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SAFETY EVALUATION

FOR THE

GENERAL ELECTRIC TOPICAL REPORT

QUALIFICATION OF THE ONE-DIMENSIONAL

CORE TRANSIENT MODEL FOR

BOILING WATER REACTORS

NEDO-24154 and NEDE-24154-P

Volumes I, II ~~and III~~

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TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

I. SUMMARY OF TOPICAL REPORT

A. INTRODUCTION

Between April 9, 1977 and April 27, 1977, three turbine trip tests were performed at the Peach Bottom, Unit 2, to examine the validity of the General Electric transient analysis methods and verify the computer codes. The first scram signal which normally would have been initiated on the position of the turbine stop valve, was bypassed in order to provide a transient comparable in severity to the worst transients analyzed in FSARs. Using the transient analysis method and the REDY computer code used in the licensing applications at that time, General Electric made pre-test predictions of pressure, neutron flux and ΔCPR on a best estimate basis. The neutron flux and ΔCPR predictions were significantly nonconservative and the pressure predictions were somewhat nonconservative.

After the tests General Electric performed post-test predictions of pressure, neutron flux and ΔCPR using the actual or measured plant parameters with best estimate modeling assumptions as well as the licensing model assumptions. The ΔCPR and the neutron flux predictions were again nonconservative for both sets of calculations. The pressure peaks were predicted conservatively. It should be noted that General Electric showed that the predictions of pressure and ΔCPR were conservative with licensing basis inputs when the first scram signal initiated on turbine stop valve position was not bypassed, i.e., under normal conditions.

The comparisons of the test results and the REDY code, the licensing basis model, confirmed the existence of a steam line pressure wave propagation phenomenon in a turbine trip transient and time varying nature of the axial core power distribution. General Electric accelerated its model development program to include steam line dynamics and representation of the core physics and thermal hydraulics in space-time domain and produced the ODYN computer code which is the subject of this review.

B. SCOPE

The scope of this review is the evaluation of the ODYN code for use in the analysis of certain transients in Chapter 15 of the FSARs.

C. SUMMARY OF ANALYTICAL MODELS

The overall system model in the ODYN code consists of a one-dimensional representation of the reactor core, and the recirculation and control system model. These two models are coupled to each other. A steady state initialization is made initially, and then the parameters for the transient are calculated.

First, the recirculation and control systems are solved for the steady state conditions. Some of the initial conditions are input and they may be plant unique. Other initial hydraulic values such as core pressure drop and bypass flow fraction, which are also input to the steady state recirculation and control model, are calculated elsewhere. These parameters are calculated in the steady state multi-channel core code (Reference 1). Using all these input values, the steady state recirculation and control model calculates the remaining hydraulic parameters in the plant. The steady state initialization in the recirculation and control model provides the loop pressure drop, core exit pressure, core inlet flow and enthalpy to the one-dimensional reactor core model. These values are used in the reactor core model to calculate the neutron kinetics, thermal hydraulics and fuel parameters for the steady state conditions.

The steady state axial power distribution is calculated by the neutronics model. The model uses cross section fits obtained from an analysis about cross sections for different relative coolant densities and control states and that are radially averaged for each axial plane. The fits are such that the axial power in the one dimensional model is required to yield the same axial behavior as in the three-dimensional BWR Core Simulator solution. The steady state thermal hydraulic solution permits the calculation of the steady state fuel temperature distribution.

During the transient, the recirculation and control system model calculates the time derivatives. At the end of the time step, the recirculation and control system model supplies the new external boundary conditions to the reactor core model. The reactor core model calculates the new neutron flux, thermal hydraulic parameters and fuel temperatures. It also provides reactor core exit quality, flow and pressure as input to the recirculation and control system model. The recirculation and control system model calculates the loop pressure drop and the reactor core model calculates the core pressure drop. These pressure drops are compared. If they are not equal within a certain limit, the recirculation and control system model derivatives are modified and the time step calculations are repeated.

The recirculation system is modeled by solving the mass, energy and momentum conservation equations for the steam line, reactor vessel and recirculation loop components which included jet pumps, recirculation pumps and associated piping. The control system is modeled as a series of connected gains, filters, integrators, and nonlinearities (limiters and function generators). The control system output is valve position and thus flow control. The one-dimensional core model comprises equations describing the neutron kinetics, thermal-hydraulics and heat transfer behavior of the core.

Major assumptions used in the modeling of the recirculation system are as follows:

1. Pressure variations in the system are described with ten nodes. One node is used for the reactor inlet; another node is used for the reactor vessel dome, and the remaining eight nodes are used to describe the behavior of the steam line.

2. Liquid and vapor mass volume balances are used to predict the reactor vessel water level changes.

3. The recirculation loop model can simulate any combination of multi-loop systems. The entire recirculation loop is assumed to be subcooled and incompressible.

4. Steam in the steam line is treated as single phase flow. Condensation of steam in the steam line is precluded during the transient.

Major assumptions used in the reactor core model are as follows:

1. A one-dimensional neutron kinetics model is assumed. The neutron flux varies axially with time. One energy group diffusion theory and six delayed

neutron groups are used. Decay heat is modeled using a simple exponential decay heat model. The one dimensional neutron diffusion parameters are obtained by collapsing the parameters obtained from the GE three-dimensional BWR Core Simulator (Reference 2).

2. A single active heated channel represents the core average conditions and another single channel represents the core bypass. A five equation model representing mass and energy conservation for the liquid and vapor, and the mixture momemtum conservation are used to calculate core thermal-hydraulic behavior.

3. Heat transfer to the moderator and fuel temperatures are calculated using an average fuel and cladding model at each axial location of the core. The gap conductance is an input parameter which may vary axially in time. The conduction parameters are temperature dependent. A radially uniform (flat) power distribution is assumed in the fuel rods.

II. STAFF EVALUATION

The staff evaluation was performed in three parts:

A. Review of the analytical models in the ODYN code and determination of uncertainities in the code modeling.

B. Review of the qualification of the code. This part of the review is accomplished in three areas:

   1. Comparison of specific models in the code with separate effects test data.

   2. Comparison of integral response of the code with the integral test data.

   3. Comparison of the code predictions with the predictions of an independent code; i.e., audit calculations.

C. Review of the safety margin; i.e., evaluation of the margin when the code is used with the uncertainties assigned in the licensing basis transient. The uncertainties of the calculations were evaluated as part of the calculational model review.

The measure of all code uncertainties is made in terms of ΔCPR/ICPR ratio. The "CPR" is an acronym for critical power ratio. It is the ratio of the critical power of the limiting bundle in the core to the power of the same bundle at the operating power of interest. The critical power is an artificial bundle power obtained by increasing the power analytically until the critical quality is reached. The analysis is performed using the GEXL correlation. Since the hydraulic and neutronic parameters change during the transient, CPR also changes during the transient. The minimum value of the CPR is called MCPR and the difference between the initial critical power ratio, ICPR, and MCPR is the ΔCPR. Hence, the ratio of ΔCPR/ICPR is a measure of the relative severity of the transient.

The uncertainties in the code are determined by making sensitivity studies. An independent parameter in the code is perturbed and the resulting change in ΔCPR/ICPR is calculated for a turbine trip without bypass transient, which is generally limiting. These independent parameters pertain to the various models such as the parameter of C_o in the Zuber drift flux model or frictional loss coefficients in the steamline. They do not pertain to system parameters which determine the actuation of the valves since licensing basis analysis require limiting settings for these systems parameters.

A. REVIEW OF ANALYTICAL MODELS

1. <u>Recirculation and Control System</u>

a. <u>Recirculation Loop Model</u>

The recirculation loop system consists of the upper plenum, steam separators, vessel dome, jet pump and recirculation loop. Mass, energy and momentum conservation equations are used to describe thermal and hydraulic behavior of the components. These equations are solved using an explicit finite differencing method which is presented in Reference 3.

During the steady-state initialization, the time derivatives are set equal to zero. A multi-channel steady state hydraulics code provides the steady state core pressure drop and the bypass flow fraction to the recirculation system model. This code is presented in

Reference 1 and has been reviewed and approved by NRC (Reference 4). The other inputs used by the recirculation system model are plant specific such as dimensions related to plant geometry, pressure loss coefficients, separator carryunder fraction and jet pump and recirculation pump characteristics.

In the initial steady state conditions the jet pump drive and suction flows can be determined from the equation of continuity and the jet pump "m" ratio. This ratio is defined as the ratio of the suction to the drive flow. It is valid for the rated conditions which are selected to correspond to steady state initial operating conditions. Using the momentum equation and the "m" ratio, the suction flow and the suction flow loss coefficient are determined. During the transient the ratio changes. The jet pump suction and drive flows (consequently recirculation loop and core inlet flows) are calculated using the momentum equations keeping the suction flow loss coefficient constant. The sum of the suction and drive flows provide the recirculation loop flow and the sum of all recirculation loop flows provide the core inlet flow.

The recirculation system models used in the ODYN and REDY codes are the same. The REDY code (Reference 5) has been reviewed for ATWS analyses and the recirculation system model has been found acceptable with some limitations (Reference 6). The following discusses and evaluates the recirculation system model. This evaluation, except for the uncertainties, is the same both for ODYN and REDY codes. The limitations found in the RED'.' code are equally applicable in the ODYN code.

During the transient, momentum equations are used to calculate the jet pump suction and drive flows. Hence, the form loss coefficients in the recirculation system affect the core flow and consequently the calculated ΔCPR. A sensitivity study performed by General Electric using the ODYN code by decreasing the diffuser form loss coefficient by 10% showed an increase of 0.001 in ΔCPR/ICPR. General Electric estimated uncertainties in the jet pump loss coefficients about 20%. These uncertainties are inferred from the uncertainity in the jet pump "m" ratio. General Electric noted that the decrease in the jet pump pressure drop loss on the order of 20% changed ΔCPR/ICPR by 0.01. This is the biggest uncertainty estimated by General Electric in the recirculation system. According to General Electric reasonable variations in other parameters such as drive flow L/A, jet pump areas or lengths (which are manufactured to close engineering tolerances) and loss coefficients at the nozzle, plenum and bulkwater did not change ΔCPR/ICPR ratios significantly. Based on these sensitivity studies the impact of these uncertainties on the values of ΔCPR/ICPR in the generally limiting transient is small.

During the transient, the transient terms of the momentum equation representing inertia may become important in determining the core flow. Recirculation pump trip tests were performed at 50%, 75%, and 100% power levels in the Oyster Creek plant and reported in Reference 5. Good agreement exists between the measured and REDY calculated core flows for the transient. This shows that the momentum equations were solved correctly to predict flow transients. Recirculation pump trip tests were also performed in Dresden-2 and they were reported in Reference 5. However, in these tests measured core flows were higher than those calculated because the actual pump inertia was higher than the value used in the analysis.

One of the jet pump modeling assumptions is that the region from the nozzle to the throat is considered to have no inertia. In order to validate the transient modeling of the jet pump, transient jet pump tests were conducted at the Moss Landing Generating facility, Reference 5. In these tests the jet pump drive flows were oscillated at several frequencies and measurements were made of the gain and phase relationship of the drive flow. Comparison of the measurements and model predictions showed good agreement up to 5 Hz. The model did not predict a resonance condition in the cold test data at 6.5 Hz; consequently, the use of the model is limited to 5 Hz. This limitation means that the code will have errors if recirculation loop flow variations are sudden. The harmonic components of the flow variation should be less than 5 Hz.

