Department of Energy
Washington, D.C. 20545
Docket No. 50-537
HQ:S:82:155

DEC 23 1982

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

Dear Mr. Check:

ADDITIONAL INFORMATION REGARDING PRELIMINARY SAFETY ANALYSIS REPORT (PSAR) CHAPTER 4 "REACTOR" ON THE CLINCH RIVER BREEDER REACTOR PLANT

Reference: Letter HQ:S:82:139, J. R. Longenecker to P. S. Check, "Reactor Design (Chapter 4) Working Meeting, November 25 and 26, 1982 - Additional Information," dated December 6, 1982

Enclosed are the project's responses to the action items identified in the reference letter. These responses concern: Chapter 4.3, Item 1.e) revised response to Nuclear Regulatory Commission QC490.29; Chapter 4.4, Items 1.c) and d) analysis of flow blockage at a fuel assembly outlet and secondary control assembly analysis. The information regarding Chapter 4.3, Item 1.e) and Chapter 4.4, Item 1.d) will be incorporated in Amendment 75 to the PSAR scheduled for submittal in January 1983.

Questions regarding this submittal may be directed to W. Pasko (FTS 626-6096) of the Oak Ridge Project Office staff.

Sincerely,

John R. Longenecker

John R. Longenecker
Acting Director, Office of
Breeder Demonstration Projects
Office of Nuclear Energy

Enclosure

cc: Service List
Standard Distribution
Licensing Distribution

D001
1/40

Enclosure

Revised

Question CS490.29

Although not necessary for review of the PSAR, all codes used in design and evaluation of the fuel and blanket rods will need to be reviewed. Based on our current understanding, the codes to be reviewed will include.

FURFAN

LIFE-III

any of the LIFE-IV series

anticipated to be used for the PSAR

FORE-2M

FRST

Response

The applicant acknowledges that codes used in design and evaluation of the fuel and blanket rods may need to be reviewed by NRC during the PSAR review. Appendix A of the PSAR provides existent information for the various codes cited. Relevant information for the codes will be made available at the time of PSAR submittal to facilitate NRC review.

NRC Concern and/or Defined Resolution:

Describe design features to prevent core assembly outlet flow blockages.

The applicant will provide the NRC with the results of an analysis which demonstrates that should a total blockage of an assembly outlet nozzle occur six hours after shutdown, no significant assembly degradation will result.

Response

The attached provides the results of the above-mentioned analysis.

An analysis was performed to determine the maximum cladding temperature due to complete blockage of a fuel assembly outlet nozzle during refueling. It was found that, even using a conservative analysis, the maximum cladding temperature for this accident (if it occurs after 6 hours from shutdown) would be less than that for operating conditions.

Conditions analyzed include:

- Hottest pin in highest power F/A of all core conditions
- $+3\sigma$ power uncertainty
- Maximum decay power
- 425°F reactor inlet temperature
- 7.5% pony motor flow
- Midplane of active core position modeled

The effective thermal conductivity of the rod bundle was calculated by the equation:

$$K_{eff} = k \frac{2(1-f_d)}{2 + f_d}$$

where f_d = area fraction of fuel rods and wire wrap in assembly

k = sodium thermal conductivity.

Only conduction heat transfer from the affected assembly to the surrounding six assemblies was considered.

Figure 1 shows the maximum cladding temperature (i.e., for center pin) as a function of time from shutdown for the accident occurring. Typically, refueling would not be expected to start in less than two days from shutdown. However, earlier times are shown on the figure for parametric purposes. It can be noted that even at six hours from shutdown, the maximum temperature would be less than 1125°F[*]. The steady state, 3σ inner cladding temperature for the hot rod at operating conditions is larger than this value, and therefore, it would be expected that this accident condition would cause an insignificant increase in cladding damage.

[*] It would take approximately 2 minutes to reach this steady state temperature.

