.s. combination $2\sigma/3\sigma$	0.35/0.52	687/976
o Absolute sum combination $2\sigma/3\sigma$	0.73/1.06	1932/2696
5. Maximum design conditions (3+4):		
o r.s.s. combination $2\sigma/3\sigma$	6.29/6.46	50287/50576
o Absolute sum combination $2\sigma/3\sigma$	6.67/7.0	51532/52296
6. Margin-to-flotation (2-5):		
o r.s.s. combination $2\sigma/3\sigma$	0.91/0.74 (15/12)*	4713/4424 (9.5/9)*
o Absolute sum combination $2\sigma/3\sigma$	0.53/0.2 (9/3)*	3468/2704 (7/5.5)*

*In percentage of nominal conditions.

TABLE 4.4-36

EXPERIMENTAL DATA REFERENCES FOR FLOW DISTRIBUTION
CALCULATIONS AT LOW FLOW (SECTION 4.4.2.6)

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