Another assumption which has been validated by tests is the assumption of complete mixing at the core inlet. Tests were performed in Monticello to verify this assumption. Core flow distributions for three core flow rates, at 29%, 50% and 85% of rated flow rates, were measured for symmetric operation of the recirculation pumps, Reference 7. Tests results indicate that the bundle flow rate does not vary more than 2.8% from that in the average bundle with 95% confidence level. This indicates that the assumption of uniform pressure distribution at the inlet of the core and complete mixing is a valid assumption for the recirculation system modeling.

The review of the analytical models and the comparison of the predictions with the tests above indicate that the recirculation loop model and the impact of associated uncertainties on ΔCPR/ICPR as presented by General Electric are acceptable. The harmonic components of the flow variation should be less than 5 Hz and the model should be valid for the analysis of transients where the fluid in the recirculation loop, downcomer and core inlet remains subcooled (incompressible). In the transients to be analyzed by the ODYN code, it is expected that these limits will not be exceeded.

b. Control System Model

The control system models were evaluated for structure as well as the methodology for evaluating plant specific properties. Plant specific properties consist of response functions, gains, and time constants for the control system.

The system models are composed of transfer functions, limiters and function generators. The transfer functions are based on typical filters and proportional, integral, derivative control laws. Limiters and function generators are used in the modeling of flow valves as a means of linearizing the gain within control loops. We have reviewed the model structure for the motor generator flow control model, the feedwater control model and the pressure regulation with the Mechanical Hydraulic Control. We find the structure of these models acceptable and typical of the type of modeling conducted with classical control system theory.

With respect to the description of the control models, the following models were evaluated:

(a) Valve Flow Control System

(b) Motor-Generator Flow Control

(c) Feedwater Flow

(d) Pressure Regulator and Turbine Controls

(e) Reactor Safety Systems

For input signals, the Valve Flow Control model receives a turbine governor signal, a sensed steamflow signal, a filtered neutron flux signal, a recirculation drive flow signal and a manual setpoint signal. The control system is modeled as a series of connected gains, filters, integrators, and nonlinearities (limiters and function generators). The control system output is valve position and thus flow control.

For input signals, the Motor-Generator Flow Control model receives a load demand error, a master manual or automatic signal as well as a loop manual or automatic signal. The control system is modeled as a series of gains, integrators, function generators, and with actuators of a drive motor, variable speed coupler, generator, and motor pump. The controlled variable is recirculation drive flow.

For input signals, the Feedwater Control System receives feedwater flow disturbances, vessel pressure corrections, a level setpoint signal, a mixture level signal, and a steam flow signal. These signals are operated on by a control modeled as a series of connected gains, integrators, filters, and non-linearities (limiters and function generators). The controlled variable is feedwater flow.

For input signals, the Pressure Regulator receives a turbine inlet pressure signal, a pressure setpoint, a turbine speed setpoint and a turbine load setpoint. These signals are operated on by a control modeled as a series of gains, filters, control laws, control valve servos and non-linearities. The controlled variable is turbine inlet pressure.

The staff review finds that these models are conditionally acceptable. Technicallly, the models are composed of transfer function, gains, filters, and synthesized nonlinearities such as deadbands and saturation limits. The technical form of the control system models is acceptable to the staff.

However, the model is used to establish initial control system settings such as gains, time constants, and control functions. Since the selection of these settings is made on a plant specific basis, the staff requires that each applicant's Safety Analysis Report reference a clearly defined basis for making these selections. The design criteria must be provided for each control system of the plant. The initial control system characteristics shall be verified as conforming to the design criteria for each control system of the plant.

c. <u>Steam Separator Model</u>

The separator is modeled using a one dimensional momentum conservation equation whereas the flow in a separator is rotational and clearly multi-dimensional. However, using separator test results (Reference 8), it was possible for General Electric to develop an empirical one dimensional momentum equation describing the flow behavior. Tests indicated that the thickness and configuration of the layer of swirling water along separator walls is independent of the inlet flow (for 200,000 lb/hr < Flow < 800,000 lb/hr) but dependent on the inlet quality. The water layer primarily affects the effective L/A in the momentum equation of the separator. Due to differences between the densities of steam and water, the primary inertial effects are due to the liquid. The tests of Reference 8 provided a relationship between the effective L/A and the inlet quality, and an empirical separator pressure drop coefficient.

General Electric states that the value of pressure drop coefficient has a conservative bias in it. The higher the pressure drop or the pressure drop coefficient, the higher is the value of ΔCPR/ICPR. However, General Electric did not quantify the conservatism in this model in terms of ΔCPR/ICPR relative to actual plant conditions. Therefore, no credit is given to this conservatism.

General Electric performed sensitivity studies decreasing the value of L/A by 30%. This resulted in an increase of 0.002 in ΔCPR/ICPR. In order to assess if the scatter of 30% in the separator L/A is sufficient, the staff reviewed the separator data in Reference 8.

The data indicates that the scatter in the separator L/A values can be high. The thickness of water layer can be used to make a fairly good estimate for L/A. Tests indicate that the thicknesses of water layer for the same conditions can vary from each other by a factor of four. Reference 8 describes the reason for these variations as an instability.

Discussions with General Electric indicate that the value of L/A used in the ODYN code included the value of L/A for the standpipe and therefore, the scatter was not a factor of 4 but it was judged to be 30%. The staff has no information how these separate L/A effects (one due to the separator and the other due to the standpipe) can be assessed.

Reviewing the analytical model we find that the separator model is acceptable; however, based on available information we judge that a factor of 2 in separator L/A variation (rather than 30%) would be more appropriate in assessing the uncertainty. Hence, we estimate the component of that ΔCPR/ICPR uncertainty for L/A will increase from ± 0.002 to ± 0.015.

d. Upper Plenum, Vessel Dome and Bulkwater Model

These components are modeled using mass, energy and momentum conservation equations. Peach Bottom tests indicate that dome pressures calculated to predict the data are higher than the experimental values. In the opinion of General Electric, the reason for the overprediction is that the energy equation for the dome region predicts that the bulk water mass very quickly becomes subcooled, the system becomes stiff, and therefore, the pressure rises very quickly. Since the rapid pressure rise leads to a rapid void collapse the staff concludes that the model is conservative. However, the Peach Bottom tests also indicate that ΔCPR predictions are not conservative. This implies that the conservatism of the bulk water model is offset by the nonconservatism somewhere else. General Electric did not quantify the conservatism in this particular model. In view of Peach Bottom tests where a trade off has occurred, no credit for conservatism can be given.

We find that the analytical methods used in these models are acceptable; however, as stated, no credit for conservatism will be given.

2. <u>Steam Line Model</u>

The steam line is modeled assuming single phase mass and energy conservation equations which are solved using an explicit finite differencing method. The steam is assumed to behave isentropically. The steam line is nodalized into six segments while the bypass line is modeled using two nodes. Safety and relief valve flow rates are treated as separate flow branches.

Sensitivity studies were performed by General Electric for various numbers of nodes for a sample test problem wherein the inlet pressure is kept constant and at the outlet turbine stop valve closure is simulated. These sensitivity studies were performed using nodal arrangements of 3,4,5,6,7, 8,20, and 40 nodes and compared with the analytical model predictions using the method of characteristics. The analyses indicate that a minimum of 7 nodes is required to predict frequencies to a reasonable degree. The comparison of amplitudes of pressure oscillations between the 8 node model and the analytical model is also reasonable. The conservatism of a model is dependent upon the integral of the pressure oscillations over a relatively short period of time since it is the integral of the pressure that is imposed on the core. The void collapse and the subsequent power increase is dependent upon the rate of change of this integral pressure. Judging from the pressure oscillations calculated from the 8 node model and the analytical model based on the method of characteristics the staff concludes that the integrated pressures are approximately the same for both models and perhaps there is a very slight conservatism in the 8 node model. Consequently, we find that the finite differencing scheme and the solution method employed in the steamline model are acceptable.

Other uncertainties in the model are in the form of friction loss coefficients and in the value of the average specific heat ratio. General Electric conducted sensitivity studies by varying the specific heat ratio and form loss coefficients. The Peach Bottom tests indicated an average specific heat ratio of 1.15. The change of this ratio to 1.25 caused an increase of 0.01 in the ΔCPR/ICPR ratio.

We reviewed the values of the average specific heat ratios for steam at 1000 psia. The value of 1.15 is valid for saturated steam with very little amount of droplets in it. The value of 1.25 is valid for a slightly superheated steam. Since there is a pressure drop along the steam line, we do not expect steam to be superheated. Hence, the value of 1.15 is acceptable. We also find the calculation of uncertainty of 0.01 in ΔCPR/ICPR ratio acceptable.

General Electric also performed a sensitivity study by decreasing the loss coefficient by 20%. This was based on the upper limit of steamline loss coefficient uncertainty. Decreasing the loss coefficient by 20% increases the ratio of ΔCPR/ICPR by 0.01. Decreasing the loss coefficient by 20% is a reasonable assumption and we find the calculation of uncertainty of 0.01 in ΔCPR/ICPR due to pressure loss coefficients acceptable.

In conclusion, our review indicates that the analytical methods used in steam line modeling and associated uncertainties are acceptable.

3. Core Thermal-Hydraulics Model

Two-phase mass, energy and momentum conservation equations were used to predict the behavior of the thermal-hydraulics of the core. Two mass and two energy conservation equations representing each phase separately and one momentum equation representing the mixture comprised the five equation model. In addition to these equations, correlations for 1) interfacial heat flux, 2) Zuber drift flux model (Reference 9), 3) two-phase pressure drop, and 4) heat transfer, are used.

The interfacial heat transfer correlation is based on the "mechanistic model" presented in Reference 9. The selection of the heat transfer correlations is based on the flow regimes. In the single-phase liquid region, the Dittus-Boelter correlation is used. In the subcooled and bulk boiling regions, the Jens-Lottes and Chen correlations are used, respectively. Two-phase pressure drop correlations are based on the Martinelli-Nelson correlation. The five equation model together with the correlations are solved using a fully implicit finite differencing method in the space-time domain. The space domain is one dimensional in the axial direction and the core is represented using 24 axial nodes.

To improve the accuracy of predictions within a node, a boiling boundary concept is defined. This concept defines a location in the axial direction for which the mixture enthalpy is equal to the enthalpy at which point subcooled boiling begins. This location establishes the boundary between the liquid and two-phase regions within an axial node at each time step and the program selects the appropriate correlation for the appropriate region. The variables solved for each node are volumetric flux, vapor fraction, pressure, vapor enthalpy and liquid enthalpy.

Two models are particularly significant in the assessment of uncertainties in the five equation model; they are Zuber drift flux and the subcooled boiling models. These are discussed in the following sections.

a. <u>Drift Flux Model</u>

The choice of two parameters, C_o and V_{gj} is important in this model. The first coefficient (C_o) is the concentration parameter which describes the slip due to cross sectional averaging of a nonuniform void fraction profile. The second tarm (V_{gj}) is the drift velocity which describes the local slip between the phases. The value of C_o is strongly dependent on the flow regimes and geometry. This dependence has been shown in many tests (Reference 9). The drift velocity is dependent on the density differences between the phases as well as on the flow regimes.