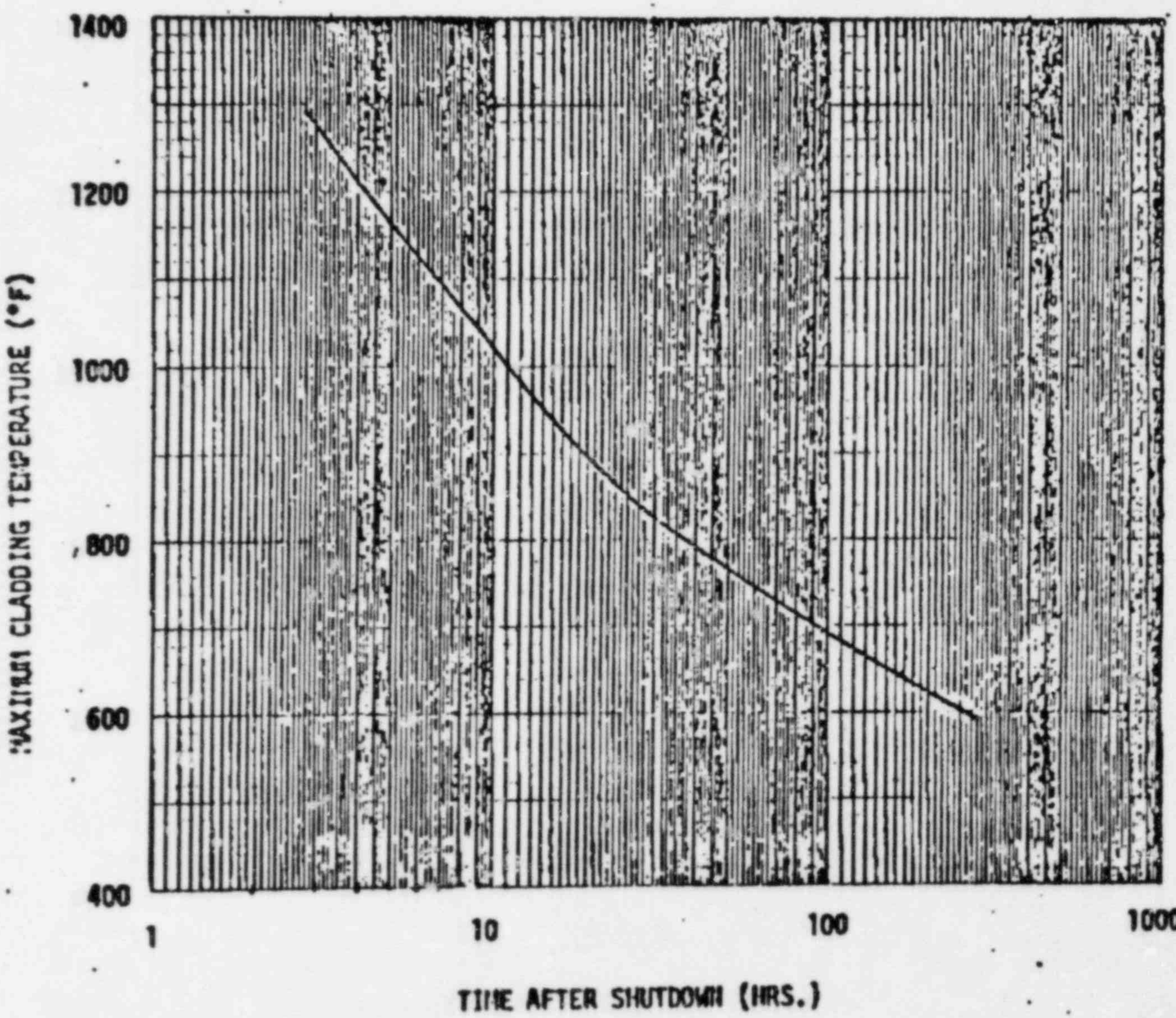

FIG. 1: VARIATION OF MAXIMUM CLADDING TEMPERATURE WITH TIME SINCE REACTOR SHUTDOWN FOR REFUELING ACCIDENT.

Revised PSAR Section 4.4

4.4.3.4.3 Secondary Control Assemblies

The steady state and transient T&H analyses for the Secondary Control Assemblies use similar analytical procedure as that for the Primary Control rod System. However, there are some differences in hydraulic design between the two systems. The Secondary Control Rod System utilizes hydraulic forces to assist scram action. To accomodate this unique feature a three-way flow split in the Secondary Control Assemblies is provided. The total coolant flow in a secondary control assembly first splits into upflow to the upper outlet plenum and downflow to the low pressure plenum. The upflow further splits into absorber pin bundle flow and bypass flow in the annulus between the pin bundle and guide tube. Discussions that follow are for each of the six SCA's.

Figure 4.4-65 illustrates the flow diagram for the thermal-hydraulic analysis of the SCA. The SCA thermal-hydraulic performance predictions begin with the determination of the control assembly flow rate, flow split, pre-scram hydraulic scram assist force, and pressure drops in the SCA compatible with the reactor pump head and flow rates in other reactor components. The computer program STALSS is developed specifically to provide this information in details. The DYNALSS code is used to predict the scram dynamics of the movable control rod from the fully withdrawn parked position. The control rod inserts into the reactor core by its own weight with the aid of hydraulic scram assist force at the beginning of the stroke. DYNALSS also provides steady state total flow rate, flow split, hydraulic scram assist force, and pressure drops in the SCA from pre-scram hydraulic calculation, but in a less detailed fashion compared to that computed by the STALSS code. The results from both codes are compared for verification purpose.

The pin bundle flow, by-pass flow, and down flow calculated by STALSS and physics design information are input to the CORTEM code. CORTEM consists of a unique module of three-way flow split including down flow in the SCA. The code treats steady state intra-assembly and inter-assembly heat transfer in a full 30-degree sector of the core. The intra-assembly heat transfer inside core assemblies is modeled based on application of the subchannel concept together with the use of bulk parameters for coolant velocity and coolant temperature within a subchannel. The inter-assembly transfer coefficient in the assembly gaps which is a function of interstitial flow and Peclet number of the collant. The result from CORTEM provides two important data for other thermal-hydraulic analyses: 1) SCA duct wall temperture distribution for ocre restraint analysis, 2) surrounding assemblies duct wall temperature distribution to be used as boundary conditions in detailed SCA pin bundle subchannel analysis.

