YANKEE ATOMIC ELECTRIC COMPANY

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YANKEE ATOMIC BOILING WATER REACTOR ANALYSIS METHODS: ANALYSIS OF A TYPICAL BWR/4 TURBINE TRIP WITHOUT BYPASS TRANSIENT

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ABSTRACT

Simulation results obtained using Yankee Atomic Electric Company's BWR analysis methods are presented along with comparison to the results of other workers for a turbine trip without bypass transient. This work was requested by the United States Nuclear Regulatory Commission to aid in its review of Yankee Atomic Electric Company's BWR analysis methods.

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1.0 INTRODUCTION

An analysis of a turbine trip without bypass event is performed for the Peach Bottom Atomic Power Station, Unit 2. This analysis was requested by the United States Nuclear Regulatory Commission to aid in its review of Yankee Atomic Electric Company's BWR analysis methods.

The analysis employs the lattice physics, steady state physics, transient physics, and system transient methods described in References 1-4. The specific models used are described in Section 2. The primary results of the analysis are transient predictions of reactor neutron power and core pressure. These results along with comparisons to the results of other workers [5] are presented in Section 3. Conclusions regarding the analysis are given in Section 4.

2.0 METHODOLOGY EMPLOYED

2.1 Steady State Physics

For the transient analysis, the steady state calculations included: 1) modelling the Peach Bottom Unit 2 (PB2) core with SIMULATE [2], 2) depleting PB2 Cycles 1 and 2, and 3) simulating the initial conditions of the transient. This final step provided the input for the reactivity calculations and initial conditions for RETRAN.

Based on information from EPRI [6], the PB2 model was formulated. CASMO [1] was employed to calculate the two group cross sections for three bundle types. These bundle layouts, shown in Figure 2.1, were the most common fuel types in Peach Bottom during Cycles 1 and 2. The cross section data in table format were input to SIMULATE along with a quarter core Cycle 1 loading pattern. The model of Cycle 1 was then depleted to EOC using the Haling option and was shuffled into the quarter core loading pattern of Cycle 2. Finally, the Cycle 2 model was depleted to EOC with the Haling option. Both loading patterns are shown in Figure 2.2. The EOC2 calculations provided the transient physics base state case with its exposure distribution and void history. There were two basic inconsistencies between the plant operation and the SIMULATE model: 1) The plant did not have the quarter symmetric loading pattern as was specified in the model. 2) By using the Haling depletion option, the model assumed that all rods were out (ARO) at EOC and that each entire cycle ran at full power-full flow which was not the case for either cycle.

The initial conditions for the transient as described in Reference 5 were input to the SIMULATE Peach Bottom model (See Table 2.1). This case was restarted from the EOC2 case and the resulting core average axial power distribution, shown in Figure 2.3, was similar to that case's Haling shape. The goal was to provide an initial condition which was consistent with GE's and BNL's calculations. The only known measure for this consistency was the core average axial power distribution. Comparing the three power distributions, SIMULATE reactivity production was weaker in the bottom of the core. To adjust the SIMULATE model, a negative thermal absorber which varied axially was added to the core. Figure 2.4 shows the amount of absorber needed to obtain an axial power distribution similar to GE's and BNL's. The resultant average axial power distribution is compared to those of the other workers in Figure 2.5. This case was used as input for the reactivity calculations.

2.2 Transient Physics

Reactivity and kinetics data required as input to the RETRAN model were generated in accordance with the methods detailed in the Transient Core Physics Report [3]. The pretransient conditions and configuration were as detailed in Reference 5; the base state SIMULATE model at the pretransient conditions used in the generation of reactivity and kinetics data was created by the steady state physics analysis as presented in Section 2.1.

RETRAN data was specifically generated for a single 12 region and 12 volume active core channel model. Feedback reactivity data as described in Reference 3, consisted of volume fuel temperature, volume moderator density and volume relative moderator density reactivity functions. The procedure for the generation of these reactivity functions is detailed in Figures 2.1 and 2.2 of the given reference. These generated functions are analogous to those graphically presented in Figures 3.7, 3.10 and 3.11 of said report. The scram reactivity curve was generated by the reported procedure detailed in Figure 2.3 at base state conditions. This scram curve is provided as Figure 2.6 in this report. Kinetics parameters - effective delayed neutron fraction, precursor decay constants, and prompt neutron generation time - were calculated at pretransient conditions.

In order to characterize a core reactor state, core average reactivity coefficients are calculated. These coefficients are not intended for use in the transient analysis, but provide indices which may be used for comparative purposes. Table 2.2 provides the characterization of the Peach Bottom core at the pretransient conditions. In addition to the core average reactivity coefficients, the axial shape index, the effective delayed neutron fraction, and prompt neutron generation time characterize the core.

2.3 Core Wide Transient Analysis Model

The model used for the licensing transient simulation is essentially the same as the model used to perform the Peach Bottom turbine trip test simulations and is described in Section 3.1.1 of Reference 4. Two minor changes were made to the Peach Bottom model to make it even more consistent with the Vermont Yankee model [4]. These changes are the following: 1) the bypass system piping up to the valve chest was lumped into the steam line control volume upstream of the turbine stop valve and 2) the recirculation system junction inertias were recalculated in a manner consistent with the Vermont Yankee model. The setpoints and flow capacities of the safety relief and safety valves are based on the information provided in the Brookhaven report [5].

TABLE 2.1

PEACH BOTTOM UNIT 2 INITIAL CONDITIONS

Reactor Power (MWt)	3440.4	
Core Flow Rate (Mlb/hr)	101.0	
Core Pressure (psia)	1050.0	
Core Inlet Subcooling (Btu/lb)	28.9	
Core Average Exposure (MWd/ST)	12776.0	
Control Rod Inventory	0	[All Rods Out]

TABLE 2.2

PEACH BOTTOM 2
TRANSIENT PHYSICS PARAMETERS

Calculated Parameter	Value
Axial Shape Index$^{(1)}$	-0.1990
Moderator Density Coefficient (Pressurization), ¢/Δu$^{(2)}$ Pressure = 1055 psia Inlet Enthalpy - 525 Btu/lbm	23.5
Fuel Temperature Coefficient at 1100°F, ¢/°F	-0.28
Effective Delayed Neutron Fraction	.005375
Prompt Neutron Generation Time, microseconds	42.34

Notes: (1) Axial Shape Index (ASI) = $\frac{P_T - P_B}{P_T + P_B}$

(2) Δu = change in relative density (percent)

Bundle Type 1
With 80 mil channels

2
1 1
1 1 2
1 2 2 2
1 1 2 2 2
1 1 2 2 2 1
1 1 1 1 1 1 1

Fuel Type 1 - 1.33 w/o U_{235}

Fuel Type 2 - 0.71 w/o U_{235}

Bundle Type 2
With 80 mil channels

4
3 2
3 1 5
2 1 1 1
2 1 1 1 5
2 1 5 1 1 1
3 2 1 1 1 2 2

Fuel Type 1 - 2.93 w/o U_{235}
Fuel Type 2 - 1.94 w/o U_{235}
Fuel Type 3 - 1.6 w/o U_{235}
Fuel Type 4 - 1.33 w/o U_{235}
Fuel Type 5 - 2.93 w/o U_{235} with 3 w/o Gd_2O_3

Bundle Type 4
With 100 mil channels

4
3 2
2 1 1
2 5 1 1
2 1 1 1 0
2 1 1 1 1 1
2 1 5 1 1 1 5
3 2 1 1 1 1 1 2

Fuel Type 0 - Water Rod
Fuel Type 1 - 3.01 w/o U_{235}
Fuel Type 2 - 2.22 w/o U_{235}
Fuel Type 3 - 1.87 w/o U_{235}
Fuel Type 4 - 1.45 w/o U_{235}
Fuel Type 5 - 3.01 w/o U_{235} with 3 w/o Gd_2O_3

Figure 2.1

PEACH BOTTOM 2 BUNDLE TYPES

Loading Pattern - Cycle 1

```
2 2 1 1 2 2 1 1 2 2 1 1 2 1 2
2 2 1 1 2 2 1 1 2 2 1 1 2 1 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 1 1 2 2 1 1 2 2 1 1 2 1 2
2 2 1 1 2 2 1 1 2 2 1 1 2 1 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 2
2 2 1 1 2 2 1 1 2 2 1 1 2
2 2 1 1 2 2 1 1 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2 2
2 2 2 2 2 2 2 2 2 2
2 2 1 1 2 2 1 1 2 2
2 2 2 2 2 2 2 2
2 2 2 2 2 2 2
```