In the model used by General Electric, these parameters are empirically determined in the form of correlations based on the test data. The data were obtained both from tubes and channels, and are reported in References 10 through 14. When the vapor fractions obtained from these parameters were used to calculate power shapes observed in BWRs, some discrepancies were observed. Consequently, General Electric introduced another correlation for C_o, and a concept of neutron effective void fraction, to provide a better fit with measured power shapes. Based on physical considerations it is conceivable why C_o used in thermal hydraulic calculations is different from C_o for neutron power calculations. The thermal hydraulic C_o is based on tube geometry while neutron effective C_o is obtained from actual core geometry. The value of C_o should be different for tubes and rod bundles because of different vapor fraction profiles and flow regimes. However, in a telecon General Electric stated that C_o valid for thermal hydraulics gave good agreement with Atlas data and C_o valid for neutron effective void fraction gave good agreement with the core data. Hence, the differences cannot be explained based on geometrical considerations alone and there is an artificial fix in the model. According to Reference 34, this fix is necessary to compensate for deficiencies in lattice physics methods.

General Electric estimates that the uncertainty in the concentration parameter, C_o, is about ± 3% at a void fraction of .70 for neutron effective vapor fraction calculations. This corresponds to a ± 5% uncertainty in void reactivity coefficient which leads to an uncertainity of ± 0.008 in the value of ΔCPR/ICPR. However, General Electric uses ± 10% uncertainty in the value of C_o for thermal hydraulic calculations. We find no reason that the uncertainty in C_o for neutron power calculations should be different because the correlation is used to calculate voids the same way as in the thermal hydraulics. General Electric does not state any uncertainty in V_{gj} for neutron power calculations but states an uncertainty of ± 20% for thermal hydraulic calculations.

Based on Reference 15, we assessed the uncertainties for thermal hydraulic C_o ± 20% and V_{gj} ± 30% respectively. Reference 15 has a different data base from the references that the General Electric used. Extrapolating the General Electric results, we estimate that ± 20% uncertainty in C_o would result in ± 33% uncertainty in the void fraction or void reactivity coefficient.

We also reviewed the void fraction data taken in the FRIGG loop, Reference 16. The FRIGG tests were performed using rod bundles. The review of the data indicated that the scatter of ± 30% in void fraction was reasonable in the low quality region. This finding also substantiated the estimate of ± 20% uncertainty in the value of C_o.

Assuming the same uncertainty for the neutron effective C_o and extrapolating the General Electric results, we have estimated that the uncertainty of ± 20% in C_o resulting in ± 33% uncertainty in the void fraction or in the void reactivity coefficient would produce an uncertainty of ± 0.053 in ΔCPR/ICPR. We presented these findings in the ACRS hearing, Reference 30.

In response to the above staff assessment, General Electric submitted additional information, Reference 31, requesting the reduction of the uncertainty in ΔCPR/ICPR. The primary argument was that the uncertainty of ± 20% in the value of C_o (seven times the uncertainty of ± 3% which had been proposed by General Electric) leading to an uncertainty of approximately ± 30% in void fraction was applicable for a low quality and a low vapor fraction region. The uncertainty becomes smaller at higher qualities. In addition, General Electric submitted another sensitivity study using neutron effective $C_o = 1.0$ and noted that this would be the bounding value for ΔCPR calculations. General Electric also noted that the transient results were weakly dependent on void fractions at low qualities in the subcooled region, Reference 32.

We reviewed the new information submitted in Reference 31, and agree with General Electric that uncertainties in vapor fraction can be reduced at higher qualities and that $C_o = 1.0$ is a bounding value for bulk boiling. General Electric stated an uncertainty of ± 5% in void reactivity coefficient at a void fraction of 70%. This corresponds approximately to an uncertainty of ± 5% in void fraction. Further review of the void fraction data in the FRIGG loop shows a scatter of ± 10% in void fraction at qualities of 5% and 10%. These qualities are considered relatively high and they correspond to vapor fractions of 40% and 60% respectively. It appears that the FRIGG loop data show a larger scatter of void fraction than that assumed by General Electric. At high qualities, the uncertainty of ± 10% in void fraction corresponds to approximately ± 10% uncertainty in C_o. However, the theoretical limit for C_o in the bulk loading region is 1.00. It can be higher but not lower in this region. The scatter of 10% on neutron effective C_o would bring the value of C_o below 1.00. Some of the scatter in the FRIGG data was due to measurement errors which could be as high as ± 10%. Hence, we accept the limit of $C_o = 1.00$ as bounding for the uncertainty studies. Further, sensitivity studies performed in Reference 32 indicate that the transient is weakly dependent on changes of neutron effective C_o in low quality region. Hence, we accept the calculation of uncertainty of ± 0.011 in ΔCPR/ICPR as suggested in Reference 31. This is approximately 30% larger than that originally proposed by General Electric.

b. <u>Subcooled Boiling Model</u>

The phenomenon of subcooled boiling is modeled by the "Mechanistic" subcooled boiling model developed by R. T. Lahey in Reference 9. The model provides a relationship for interfacial heat flux between the bubbles and surrounding liquid. It consists of two terms: one term shows the effect of the temperature difference between the phases and the other shows the effect of the wall heat flux.

The model has been verified using the data obtained by S. Z. Rouhani (References 17 and 18). These data were obtained from a vertical annular channel. In determining the uncertainty of the correlation, General Electric provided a sensitivity study using a coefficient "n" in the correlation. The nominal value of "n" is 1.0. For n = 1.25, a change of 0.009 in ΔCPR/ICPR is obtained. If 1.50 is assumed, the change in ΔCPR/ICPR is 0.014. GE states that the value of 1.25 provides a reasonable uncertainty for the model but does not provide any supporting evidence or data.

We reviewed the void fraction vs. axial height curves drawn for various "n" values and find that the void fraction difference between the two curves drawn for n = 1.0 and n = 1.5 is about 3.5% in absolute or 18% relative to the average measured value of the void fraction in the subcooled region. Some of the rod bundle experiments performed in the Frigg loop (Reference 16) show 100% (relative) scatter of the data. In general, the scatter is 15 - 30% relative to the average void fraction. We believe ± 30% scatter is a reasonable estimate of uncertainty. Therefore, we increased the uncertainty in ΔCPR by a factor of 1.67 (30/18) which results in ± 0.023 in the uncertainty value of ΔCPR/ICPR for the subcooled boiling model. We estimate the corresponding minimum and maximum values of "n" to be 0.5 and 2.0 respectively. General Elective is required to make sensitivity studies to verify that these values correspond to ± 0.023 uncertainty in ΔCPR/ICPR.

The review of the analytical models describing the thermal-hydraulic behavior of the core indicates that these models are acceptable provided the uncertainties of various components are increased to the values recommended by the staff.

4. Core Physics Model

a. Assumptions in the Neutronics Model

The neutronics model of ODYN is based on time-dependent, one-dimensional, one-group, diffusion theory. The model includes the effect of delayed neutrons and the calculation is performed in the axial dimension of a BWR. Radial effects due primarily to Doppler, moderator, and control state are taken into account in collapsing a three-dimensional model to the one-dimensional axial model. Of the three effects, the control state variation due to scram during a transient is the most important. Some care must, therefore, be taken in choosing the initial weighting functions to account for these effects.

We have reviewed the assumptions in this neutronic model with the current state-of-the-art for performing space-time coupled neutronic and thermal-hydraulic calculations. We conclude, based on our review, that the assumptions on which the neutronic model of ODYN are based are acceptable.

b. Derivation of Equations for the One-Group, One-Dimensional, Time-Dependent Neutronic Model

We have followed in a step-by-step manner the derivation of the one-group, space-time neutronics model presented primarily in

Appendix A of Volume I of the report. This derivation proceeds from the time-dependent form of the three-dimensional neutron diffusion equation for the fast flux as used by the General Electric three-dimensional reactor simulator (Reference 2) along with appropriate equations for delayed neutrons. The three-dimensional time-dependent neutron flux is represented as a product of radial and axial time-dependent components. Weighting functions are next introduced to make this factorization unique and to minimize errors in the procedure in some sense. The weighting functions are taken, according to the adiabatic approximation, as the solution to a steady-state eigenvalue problem to be solved at various points in time. In practice, the weighting functions are calculated only at time zero for as many BWR operating states as is necessary. This procedure results in the final form used for the one-group, one-dimensional, time-dependent equations along with defining equations for the nuclear parameters that are used. The derivation also includes discussion of the average axial power distributions, initial normalization procedures, and boundary conditions.

Section 5 of Volume I of the report discusses the integration of the spatial and time variables to obtain the discrete form of the one-group, one-dimensional, time-dependent equations. The procedures used for this are straight forward. This section also discusses the radial weighting function and the treatment of the control state. Cross section related parameters are functions of axial core height, control state, and relative water density. These parameters are fit to quadratics in the relative water density.

Our review of the derivation of the equations for the one-group, one-dimensional, time-dependent neutronics model has been performed by deriving and verifying each of the equations presented in Volume I of the report. We conclude, based on our step-by-step review of all the neutronic equations, that the derivation of nuclear parameters and equations for the one-group, one-dimensional, time-dependent model is acceptable.

c. <u>Calculation of Neutronic Input Parameters</u>

General Electric uses its Lattice Physics Model and its Three-Dimensional BWR Core Simulator to process nuclear data for the ODYN code. The Lattice Physics Model is described in Reference 19. The Three-Dimensional BWR Core Simulator is described in Reference 2. Both of these codes have been reviewed and approved by the NRC for use in BWR applications.

The Lattice Physics Model, as its name implies, is used to generate nuclear parameters for use as input to the BWR Core Simulator. This data is generated as a function of fuel type, control, temperature, void fraction, void history, and exposure. Before being used by the BWR Core Simulator, the data is transformed from the Lattice Physics Model void fraction to the Neutron Effective Void (NEV) model void fraction. This empirical procedure was developed by GE to remove a discrepancy between BWR Core Simulator results and operating reactor data. The BWR Core Simulator is used to perform the three-dimensional analyses that are required for obtaining the data for processing into parameters and cross sections for the ODYN code.

Our review of the calculation of neutronic input parameters is based on the use of NRC reviewed and approved codes and on comparisons of three-dimensional and ODYN steady-state neutronic analyses. The approved codes are (1) the Lattice Physics Model (NEDE-20913-P, "Lattice Physics Methods," C. L. Martin, June 1976 and NEDO-20939, "Lattice Physics Methods Verification," C. L. Martin, June 1976) and (2) the BWR Core Simulator (NEDO-20953, "Three-Dimensional BWR Core Simulator," J. A. Woolley, May 1976 and NEDO-20946, "BWR Simulator Methods Verification," G. R. Parkos, May 1976). The steady-state calculations compared the BWR Core Simulator and ODYN results for scram reactivity and core averaged axial power distributions, among other things, for a number of different reactors and operating states.