Based on the SCA surrounding assemblies duct wall temperature distribution, the subchannel flow and temperature distribution is calculated by FULMIX which models the circular bundle of the SCA. The flow split uncertainties which are the major hot channel factors in the SCA are generated by STALSS and DYNALSS. Hot spot factors calculated by FATHON based on secondary pin pitch/diameter ratio are used to calculate absorber pin peak cladding temperatures due to wire wrap. The absorber pin temperatures and plenum pressure with and without uncertainties are calculated by CONROD.

Transient thermal-hydraulic performance predicitions for the SCA are performed with COBRA based on the reactor thermal-hydraulic transient data calculated by DEMO.

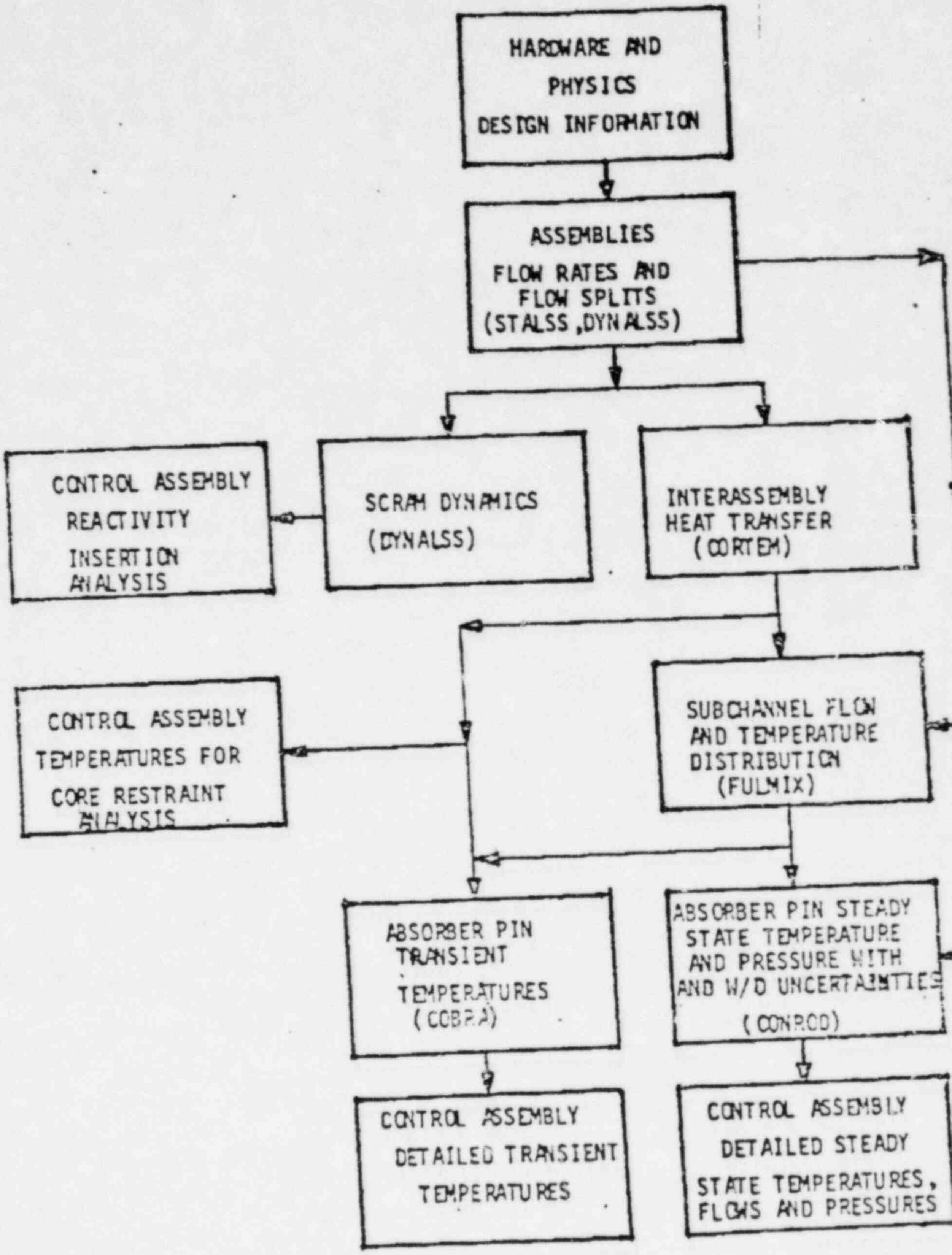

FIGURE 4.4-65 SECONDARY CONTROL ASSEMBLY AND ABSORBER PIN THERMAL-HYDRAULIC ANALYSIS FLOW DIAGRAM