Loading Pattern - Cycle 2

```
2 2 4 2 2 2 4 2 2 2 4 2 2 4 2
2 2 2 2 2 2 2 2 2 2 2 2 2 4 2
4 2 4 2 4 2 4 2 4 2 4 2 4 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 4 2
2 2 4 2 4 2 4 2 4 2 4 2 4 2 2
2 2 2 2 2 2 2 2 2 2 2 2 2 4 2
4 2 4 2 4 2 4 2 4 2 4 2 4 2 2
2 2 2 2 2 2 2 2 2 2 2 4 2 2
2 2 4 2 4 2 4 2 4 2 4 2 2
2 2 2 2 2 2 2 2 2 4 2 2 2
4 2 4 2 4 2 4 2 4 2 2
2 2 2 2 2 2 2 4 2 2
2 2 4 2 4 2 4 2 2 2
4 4 2 4 2 4 2 2
2 2 2 2 2 2 2
```

Figure 2.2

PEACH BOTTOM 2 SIMULATE INPUT DATA

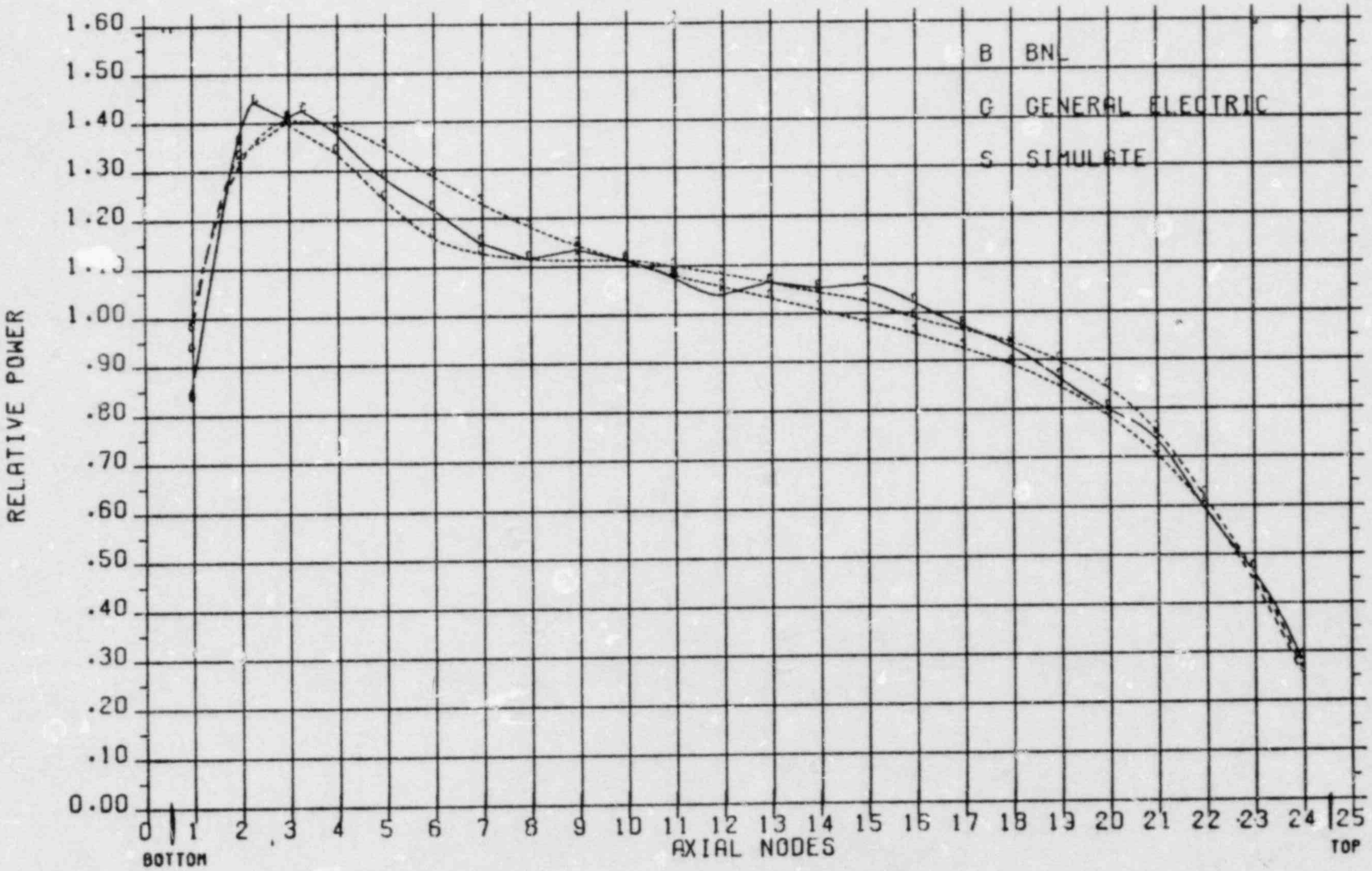


Figure 2.3

COMPARISON OF SIMULATE AXIAL POWER DISTRIBUTION (WITHOUT ADDED THERMAL ABSORBER) TO GE AND BNL DISTRIBUTIONS

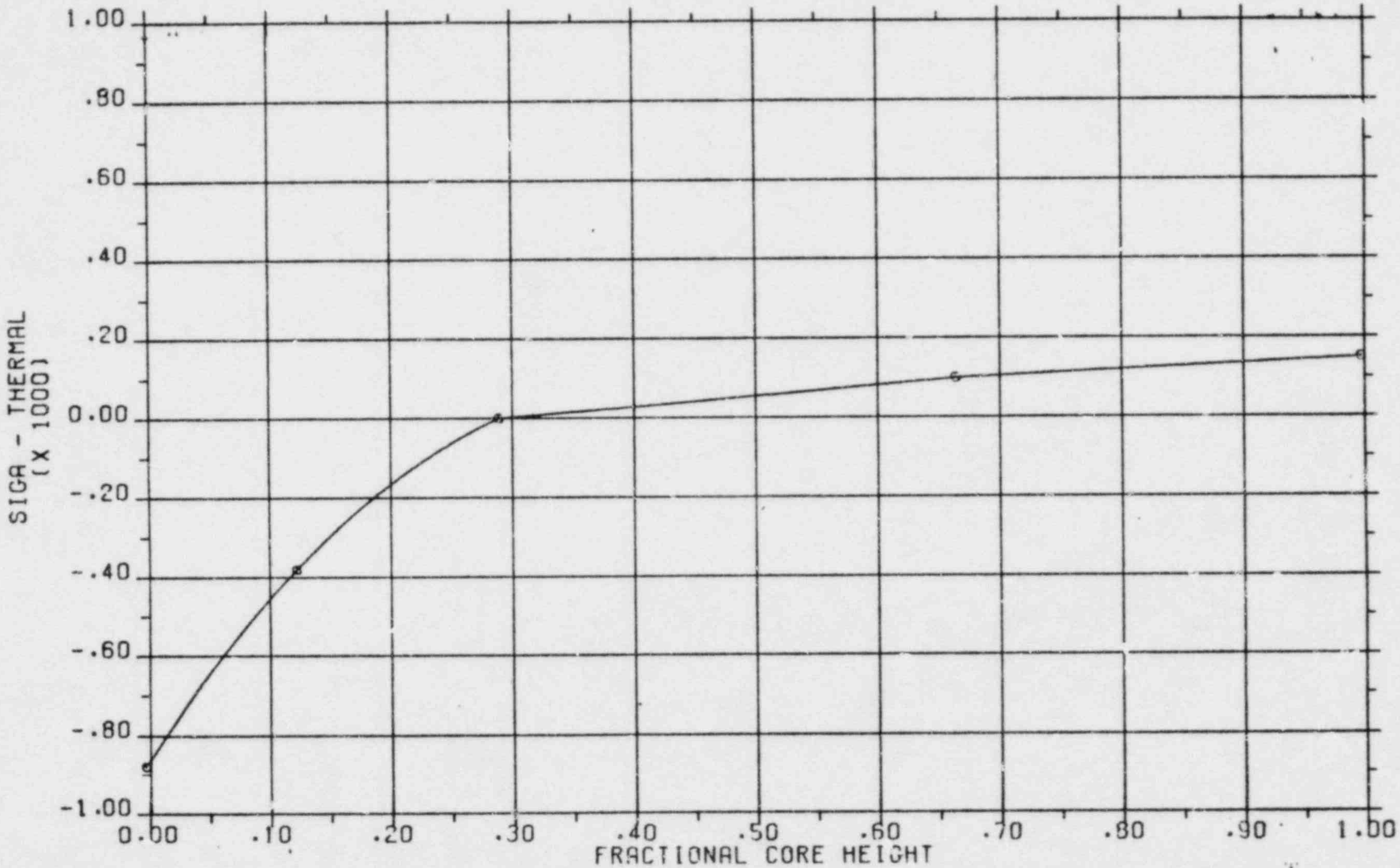


Figure 2.4

AXIAL DISTRIBUTION OF THERMAL ABSORBER ADDED TO SIMULATE

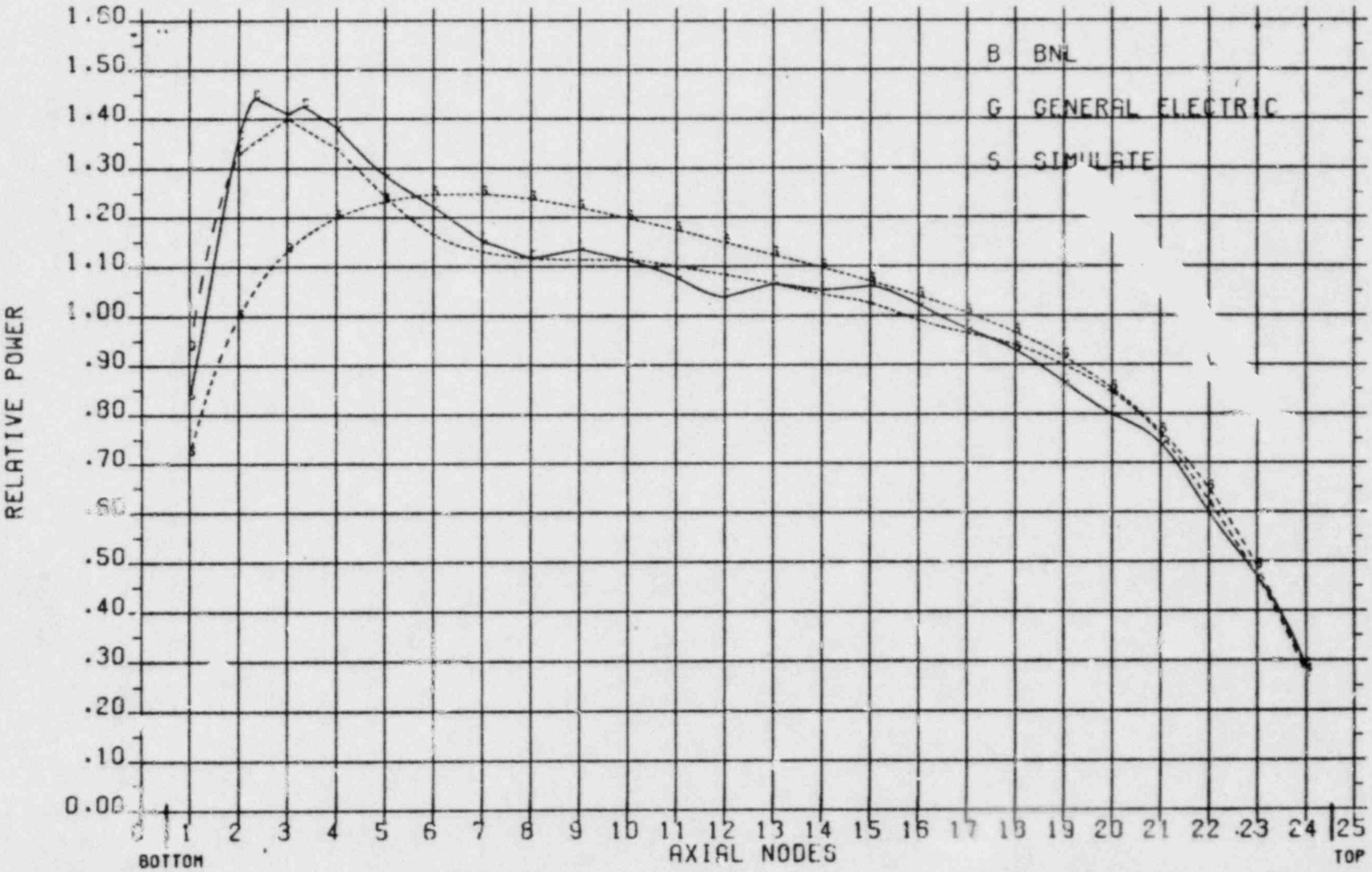


Figure 2.5

COMPARISON OF SIMULATE AXIAL POWER DISTRIBUTION (WITH ADDED THERMAL ABSORBER) TO GE AND BNL DISTRIBUTIONS