Some of the uncertainty values used by General Electric in response to our Question 12 need to be revised in our judgement. We believe that the Doppler reactivity coefficient uncertainty should be increased from ± 6 percent to about ± 10 percent. This increase is based on the uncertainties inherent in the calculation of Uranium-238 resonance absorption, the calculation of the Dancoff factor in the complex BWR lattice, the calculation of spatial weighting factors, and the computation of effective fuel temperatures. This change in the Doppler uncertainty will have very little effect on the calculated ΔCPR/ICPR ratio. We estimate that this will increase the uncertainty in ΔCPR/ICPR from ± 0.0015 to ± 0.002. We believe that the scram reactivity uncertainty should be increased from ± 4 percent to about ± 10 percent. This increase is based on the uncertainties inherent in calculating the initial scram reactivity rate and total control rod worths. We estimate that this will increase the uncertainty in ΔCPR/ICPR from ± 0.01 to ± 0.02. The General Electric values for the Lattice Physics Model and BWR Core Simulator uncertainties in the void reactivity coefficient calculation are acceptable as given in response to our Question 12.

Since the uncertainties in the neutron effective void fraction are assessed in the thermal hydraulic section, we did not consider it as part of void reactivity uncertainty. Hence, the uncertainty in ΔCPR/ICPR value is reduced from ± 0.020 to ± 0.018.

We conclude, based on our review, that the procedures and calculations performed to provide the neutronic parameters for input to the ODYN code are acceptable.

5. <u>Fuel Heat Transfer Model</u>

Heat transfer to the coolant and temperatures within the fuel are calculated assuming a single cylindrical fuel element for each axial location. The fuel heat transfer model used in the ODYN code calculates fuel temperatures as a function of time in the transient as input to the Doppler reactivity calculation. The cladding wall temperatures are also calculated as input to the transient cladding-to-coolant heat transfer model. The ODYN code allows for axial variation of the neutron flux, as well as of coolant flow, density, and pressure. This results in an axially varying set of input conditions for the fuel heat transfer model. The resulting temperature calculations are then solved for a series of discrete axial elevations in the core.

The fuel and cladding conductivity and heat capacity are assumed to be temperature dependent. A gap thickness is specified between the fuel and the cladding and an input gap conductance is used. Axial and time variations in the gap conductance may be given, but a constant value is used for safety analyses. The external heat transfer coefficient and coolant temperature are obtained from the thermal-hydraulic portion of the code. The heat generation rate in the fuel pellet is obtained from the axial power distribution which is determined by the neutronics segment of ODYN. The radial heat distribution in the fuel rod is assumed to be independent of axial position and independent of time.

General Electric derived the fuel heat transfer model from the general heat flow equation. The equation is expressed with axisymmetry and zero axial conduction assumed. The resulting, one-dimensional, transient heat conduction equation is solved by the Crank-Nicholson finite-difference technique. The solution is approximate, but the procedure is widely practiced and is well documented in the open literature. General Electric has limited its description of the fuel heat transfer model to the formulation of this final equation.

The resulting heat conduction equation is applied to a single rod with a radially averaged heat generation rate. This rod is used to represent all of the fuel rods in the reactor core. Because axial conduction is assumed to be negligible, the equation can be solved independently for each discrete axial position in the core. The finite-difference technique also requires a radial nodalization of the fuel rod. The nodes may be of arbitrary size. General Electric has assumed that the fuel pellet is divided into seven radial nodes and the cladding into two nodes. The coefficients for both the steady-state and transient forms of the resulting finite-difference equations are given in Tables 7-1 and 7-2 of the model description.

A number of limiting assumptions have been considered in our review of the fuel heat transfer model.

1. The ODYN core transient model is designed to handle short-term events which occur on a time scale of seconds. This makes it possible to ignore the effects of long-term fuel behavior phenomena, such as creep and swelling. Pre-transient conditions, such as the average fuel-to-cladding gap size, are calculated with more detailed fuel performance codes, such as GEGAP-III (Reference 20) and subsequently used as input to ODYN.

2. The ODYN core transient model is designed to handle average, rather than extreme, fuel conditions. The fuel rod input parameters represent an average of all fuel rods at a given axial location. Bounding input parameters are, in general, more difficult to establish and thus are more critical in the overall analysis. However, the ODYN code, as a whole-core analysis, requires only the average conditions. In this respect, we note that the ODYN transient model does not have a hot channel capability, where extreme fuel conditions would be required as input.

Both of these limiting assumptions were considered in our review of the gap conductance values used by the ODYN code. We have reviewed (Reference 21) the selection of the axial and time variation of gap conductance to determine whether the selected values are appropriate for different transients. General Electric stated that the core average gap conductance values are calculated by GEGAP-III (Reference 20) which is approved by NRC. The calculated conductance is input for all axial nodes and is kept constant during the transient.

A sensitivity study was also performed for the most limiting pressurization event in which the ΔCPR decreases when axial varying gap conductance is used. It was shown that most of the high power axial nodes have higher than core average gap conductance. During the transient, higher gap conductance will lead to faster heat transfer from the fuel to the moderator/coolant which generates more steam voids. This results in lower stored heat in the higher power nodes. In addition, the faster conversion of fuel stored energy to steam voids in the core helps to mitigate the transient due to negative void reactivity feedback. Therefore, the transient with axial varying gap conductance is less severe than that with constant gap conductance.

During limiting pressurization transients, it is expected that the fuel gap conductance will be higher than its initial steady-state value due to the increase in the thermal expansion of the fuel pellet. As discussed above, higher gap conductance leads to a less severe transient. General Electric has not taken credit for this fact, but has stated that the use of constant conductance throughout the transient compensates for uncertainties in thermal conductivity and specific heat of the fuel and cladding. We have examined these properties and find that they are appropriate over the temperature range specified by GE (300-1500 °K for fuel thermal conductivity). Therefore, it is concluded that the use of a constant, core average gap conductance in the proposed ODYN licensing calculations is appropriate.

We have also questioned the use of a specific core average gap conductance value of 1000 Btu/hr-ft^2-°F for the analysis of the Peach Bottom Unit-2 turbine trip event. General Electric has shown (Q-11, Volume II) the calculated peak neutron flux as a function of time for gap conductance values of 500, 1000, and 1500 Btu/h-ft°-°F. Small differences in neutron flux are observed for the 500 and 1000 Btu/hr-ft^2-°F values. This is because the entire flux pulse is only a few tenths of a second wide and a fast fuel time constant is needed to produce a moderate density feedback through the rod heat flux. The peak neutron flux is minimum for the 1500 Btu/hr-ft^2-°F value, showing that large values of gap conductance will mitigate the calculated flux response. This conclusion is in agreement with that found for axial and time varying conductance values. It also shows that a core average gap conductance value of 1000 Btu/hr-ft^2-°F is not, in itself, an adequately qualified conductance value for core transient analyses. We conclude that conductance values should be based on an approved fuel performance code.

We have also reviewed the use of a radially averaged heat generation rate rather than a radially-dependent heat generation rate. We questioned the conservatism of this assumption because flux depressions, and therefore a radially-dependent heat generation rate is expected in BWR fuels. General Electric has acknowledged that the radial power distribution within the fuel rod is not uniform. This is because the plutonium build-up and self-shielding of the fuel results in a radial power shape peaked sharply at the outside of the fuel pellet. Heat transfer from the inside of the pellet to the cladding occurs by diffusion through the fuel material. When the power is peaked at the outside of the pellet, the average distance from the area of maximum heat generation to the edge of the pellet is less. This results in a shorter time constant than in the uniform power production case. A reduction in the thermal time constant results in faster feedback of heat flux to the moderator/coolant and reduces the consequences of the pressurization transient in the same manner that higher gap conductance does. Hence, a uniform power distribution assumption inside the fuel pellet is conservative from the moderator/coolant standpoint.

Although the use of a uniform radial pin power distribution and small gap conductance values lead to conservative moderator/coolant conditions, these assumptions also lead to higher fuel temperatures. The higher fuel temperatures, in turn, lead to increased Doppler broadening in the fuel pin which is non-conservative for transient analysis. The ODYN code assumes that all fuel at the same axial location in the core has the same temperature profile. Analyses have shown that this approach may tend to underestimate the Doppler reactivity effects because the fuel pins which have the greatest resonance capture rates are near the bundle periphery and operate at higher average temperature than that calculated by the code. This assumption is valid only for fuel assemblies with uniform enrichment. However, the Doppler reactivity contribution to BWR transient analysis appears to be of lesser importance than the scram and moderator void reactivity contributions. The use of a uniform radial pin power distribution is therefore appropriate in the analysis of events where Doppler reactivity effects are small.

We have also questioned the application of the Crank-Nicholson method to the fuel heat transfer equation. This method suffers complications when heat generation varies with position and time, when thermal properties vary and when non-linear boundary conditions are used. General Electric has stated that the method of solution suffers complications only when the time steps are too large relative to the fuel thermal time constant or when the fuel properties change more rapidly than the time step of the solution. It was further stated that the BWR fuel thermal time constant is in the range of 5-8 seconds compared to 0.01 second time steps taken by the ODYN code. Such extensive time stepping is required for the hydraulic analysis and will accommodate all non-linearity problems of the fuel behavior. It was also noted that the gap conductance is conservatively held constant in the transient calculation. We therefore conclude that the method of solution is appropriate for safety analyses.

In summary, we find that the ODYN fuel heat transfer model is appropriate for whole-core analysis of short-term events. We note that the code is used for whole-core analysis and is not proposed for hot channel calculations. We have also examined the list of events selected (Volume III, table 2-1) for analysis with ODYN and find that these events are of short duration or are limited in expected fuel temperature increase. We conclude, therefore, that the ODYN fuel heat transfer model is appropriate for the safety analysis of these events.

6. Summary of Code Uncertainties

a. Margin in ΔCPR Calculations

In summary, the staff agrees with some of the code uncertainties calculated by General Electric. However, some of the code uncertainties are low and the staff recommends higher values. A comparison of the code uncertainties and the corresponding bounding values as recommended by General Electric and the staff is presented in Table I.

General Electric claims an expected conservative bias of 0.02 (Table 3-3, Volume III) in the calculation of the value of ΔCPR/ICPR due to the modeling of the gap conductance. However, the sensitivity studies performed using different values of gap conductance (Q-11, Volume II) as well as the comparison of the Peach Bottom test data with the ODYN predictions do not indicate that such a conservatism in ΔCPR calculations exists. Consequently, we do not believe that the predictions have a conservative bias.

Our review shows that the ODYN code is a best estimate code and there is no inherent conservatism in predictions of ΔCPR/ICPR when best estimate input values are used. Consequently, we do not give credit for this claimed conservatism of 0.02.

General Electric estimated the total code uncertainty (Table 3-3, Volume III) using the method of linearization. This method can

TABLE-I

COMPARISON OF CODE UNCERTAINTIES AND CORRESPONDING
BOUNDING VALUES AS ESTIMATED BY
GENERAL ELECTRIC AND THE STAFF

	GE		STAFF	
	Bounding Values of Parameters	$\pm\Delta CPR / ICPR$	Bounding Values of Parameters	$\pm\Delta CPR / ICPR$
I. Reactor Core Model				
(1) Nuclear Model				
(a) Void Coefficient	$\alpha_v \pm 13\%$	0.020	$\alpha_v \pm 11\%$	0.018
(b) Doppler Coefficient	$\alpha_d \pm 6\%$	0.002	$\alpha_d \pm 10\%$	0.002
(c) Scram Reactivity	$\alpha_s \pm 4\%$	0.010	$\alpha_s \pm 10\%$	0.020
(d) Prompt Neutron Heating		0.006		0.006
(2) Thermal Hydraulic Model				
(a) Drift Flux Parameters	$C_o \pm 3\%$		$C_o = 1.00$	
	$V_{gj} \pm 20\%$	0.008	$V_{gj} = \pm 30\%$	0.011
(b) Subcooled Void Model	n = 1.25	0.009	n = 0.5 2.0	0.023
(3) Fuel Heat Transfer Model				
(a) Pellet Heat Distribution	(Conservative)		-	-
(b) Pellet Heat Transfer Parameters	(Conservative)		-	-
II. Recirculation System Model				
(1) System Inertia	(L/A) + 200%	0.002	L/A + 200%	0.002
(2) Jet Pump losses	K - 20%	0.010	K - 20%	0.010
(3) Core Pressure Drop	Δ + 1.5 psi	0.005	Δp + 1.5 psi	0.005
(4) Separator (L/A)	-30%	0.002	-200%	0.015
(5) Separator ΔP	(Conservative)		-	-
III. Steam Line Model				
(1) Pressure Loss Coefficients	K - 20%	0.010	K - 20%	0.010
(2) Specific Heat Ratio	γ + .10	0.010	γ + .10	0.010
Total:		0.031		0.044

estimate the output distribution only approximately. The method also assumes the independence of the parameters. The appropriateness of the linear method should be verified by response surface and Monte Carlo analyses. However, as will be shown subsequently, the results of the statistical analyses performed in Volume III are not acceptable. New statistical analyses, if performed by General Electric, should be based on code uncertainties based on comparison of code predictions with the test data. Consequently, we use the value of total code uncertainty calculated from model sensitivity studies and method of linearization in determining the margin of ΔCPR/ICPR in Option A (to be presented in Staff Position) where statistical analysis is not required. The total code uncertainty in Table 3-3 of Volume III as per General Electric is ± 0.031. Based on our review we increase this value to ± 0.044.

b. <u>Margin in Pressure Calculations</u>

General Electric has not performed analyses to determine the uncertainties in the calculation of pressure. Hence, it will be necessary for General Electric to perform these calculations using staff recommended values of the parameters listed in Table I for the Main Steam Isolation Valve closure event. We believe that there is sufficient conservatism in the ASME vessel overpressure limit to permit General Electric to use approximate linear methods to determine the uncertainty in the output. This uncertainty (2σ) should be added to the ODYN calculated pressure. If General Electric demonstrates that this uncertainty is very small (e.g., by a factor of 10 or more) relative to the uncertainty in determining ASME vessel overpressure limit, no addition of uncertainty to the calculations of pressure is needed.

B. QUALIFICATION OF THE ODYN CODE

1. Qualification of Neutronics Model - Comparison of ODYN with BWR Core Simulator

One of the ways in which the ODYN code may be qualified is by comparison of ODYN results with those obtained by using other codes and analytical methods. These comparisons should include both steady-state and dynamic calculations. A calculation of a BWR turbine trip without bypass licensing basis transient is compared in a later section to a calculation performed by our consultants at Brookhaven National Laboratory (BNL). This section will discuss some steady-state comparisons made by General Electric of ODYN and the BWR Core Simulator.

The BWR Core Simulator Code (NEDO-20953, "Three-Dimensional BWR Core Simulator," J. A. Woolley, May 1976 and NEDO-20946, "BWR Simulator Methods Verification," G. R. Parkos, May 1976) has been reviewed and approved by the NRC. This code, as used by General Electric, predicts measured power distribution peak to average ratios as follows:

(a) Axial power distribution	- 5% for uncontrolled assemblies
	- 10% for controlled assemblies
(b) Radial power distribution	- 5% underestimate relative to the process computer
(c) Nodal power distribution	- 4% for gamma scan data
	- 7 to 8% for process computer data

The BWR Core Simulator calculation of the criticality of first cycle and reload BWRs results in a small bias which is taken into account for reactivity determinations of cold, xenon-free and hot operating conditions. The standard deviation of these criticality calculations is about 0.002 in units of reactivity.

The quantities to be compared are the core averaged axial power shape, the scram reactivity, and the void reactivity coefficient. These neutronic parameters were selected for comparison because of their importance in the turbine trip without bypass licensing basis transient. In addition, it is the space time evaluation of these quantities that distinguishes the ODYN calculation from a point kinetics evaluation of pressurization type transients.

The comparison of the core averaged axial power distribution, as computed by the BWR Core Simulator and ODYN, is given by the response by GE to our Question 36. This response states that the collapsing scheme employed in the generation of nuclear parameters ensures that the steady-state core averaged axial power distribution and criticality computed by ODYN are identical to the BWR Core Simulator results. The response also indicates that, for a number of plants and operating states, The ODYN core averaged axial power distribution agreed to within 0.5 percent of the results obtained with the three-dimensional BWR Core Simulator.

The scram reactivity was compared for three BWR-4 reactor operating states. The initial scram rate (ISR), defined as the scram reactivity insertion rate during the first second from the time scram is initiated, is the quantity chosen for comparison rather than the total scram worth. The ISR has been shown to be a critical quantity for short duration power burst transients of the load rejection type.

One operating state was for the beginning of cycle 2 but with all control rods out. ODYN results for the neutron flux and scram reactivity as a function of control rod insertion (or time) compared well with results obtained with the BWR Core Simulator. The ISR for ODYN was about 0.93 of the value obtained with the BWR Core Simulator. The second comparison was for the same BWR-4 but with some control rods inserted to achieve a critical condition. The ISR for ODYN was about 0.86 of the value obtained with the BWR Core Simulator. The third comparison was for another BWR-4 at 50 percent of full rated power and 100 percent of full rated core flow. This reactor had a considerable control rod inventory and, therefore, provides a severe test for the ODYN code. The ISR for ODYN was about 0.94 of the value obtained with the BWR Core Simulator. For all three comparisons the ODYN result for ISR was smaller than that obtained with the BWR Core Simulator and was therefore conservative.

The void reactivity coefficient derived from ODYN calculations was compared to the void reactivity coefficient derived from the BWR Core Simulator. This coefficient is obtained by knowing the core averaged void fraction and reactivity at two different reactor operating states. The different reactor operating states were obtained by changing either the reactor pressure or flow. For the variety of cases examined, GE states that the ODYN and BWR Core Simulator void reactivity coefffcients agree to within 5%.

Our review of the comparison of steady-state BWRs calculated using the one-dimensional ODYN code with comparable calculations using the three-dimensional BWR Core Simulator code has been performed (1) by reviewing GE results for the scram reactivity and void reactivity coefficients and (2) by reviewing the GE response to our request for additional information on steady-state comparisons between the two codes.

We conclude, based on our review, that steady-state ODYN code calculation of core averaged axial power distributions, scram reactivity, and void reactivity coefficients are either in-good agreement with or conservatively calculated with respect to comparable steady-state results obtained with the BWR Core Simulator code.

2. Qualification of the Thermal Hydraulic Model

Several comparisons of the ODYN thermal hydraulic model to standard GE design models were performed. The standard GE design model was submitted in Reference 1 and was approved by NRC in Reference 4. Both steady state and transient conditions were analyzed.

The steady state analysis first compared the thermal hydraulic characteristics (void fraction vs. axial location) of two typical BWR fuel channels (high and low power channel). The results of this comparison show good agreement between the models. This was expected since both models are very similar. The maximum void fraction variation between these models was approximately 5% for the high power channel and about 17% for the low power channel. These variations are for the axial locations where the void reactivity change is expected to be most significant for the transient calculations, i.e., >4 ft axial height. The steady state analysis also compared the change in void fraction vs. axial location for a 10 psi pressure change. The maximum void fraction variation between the models for this comparison was approximately 0.5% for the high power channel and about 5% for the low power channel. These variations are within the range of uncertainty for these type of thermal hydraulic calculations. See also discussions in Section II.A.3.a and b.

For the transient analysis comparison, the ODYN channel thermal hydraulic model result was compared to an analytical solution for exponential flow decay. The comparison required that ODYN be modified to include a constant axial heat flux distribution, and steam and drift flux properties. This was done because the tests were run with a uniform axial heat flux and calculational convenience required the choice of constant steam and drift flux properties. The tests were performed using a single heated tube containing Freon-114 at relatively low pressures and temperatures; about 125 psia and 160°F respectively. There is ± 10% uncertainty in the measurements of the void fraction. The calculational uncertainty seems to be on the order of ± 5%. These tests verified that the analytical modeling technique including the drift flux model is acceptable and can be used to predict the vapor fraction. Judging the comparisons between the predictions and the test data and the special nature of these tests, the staff estimates that an uncertainty of ± 30% in transient void fraction in low qualities and ± 5% in high qualities for a rod bundle geometry during reactor transients is reasonable and consistent with the findings in the analytical model review in the previous section.

3. <u>Qualification Using Integral Tests</u>

In the past several years General Electric has undertaken a test program to verify the analytical methods for reactor pressurization transients. The tests of major interest for the current discussion consist of four turbine trip experiments. Three of these tests were performed at Peach Bottom Unit 2 (PB-2) in April 1977 and the remaining test was performed at a foreign reactor (KKM) in June 1977. These tests provide the experimental data base for verification of the ODYN code. The test results will be summarized in this section. A detailed description of the PB-2 test is presented in Reference 22.

General Electric stated that ODYN has been developed from first principles and independent of these results. The staff notes that in the ODYN code the only artificial fix is the neutron effective void correlation. The comparisons with integral plant tests provide an independent check of the ODYN code. The evaluation concentrates on the differences between test results and corresponding ODYN predictions. The parameters which are considered in these comparisons are steamline pressure, reactor vessel dome pressure, core exit pressure, and transient neutron flux distribution. These parameters are of primary importance in simulation of the pressurization transient. An accurate, ODYN simulation of these parameters would provide some verification of the assumptions for the transient models.

a. <u>Peach Bottom Tests</u>

The inputs used for this comparison were best estimate or measured values for the current (April 1977) Peach Bottom Unit 2 EOC2 conditions. The three Peach Bottom Unit 2 (PB-2) tests were conducted at power levels of 47.4, 61.6, and 69.1 percent of full rated power. The tests were intended to be conducted at 100 percent of the rated flow. However, the second test was conducted at 82.1 percent of rated flow due to xenon. These three tests had different control rod distributions and fractions. For these three tests the first scram signal on the position of the turbine stop valve was disabled so that the scram would occur on high neutron flux. Disabling of the primary scram signal was necessary to obtain a significant power increase as a function of time for tests. Control rod insertion was assumed to vary linearly with time and was based on measured data. A constant value of 1000 BTU/hr-ft^2-°F was used for the fuel rod gap conductance. Sensitivity studies performed by General Electric showed that the neutron flux as a function of time was insensitive to large changes in the gap conductance for these tests (See Q-11, Volume II).

The GE BWR Core Simulator was used to generate three-dimensional power distributions and to collapse the nuclear parameters according to the ODYN procedures. The initial core averaged axial power distribution calculated with ODYN can then be compared with power distributions obtained with the PB-2 process computer. Comparison shows that ODYN agrees quite well with the process computer core average axial power distributions for all three tests. This means that the GE neutronic procedures for generating nuclear parameters are internally consistent and provides the proper initial conditions for the start of the transient calculations.

A comparison of the total core power as a function of time provides an integral test of the important reactivity feedback due to scram and moderator density changes. This comparison would also be indicative of the adequacy of the core pressure and inlet flow calculations. The comparison shows that ODYN predicts the initial and fall-off part of the turbine trip transients correctly but overpredicts the peak total core power response for all three tests. It should be noted that the calculated consequences of the turbine trip tests are sensitive to scram delay time and the power fraction for prompt moderator heating. It should also be noted that small changes in reactor operating state conditions such as, for example, core pressure, cause relatively large changes in the flux transient because of the large net reactivity of the transients.