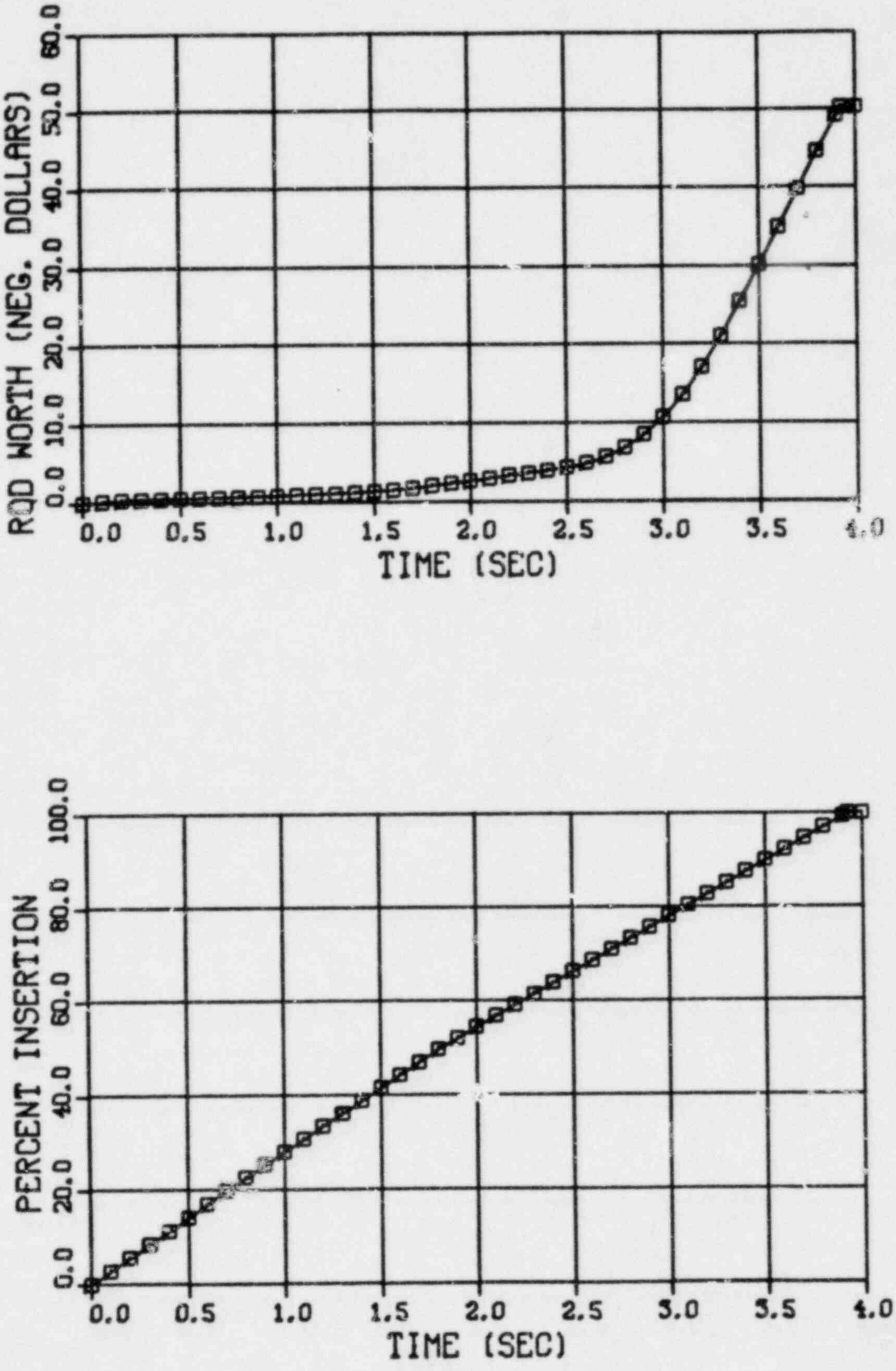


Figure 2.6

ROD WORTH VERSUS POSITION AND ROD POSITION VERSUS TIME AFTER INITIAL ROD MOVEMENT

3.0 ANALYSIS OF TURBINE TRIP WITHOUT BYPASS EVENT

3.1 Initial Conditions

In initializing the model, we tried to make the initial state as consistent as possible with the conditions described in the Brookhaven report [5] while still employing Yankee Atomic Electric Company methods. The core axial power distribution is based on the 3-D SIMULATE prediction (Section 2.1). The bypass flow is based on a FIBWR [7] prediction consistent with the SIMULATE power distribution. Core inlet enthalpy is set so that the amount of carryunder from the steam separators and the quality in the liquid region outside the separators is as close to zero as possible. This is done to maximize the initial pressurization rate. A summary of the initial operating state is provided in Table 3.1.