The reactivity components displayed for these ODYN calculations show that when scram occurs the power burst is quickly quenched. This is due to the control rod distribution and fraction for each test. The Doppler reactivity component plays only a secondary role. The reactivity components again demonstrate the necessity for their accurate assessment in any calculations of these type of transients.

A further indication of the adequacy of the ODYN calculation can be ascertained by comparing the core power as a function of time at the Local Power Range Monitor (LPRM) detector positions. The miniature fission detectors that comprise this LPRM system are distributed both radially and axially within the reactor core. Analysis of the PB-2 data shows that the radial variation of the neutron power with time is similar for each detector on an axial level. This means that a one-dimensional axial calculation such as ODYN should be an adequate representation for these tests. The neutron power as a function of time does vary, however, with axial position as shown by the experimental data for the A, B, C, and D level LPRMs which are located at 1-1/2, 4-1/2, 7-1/2, and 10-1/2 feet from the bottom of the core. Comparison of the ODYN results for the neutron power as a function of time for these four detector levels with test data shows similar trends as observed for the total core power. The ODYN results and test data agreement for each LPRM level is similar to that for the total core power for each of the tests. This indicates that the ODYN calculation correctly models the axial neutron flux variations as a function of time for these PB-2 turbine trip tests.

Figures 1 through 3 (reproduced from NEDO-24154) present comparisons of axial neutron flux variations as measured (calculated by the process computer) and calculated by the ODYN code before the initiation of the tests. Figures 4 through 6 present comparisons of prompt neutron power as measured and calculated by the ODYN code during the tests.

The ODYN calculated steamline pressure compares well with the PB-2 data for predicted wave travel time and frequency of pressure oscillations. However, the calculated pressure curves are more spread out and the amplitudes are smaller than the measured steamline pressure. General Electric has attributed this difference to the coarseness of the spatial mesh in the steamline modeling. As