3.2 Analysis Results

The transient is initiated by a rapid closure (0.1 sec. closing time) of the turbine stop valves. It is assumed that the steam bypass valves, which normally open to relieve pressure, remain closed. A reactor protection system scram signal is generated by the turbine stop valve closure switches. Control rod drive motion is assumed to occur 0.27 seconds after the start of turbine stop valve motion. Predictions of the system parameters of main interest are shown in Figures 3.1 through 3.6.

3.3 Comparisons to Results of Other Workers

This section presents comparisons of predictions for the described transient between the system models of Yankee Atomic Electric Company (YAEC), General Electric (GE) and Brookhaven National Laboratory (BNL). These comparisons are made to aid the Nuclear Regulatory Commission in its evaluation of YAEC methods and do not constitute a critique of either worker's methodology. The GE and BNL results presented were obtained by manual scaling from figures in Reference 5. Comparisons of neutron power, core average heat flux, core pressure, and core inlet flow are presented in Figures 3.7 through 3.10. In general, the YAEC results indicate a more severe transient than the results of either GE or BNL.

TABLE 3.1

SUMMARY OF SYSTEM TRANSIENT MODEL INITIAL CONDITIONS

Core Thermal Power (MWth)	3441.2
Turbine Steam Flow (% NBR)	105.0
Total Core Flow (10^6 lbm/hr)	102.5
Core Bypass Flow (10^6 lbm/hr)	7.3
Core Inlet Enthalpy (Btu/lbm)	521.7
Steam Dome Pressure (psia)	1034.0
Turbine Inlet Pressure (psia)	983.3
Total Recirculation Flow 10^6 lbm/hr)	35.6
Core Plate Differential Pressure (psi)	18.7
Average Fuel Gap Conductance (Btu/hr-ft^2-F)	1000.0
Narrow Range Water Level (in)	29.0

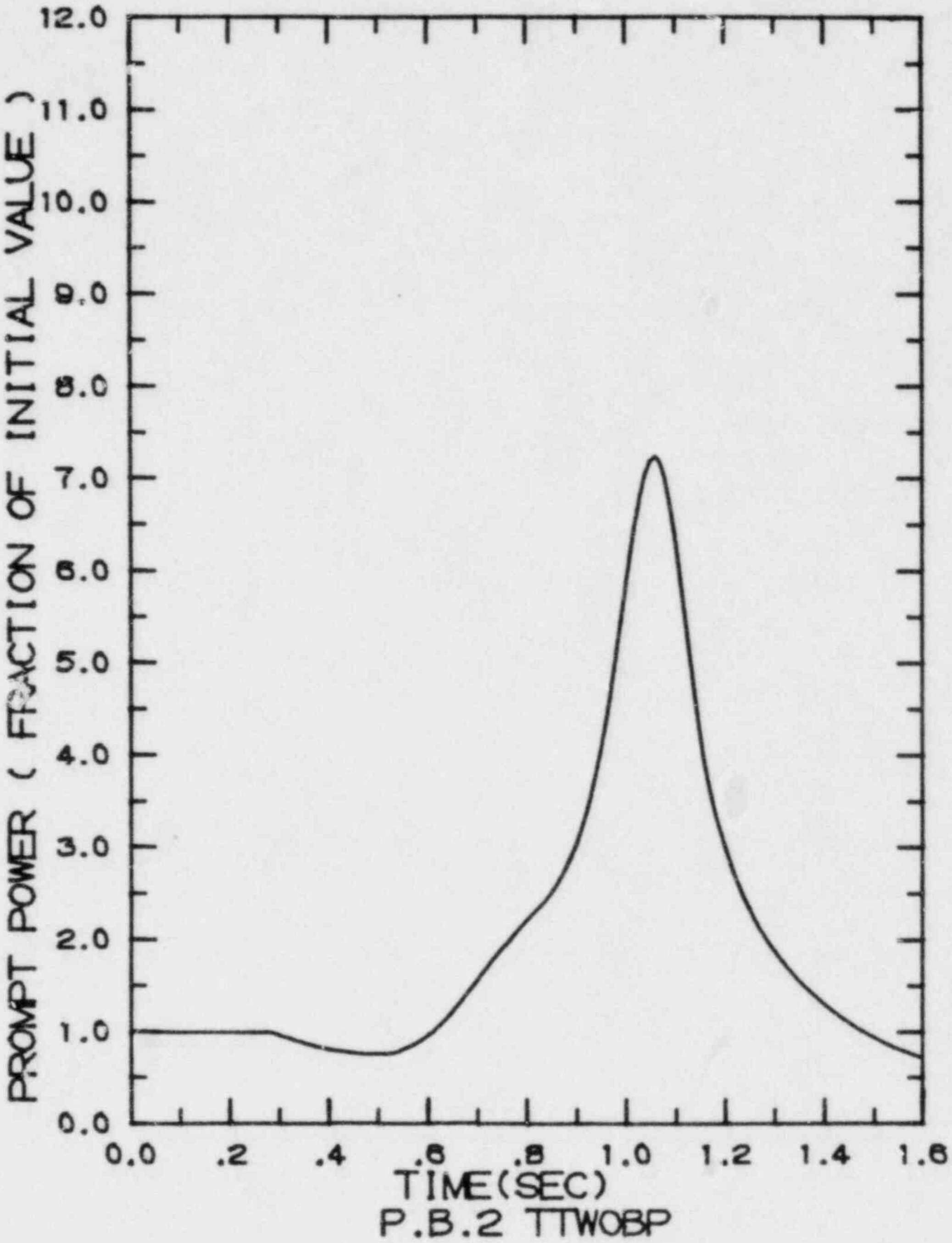
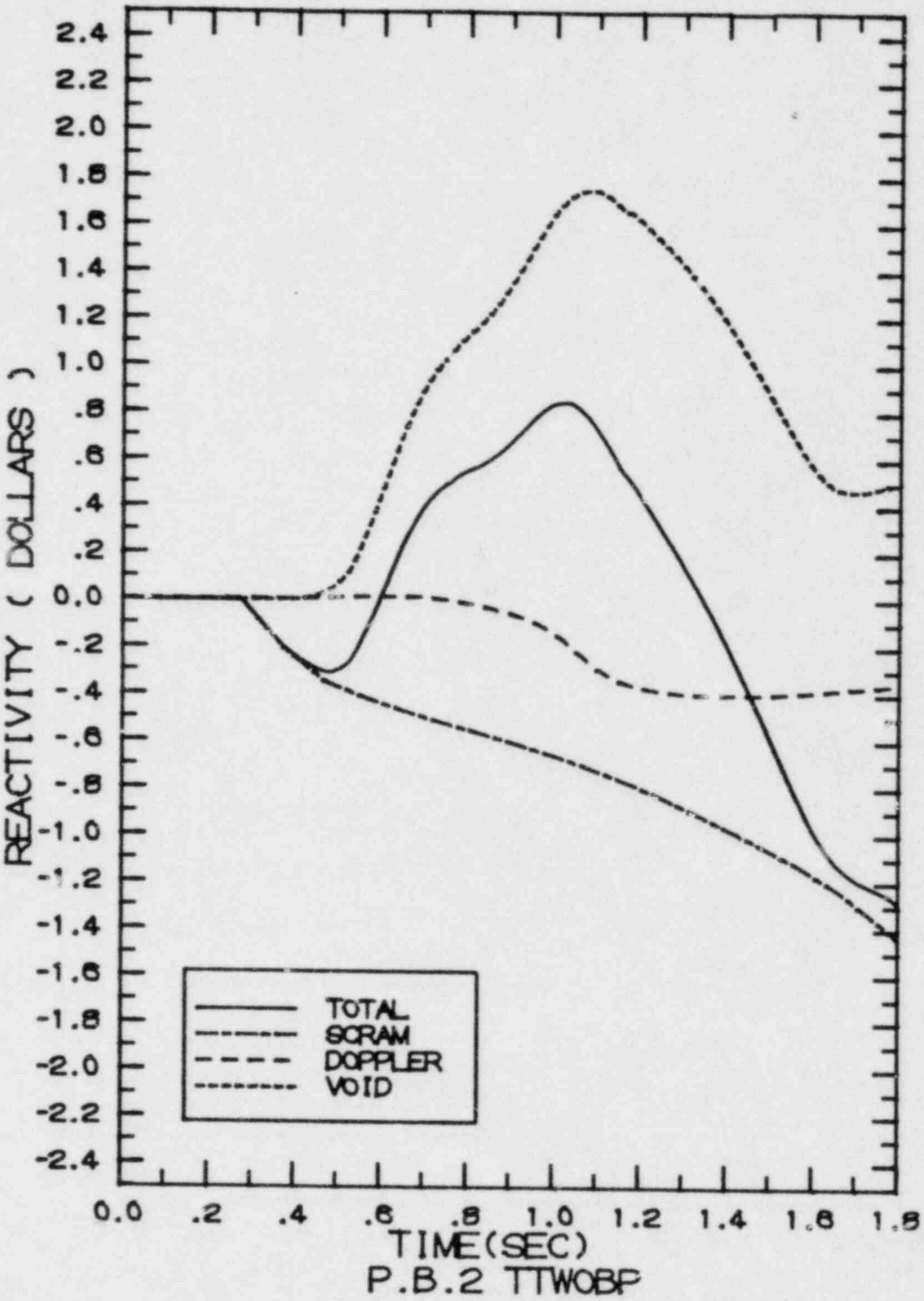


Figure 3.1

NEUTRON POWER PREDICTION



F'gure 3.2

TRANSIENT REACTIVITY COMPONENTS

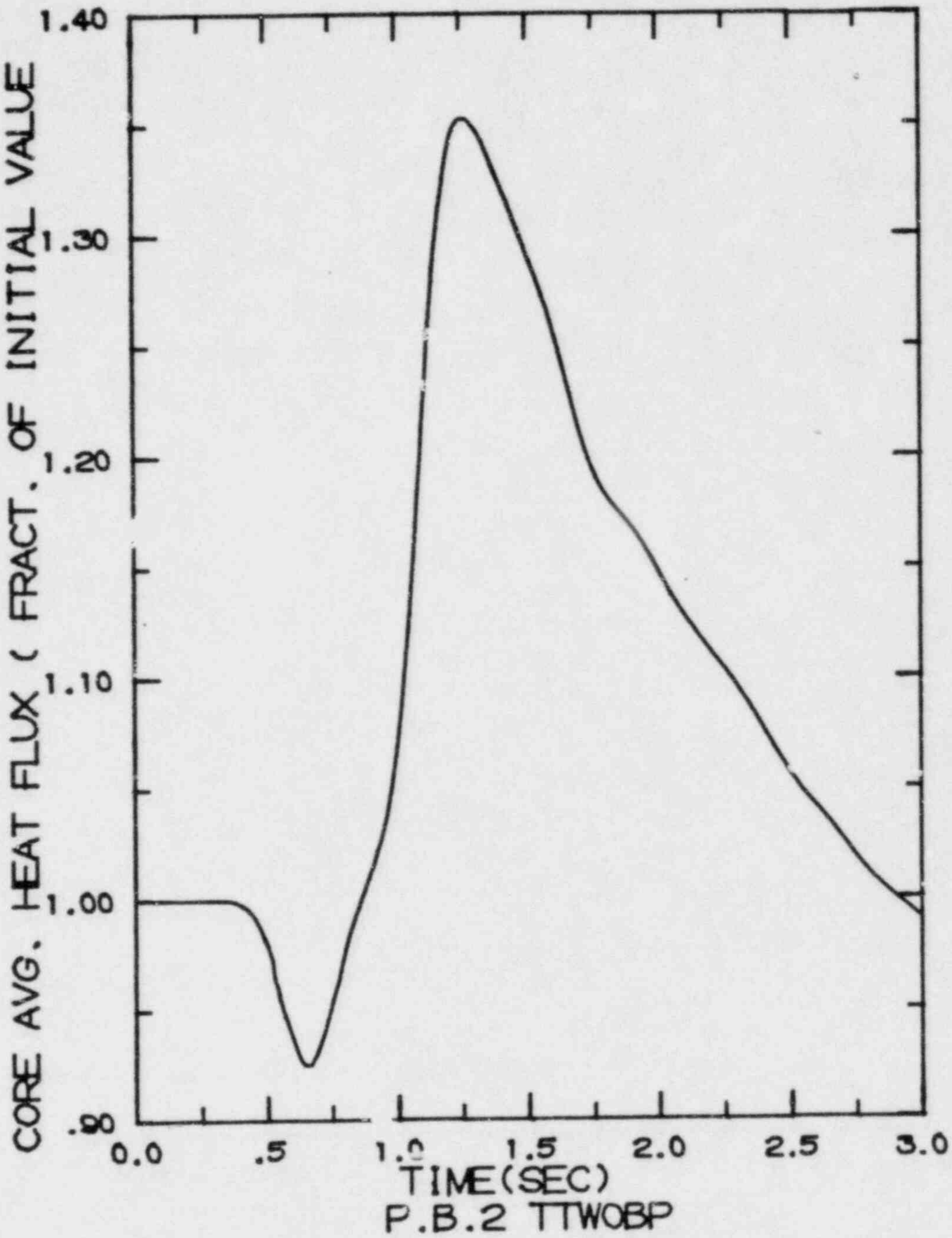


Figure 3.3

CORE AVERAGE HEAT FLUX PREDICTION

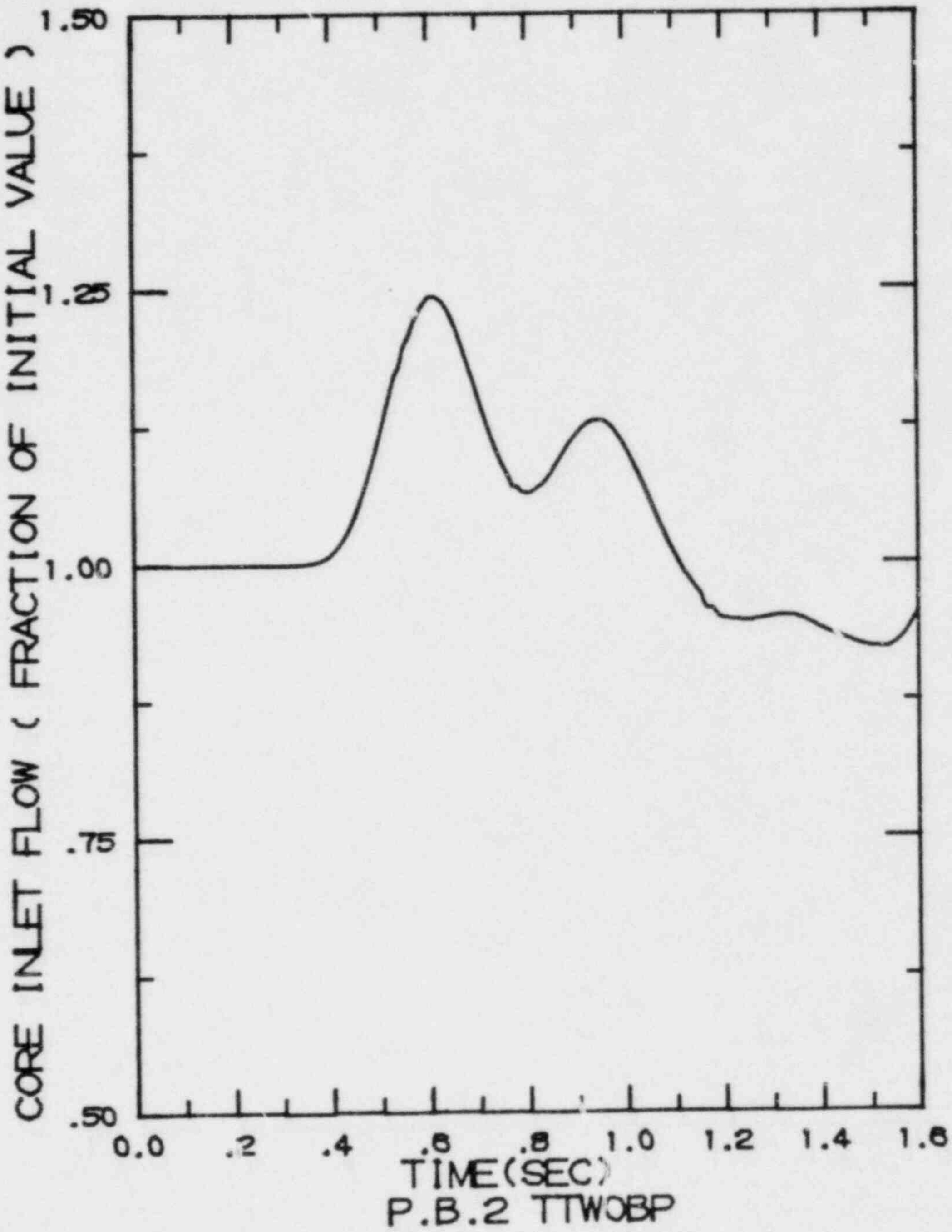


Figure 3.4

ACTIVE CORE INLET FLOW PREDICTION

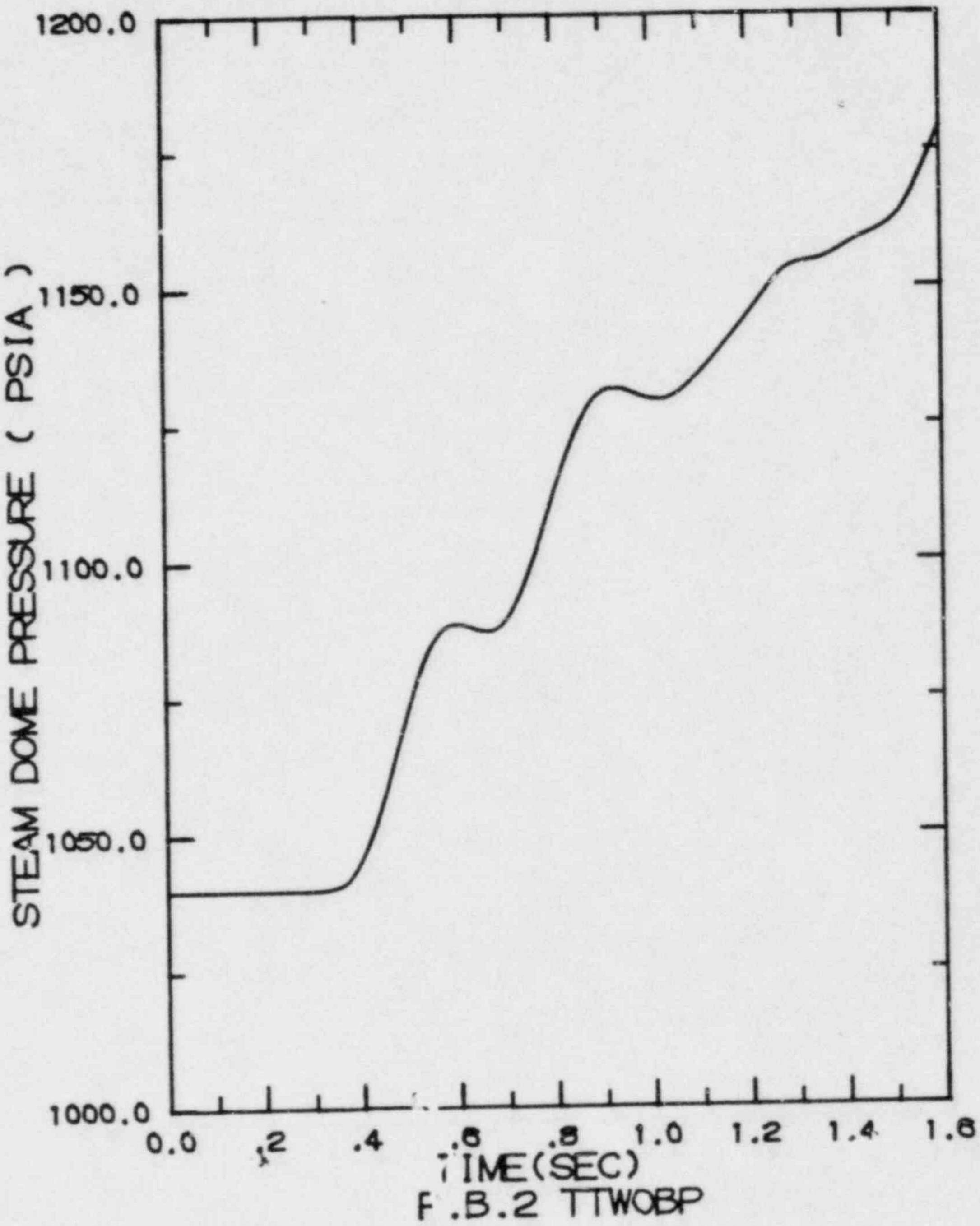


Figure 3.5

STEAM DOME PRESSURE PREDICTION

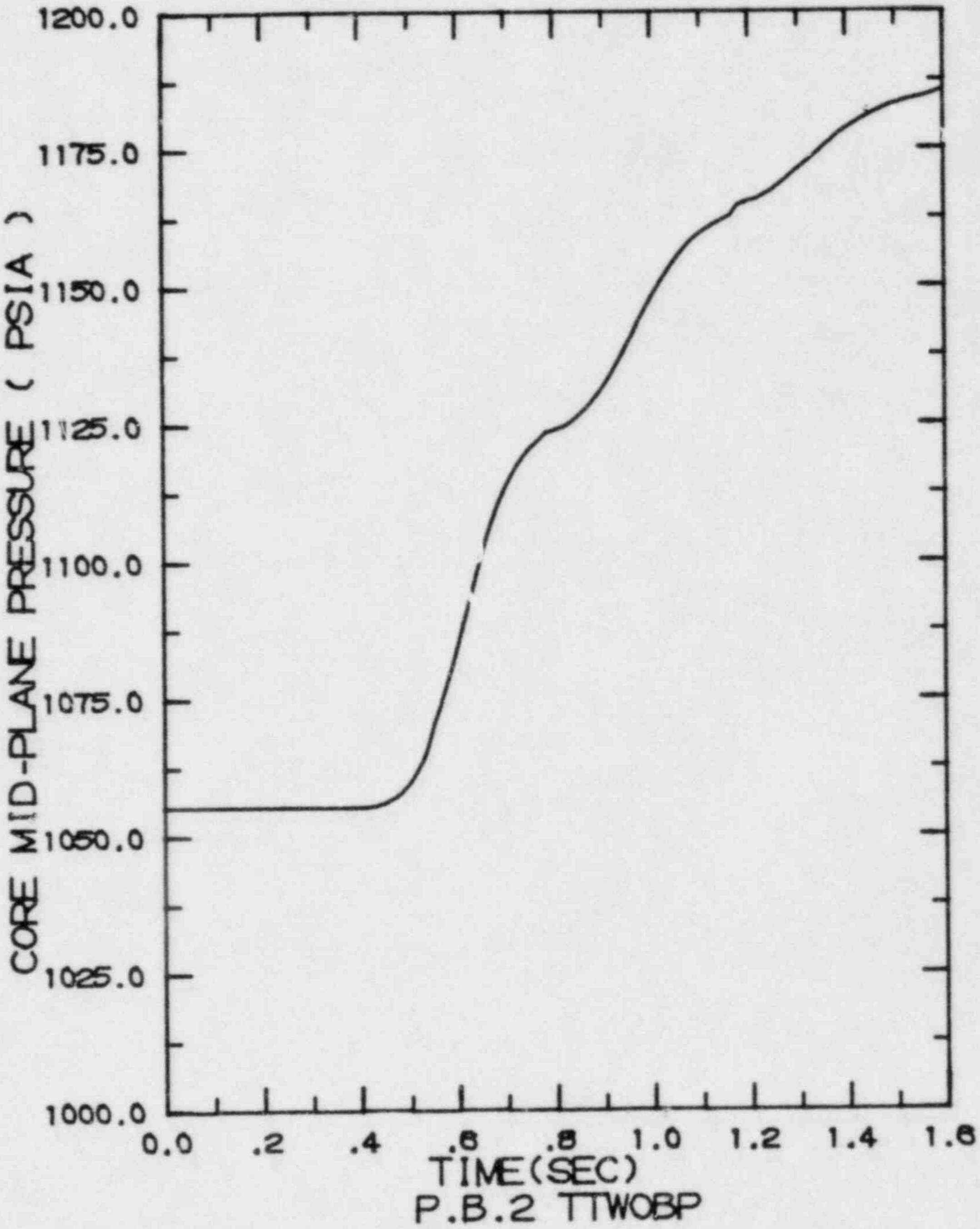


Figure 3.6

CORE MID-PLANE PRESSURE PREDICTION