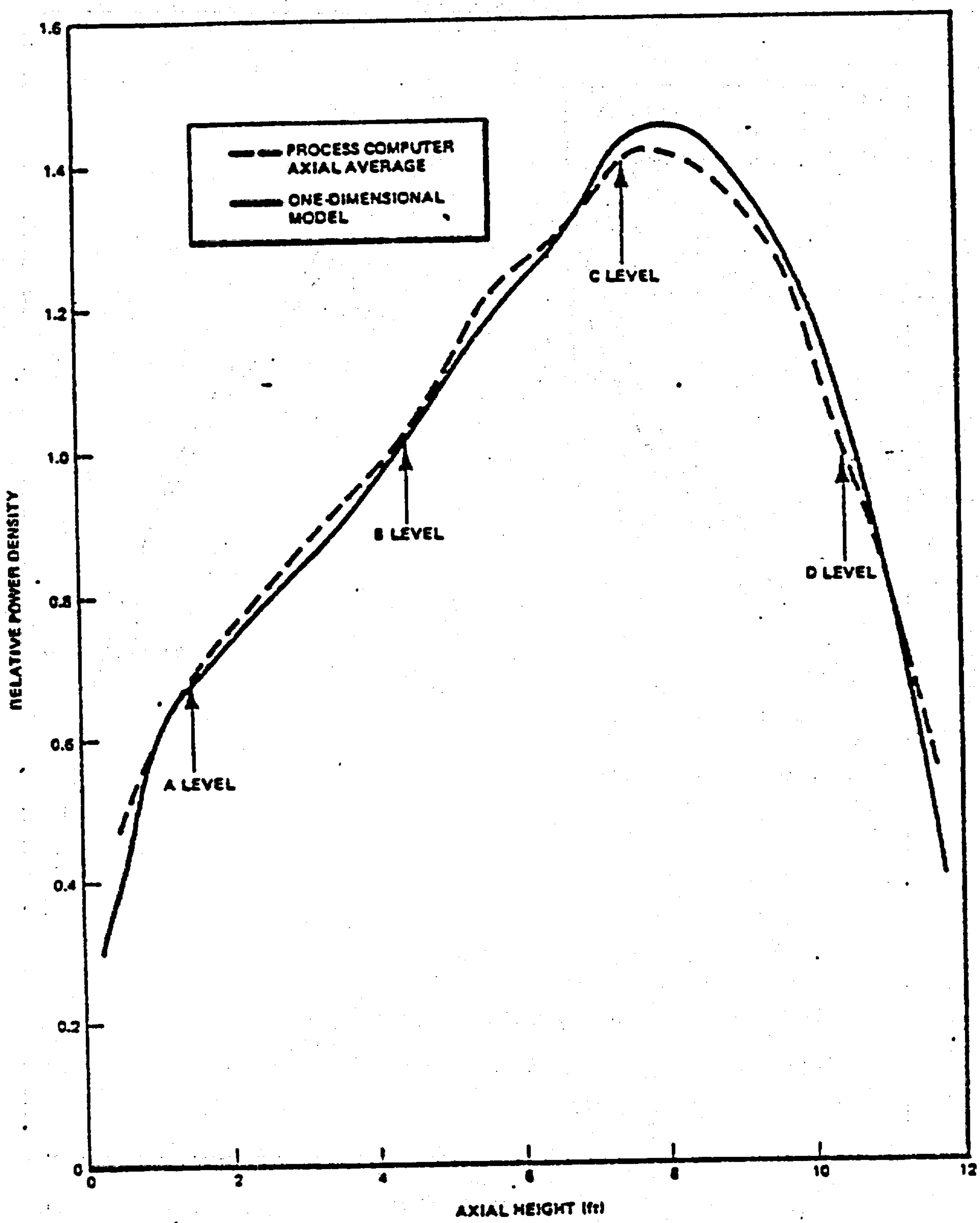

Figure 1 Axial Power Profile Turbine Trip 1

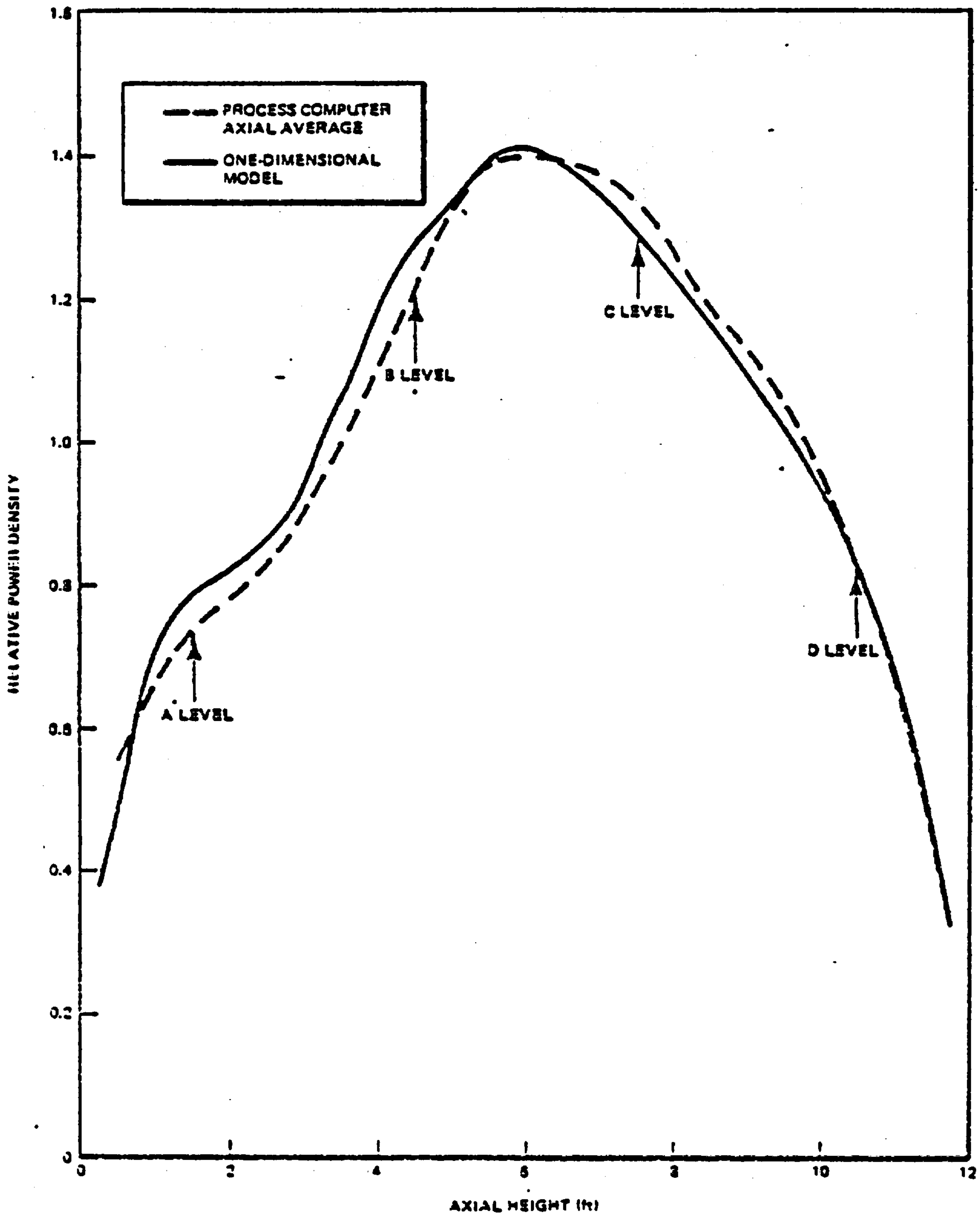

Figure 2 Axial Power Profile Turbine Trip 2

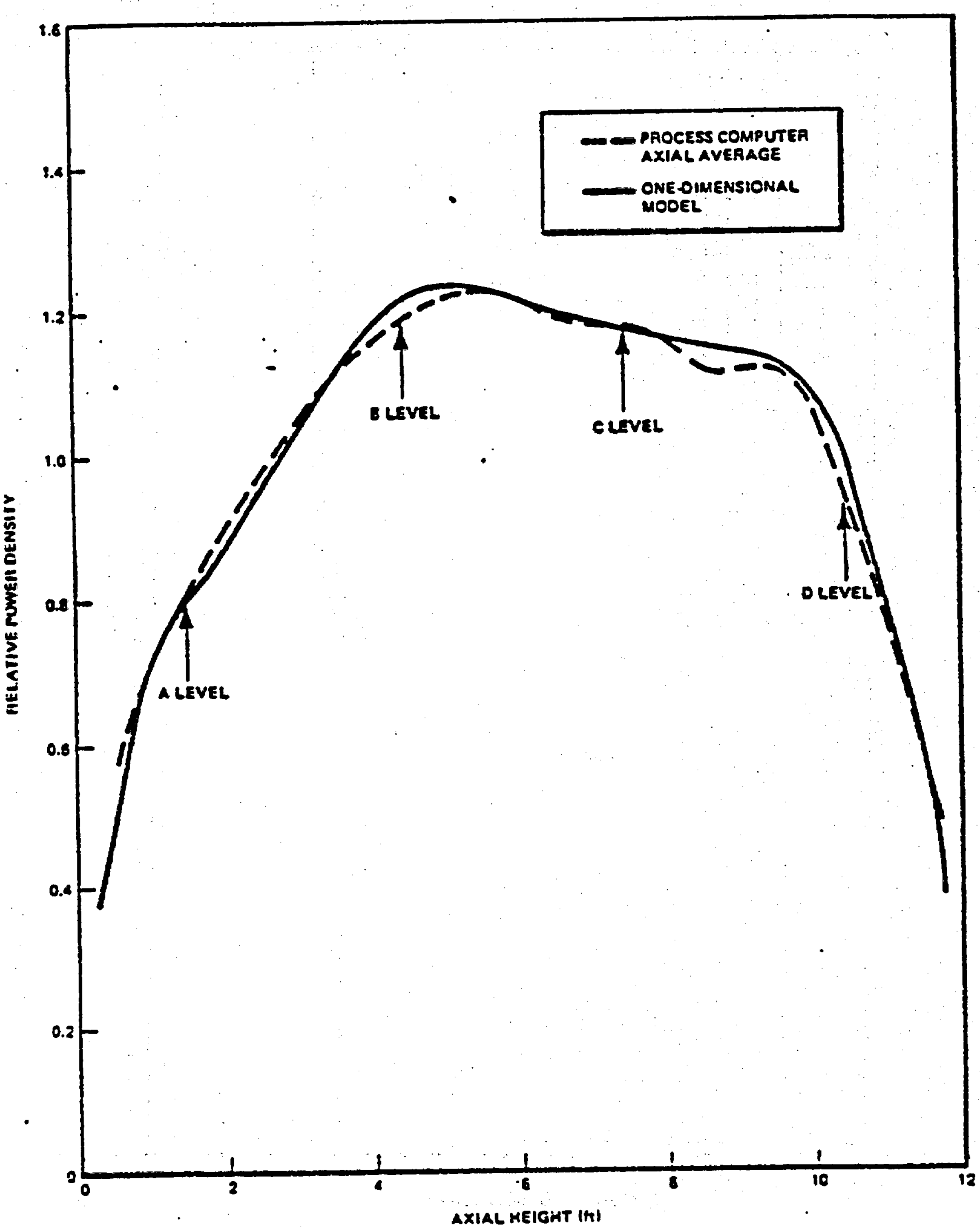

Figure 3 Axial Power Profile Turbine Trip 3

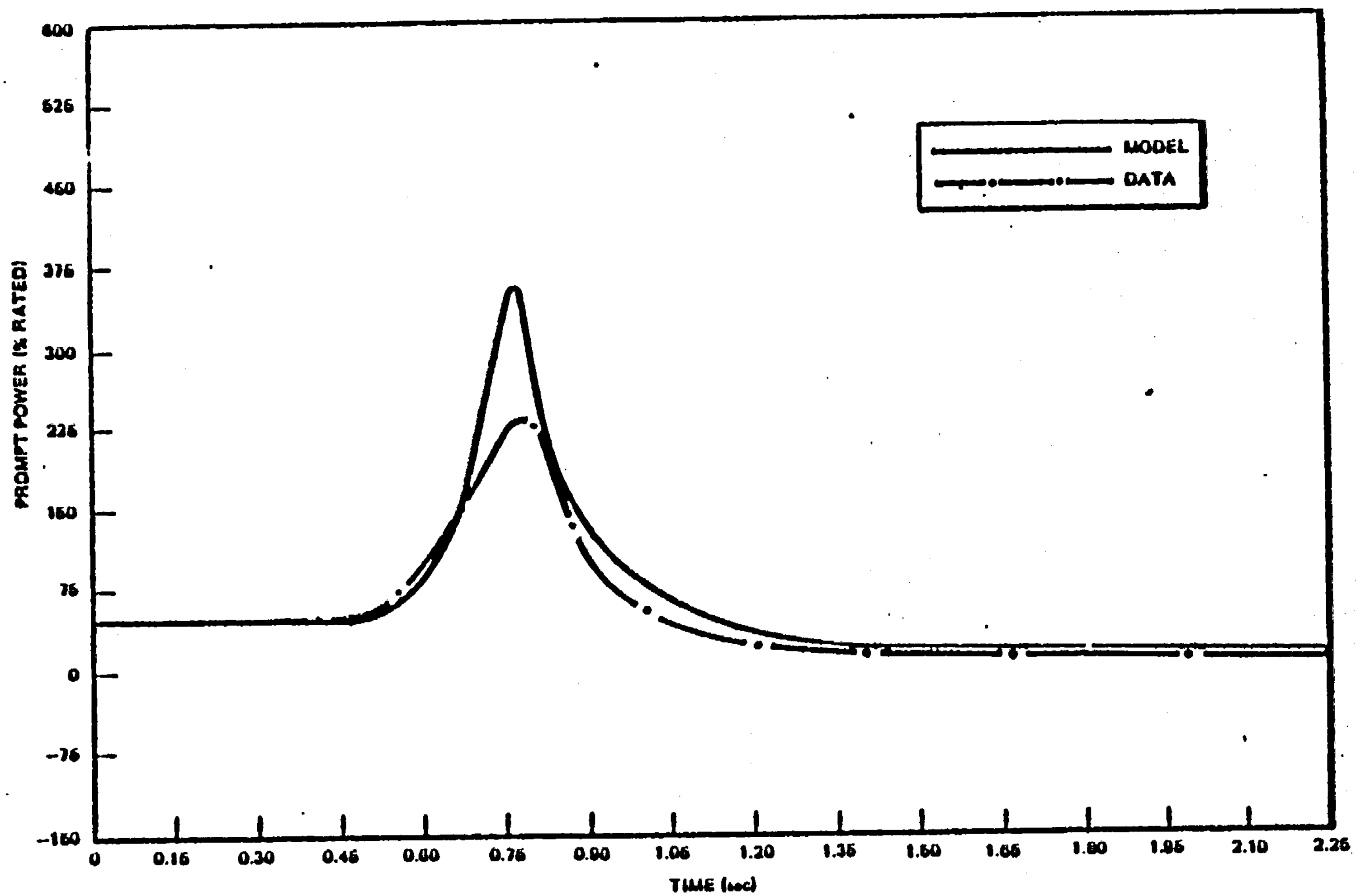

Figure 4 Peach Bottom-2 Turbine Trip 1 Prompt Neutron Power

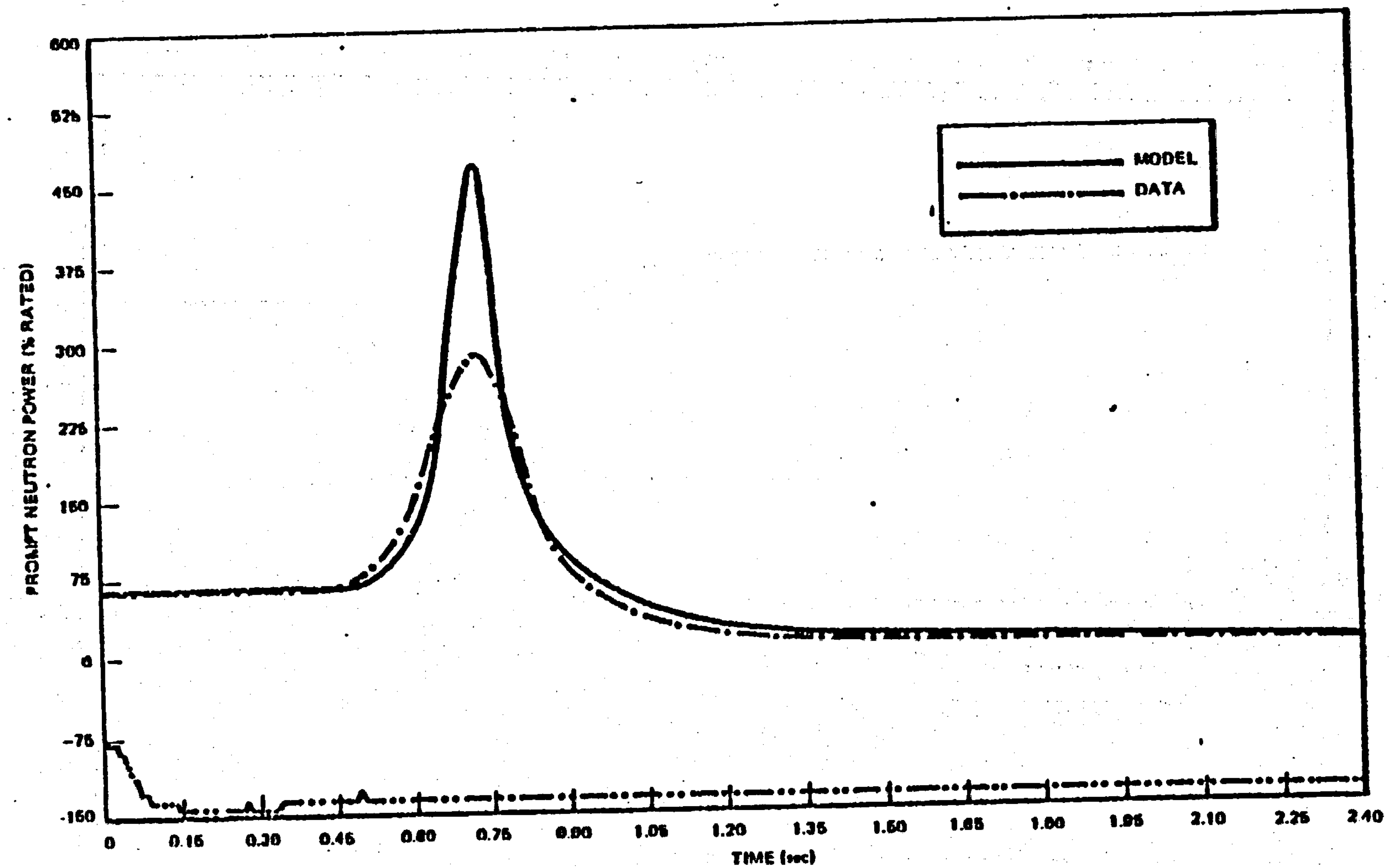

Figure 5 Peach Bottom-2 Turbine Trip 2 Prompt Neutron Power

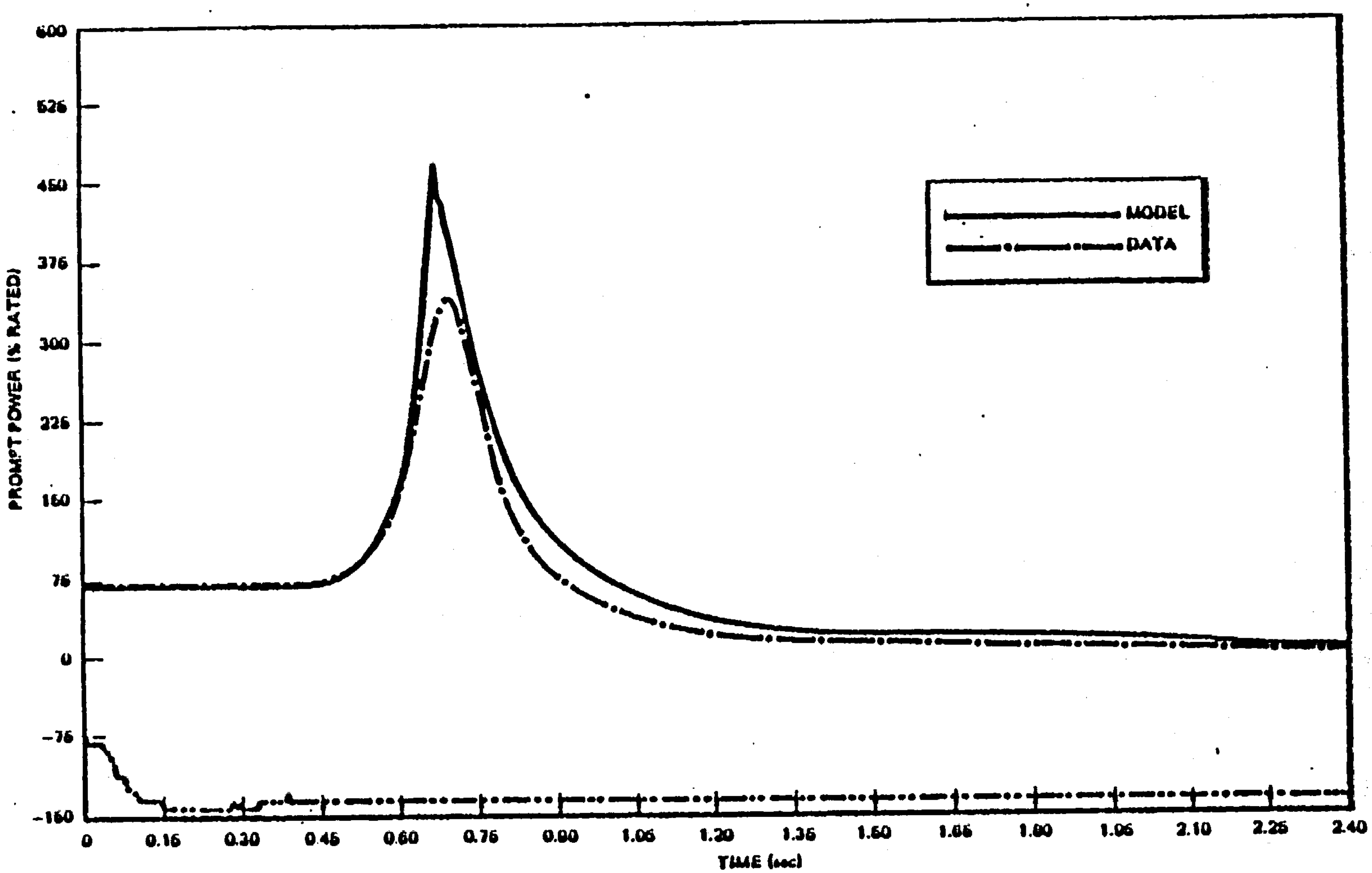

Figure 6 Peach Bottom-2 Turbine Trip 3 Prompt Neutron Power

evidence of this hypothesis, General Electric showed that the steamline pressure calculation for the KKM test, which had a finer spatial mesh, was quite accurate. General Electric has also pointed out that the steamline pressure response shape is not as important to the transient behavior as is the integrated value of the steamline pressure response.

We do not agree entirely with General Electric. In answer to Q-19 in Volume I, General Electric performed a sensitivity study showing the effect of nodalization (different mesh sizes) and comparing the results with the analytical model which uses method of characteristics. The difference in amplitudes in this comparison is on the order of 10% while the difference in amplitudes in Peach Bottom tests and ODYN predictions is 50%. In addition, expected differences have opposite trends. The accuracy claimed by General Electric in the KKM test can be due to the adjustment of the valve opening time. This adjustment was made by General Electric to obtain a better agreement with the measured pressure data. It appears that the steamline model does not predict the amplitudes of oscillations accurately. This is also substantiated by the staff audit calculations. However, we agree with General Electric that the integrated steamline pressure response is more important in determining the transient behavior than the amplitude of individual oscillations occurring at these frequencies. The Peach Bottom tests indicate that the dome pressures do not oscillate and this is the pressure to which the reactor is subjected. Comparison of the dome pressures indicate that the dome pressure calculations performed by the ODYN code are conservative relative to data; i.e., the overall rate of pressure rise as well as the magnitude of the calculated pressure were higher than the data indicate.

The initial dome pressure rise for the PB-2 tests was predicted accurately by ODYN. The calculation overpredicts the pressure rise near the peak of the first pressure oscillation, thus conservatively modeling void collapse for reactivity feedback. ODYN appears to overpredict the peak vessel pressure rise which demonstrates a conservative basis for overpressure protection analyses. General Electric states (Reference 23) that the overprediction is due to the assumption made in the energy equation for the dome region. The overprediction of dome pressure is considered a desirable conservatism. Within the period of time that the neutron flux pulse occurs, the dome pressure overprediction is approximately 16 to 28% higher than the data.

Reviewing the steamline and dome pressure transients and based on the sensitivity studies performed by General Electric, we require that the steamline be modeled by at least 8 nodes with maximum size of 100 ft for a node.

The core exit pressure is one of the most important parameters for the prediction of the pressurization transient neutron flux response. As was the case in dome pressure comparisons, the initial rise in core exit pressure was followed well by the model for all three tests. From the comparisons of ODYN to the PB-2 test results, General Electric has concluded that the steamline dome and vessel thermal hydraulic models simulate the overall core pressure rise rate well in all three experiments. This lends confidence to the code predictions through the full range of power levels. Measurements indicate some oscillations in core exit pressure. These oscillations have been attributed to instrument line effects by General Electric. This is corroborated by the lack of associated oscillations in neutron flux measurements.

The neutron flux predictions by the ODYN code were conservative relative to data. We estimate that the peak neutron flux is higher by 54 to 86% than the data and the integral of the nuclear power (which is a measure of the amount of energy generated) is also higher by approximately 16 to 42% than the data. Hence, the neutron fluxes were predicted conservatively in all three tests.

As a final step, General Electric has presented a calculation of ΔCPR/ICPR for test and model. We reviewed the calculational procedure and consider it appropriate. The results show that the ΔCPR/ICPR for ODYN predicted transient conditions is within 0.01 of the values which would be predicted from test conditions; i.e., the ΔCPR/ICPR values calculated using the measured flow from jet pump ΔP measurements, the measured pressure and the measured power during the tests. The ODYN transient conditions predicted two out of three ΔCPR/ICPR values conservatively. The differences are between -5.1% and 6.8% relative to values calculated using the data (minus means nonconservative). The differences in these three test results in terms of ΔCPR/ICPR give $\mu = 1.14\%$ which represents a very slight conservatism for the mean and $\sigma = \pm 6.39\%$ for the standard deviation. Since the data (three points) are very limited, the results do not have a high degree of confidence. Table II presents these values of ΔCPR/ICPR.