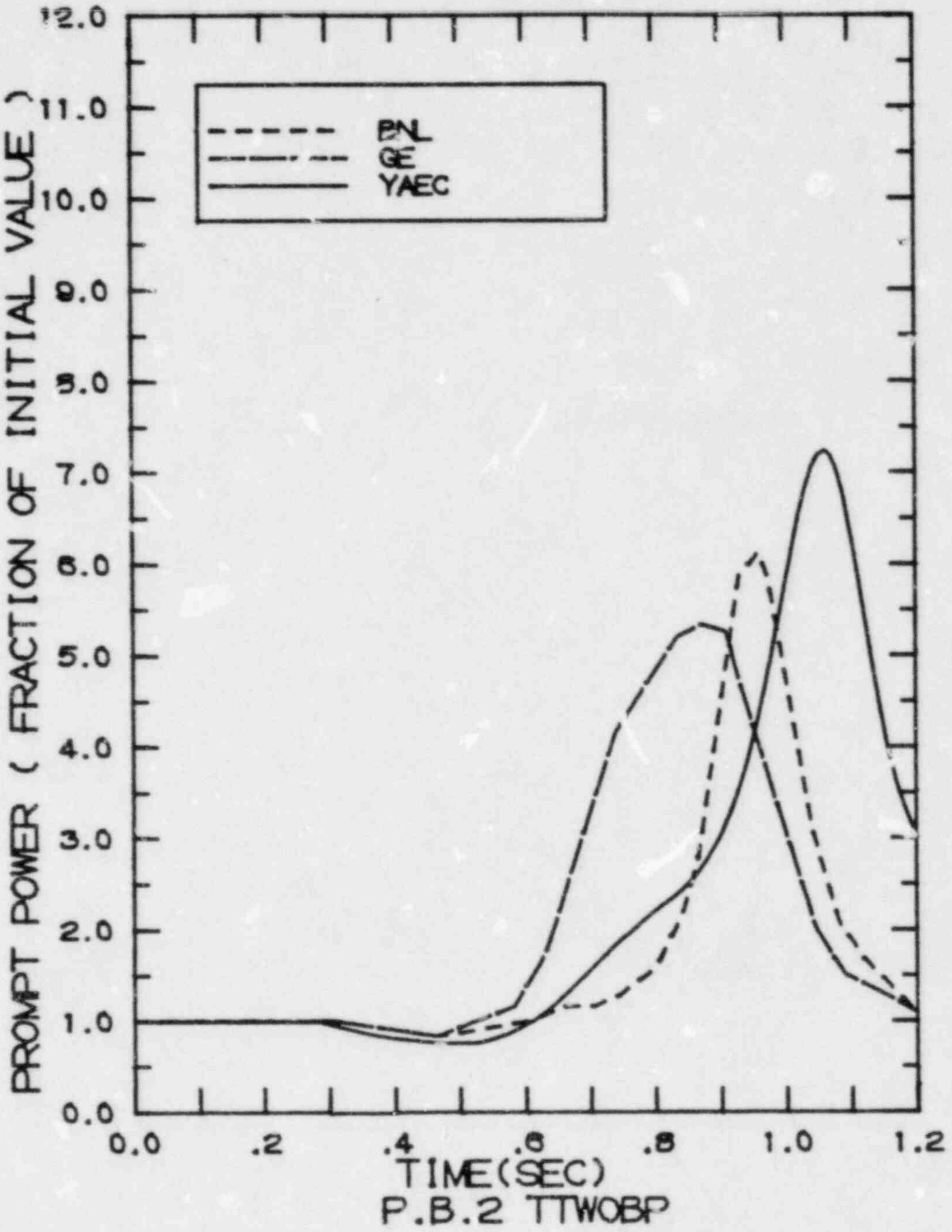


Figure 3.7

COMPARISON OF NEUTRON POWER PREDICTION TO GE AND BNL PREDICTIONS

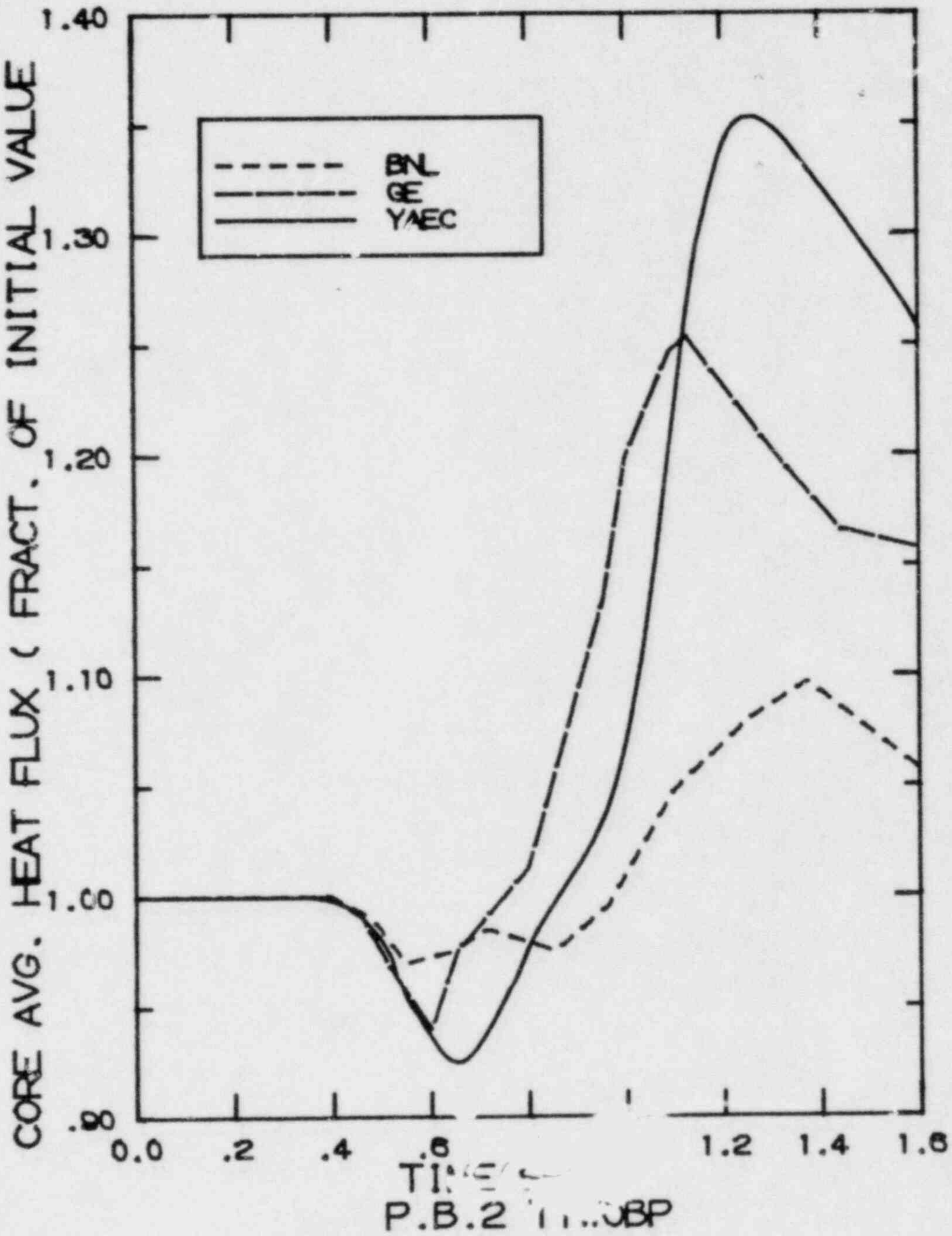


Figure 3.8

COMPARISON OF CORE AVERAGE HEAT FLUX PREDICTION TO GE AND BNL PREDICTIONS

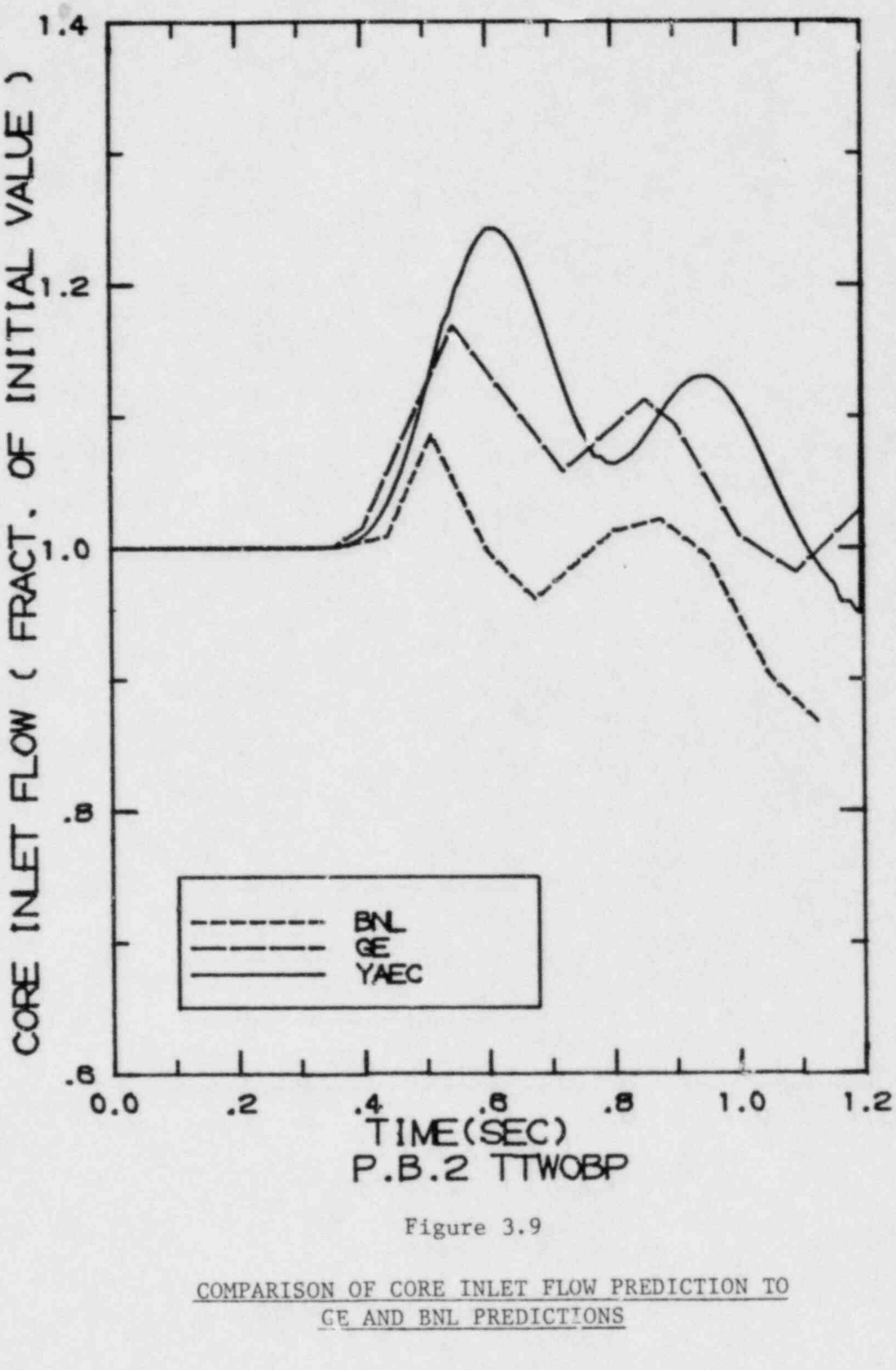


Figure 3.9

COMPARISON OF CORE INLET FLOW PREDICTION TO GE AND BNL PREDICTIONS

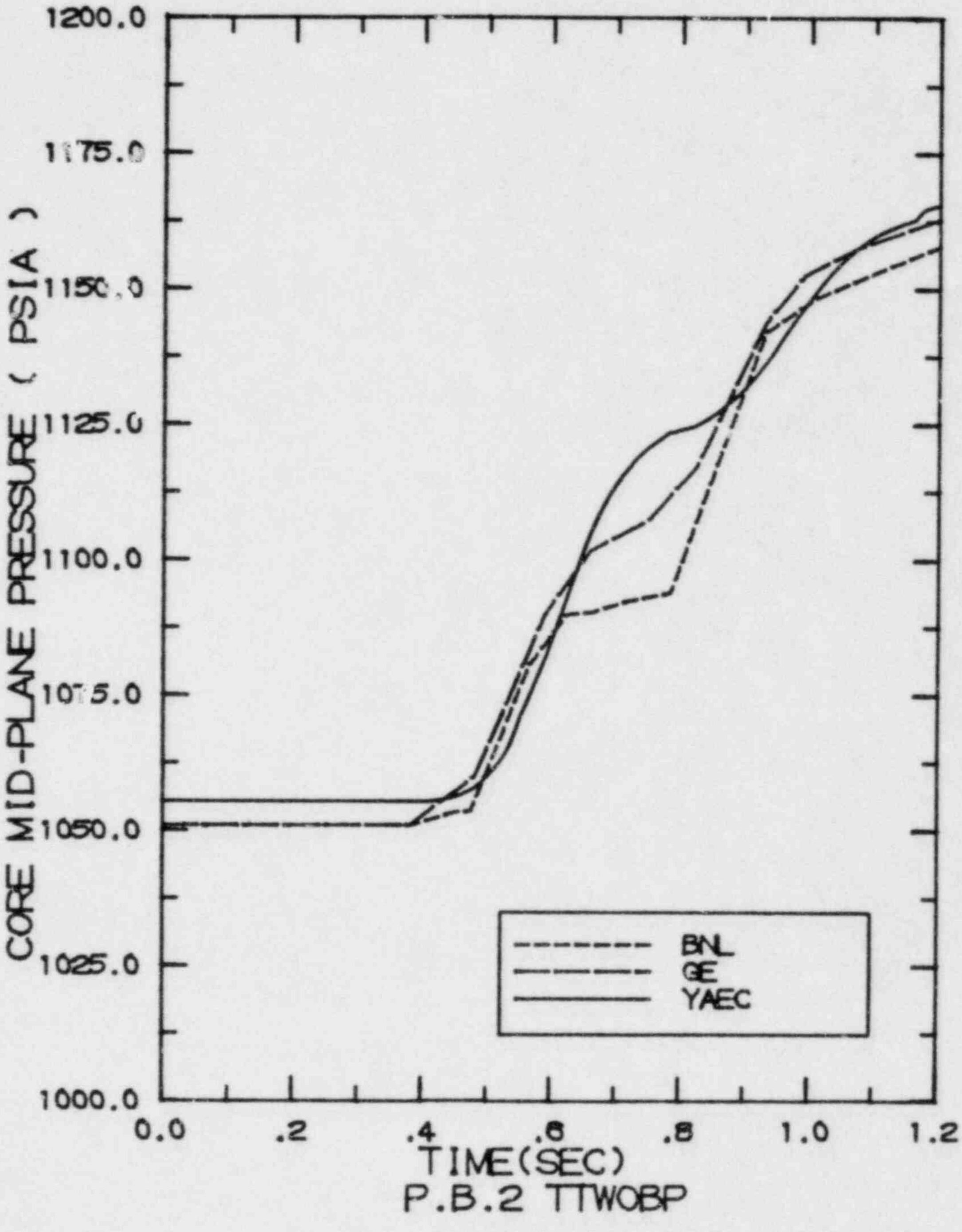


Figure 3.10

COMPARISON OF CORE MID-PLANE PRESSURE PREDICTION TO GE AND BNL PREDICTIONS

4.0 CONCLUSIONS

The methods used in analyzing the transient here are the same as those used in our recent analysis of the Vermont Yankee Nuclear Power Station [8] except that no artificial adjustments to the 3-D simulator input data were made in the Vermont Yankee analysis. Comparison to the results of other workers showed similar trends with the largest difference being in the neutron power prediction. Here, the YAEC simulation predicts a larger amount of energy release than the other two workers. This is evidenced in the YAEC simulation's prediction of the initial peak in core average heat flux, which is higher than that of the other workers. Not knowing all the input data used by the other workers, it is difficult to conclude the precise reasons for the differences in the predictions.

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