TABLE II

COMPARISON OF MAXIMUM ΔCPR/ICPR VALUES FOR PEACH BOTTOM AND KKM TESTS

Test	Initial CPR	ΔCPR/ICPR (Data)	ΔCPR/ICPR (ODYN)
Peach Bottom			
Turbine trip 1	2.536	0.170	0.173
Turbine trip 2	2.115	0.136	0.129
Turbine trip 3	2.048	0.132	0.141
KKM			
Turbine trip	1.279	0.077	0.084

Review of the test results indicate that all model conservatisms claimed by General Electric such as conservatism in calculation of the steam dome pressures and neutron flux, conservatism in collapsing of 3-D core neutronics and thermal hydraulics, conservatism in the gap conductance input parameters and any other conservatism claimed in the computer model are either so small that it did not make any difference in calculating ΔCPR for these three tests or all of these claimed conservatisms are offset by an unidentified nonconservatism somewhere else, perhaps in calculation of flow. It is evident that the calculations of ΔCPR are not conservative for all of the tests.

They can only be regarded as best estimate or accurate predictions. Hence, based on the Peach Bottom tests we do not give any credit for the conservatism in the models used in the ODYN code. The code will be regarded as best estimate for ΔCPR calculations and any discrepancy between the test results and the code will be treated as an uncertainty or an error. Further tests would be needed to reduce these uncertainties.

b. <u>KKM Test Comparison</u>

A brief summary of the test conditions is contained in Volume II. KKM plant has an unusual configuration, in that, it has two turbines and two sets of steamlines with a reheater line in each steamline. It presents some special model considerations for ODYN simulation. A special version of ODYN was developed to simulate this configuration.

Also unique to this test comparison as opposed to the PB-2 comparison is the modeling of turbine stop valve and bypass valve actuations. Measured turbine stop valve and bypass valve positions between initial and end of actuation were not available for this transient. The stop valve behavior can be reasonably estimated from the opening-to closing time. However, the transient response is quite sensitive to the bypass valve behavior. The bypass valve opening speed of the ODYN model was adjusted until the calculated transient turbine inlet pressure agreed with measurement. This adjustment was made for only the initial bypass valve opening speed and, thereafter bypass valve position was controlled based on the plant control parameters. The remainder of the test modeling is similar to that of the PB-2 test comparison. The fuel rod gap heat transfer coefficient was selected to be 600 Btu/hr-ft^2-°F.

The turbine trip test conducted at KKM provided a reactor and operating state that was quite different from PB-2. The test at KKM corresponded to an end-of-cycle condition with all control rods fully withdrawn and with the reactor at 77 percent of full rated power and 86.5 percent of full rated core flow. The reactor itself is considerably smaller in size than PB-2 and has a somewhat different system including two turbo-generating units. The turbine trip test at KKM resulted in a milder transient than the tests at PB-2. The ODYN results compared to the test data showed the same general agreement as was observed for PB-2 for the response of the core power as a function of time. The calculated steamline pressure response for the KKM turbine trip appears to be in good agreement with measurement. The KKM comparisons appear to be in slightly better agreement with measurement than do the PB-2 comparisons. As previously discussed, the characteristics of the bypass valve were adjusted to give a good agreement with the measured steamline pressure.

The measurements of dome pressure showed some oscillations. Since these oscillations did not manifest themselves in the neutron flux measurements, they were attributed to instrument line disturbances and were not considered to be actual pressure oscillations. There were also oscillations in core exit pressure measurements. Similar oscillations were not observed in the PB-2 tests. These oscillations were also attributed to the instrument line disturbances since no oscillations were observed in neutron flux.

The calculated pressure responses pass through the data up to 1.8 seconds of the transient time. After 1.8 seconds the calculated pressure are higher than those measured. The calculated core exit pressure had a 40 millisecond delay behind the data. This was attributed to the modeling of steam separator inertia. We agree with General Electric that the overall shape of the core exit pressure response is duplicated well by the ODYN code. The agreement between the calculated and measured pressures in the dome and the steamline is also reasonably good. There is no conservatism in calculation of pressures up to 1.8 second of transient time.

The measurement of neutron flux indicates a double peak behavior. This double peak was attributed to an oscillation in core pressure which was thought to be enhanced by KKM bypass characteristics. The ODYN code overpredicted the initial neutron flux peak by approximately 53% and underpredicted the second peak. We estimated that the integral of the calculated nuclear power was higher by approximately 20% than the data. Figures 7 and 8 present the comparisons of measured and calculated axial neutron and prompt neutron fluxes respectively.

The calculated value of ΔCPR/ICPR was about 9.1% conservative relative to the value calculated using measured quantities (see Table II).

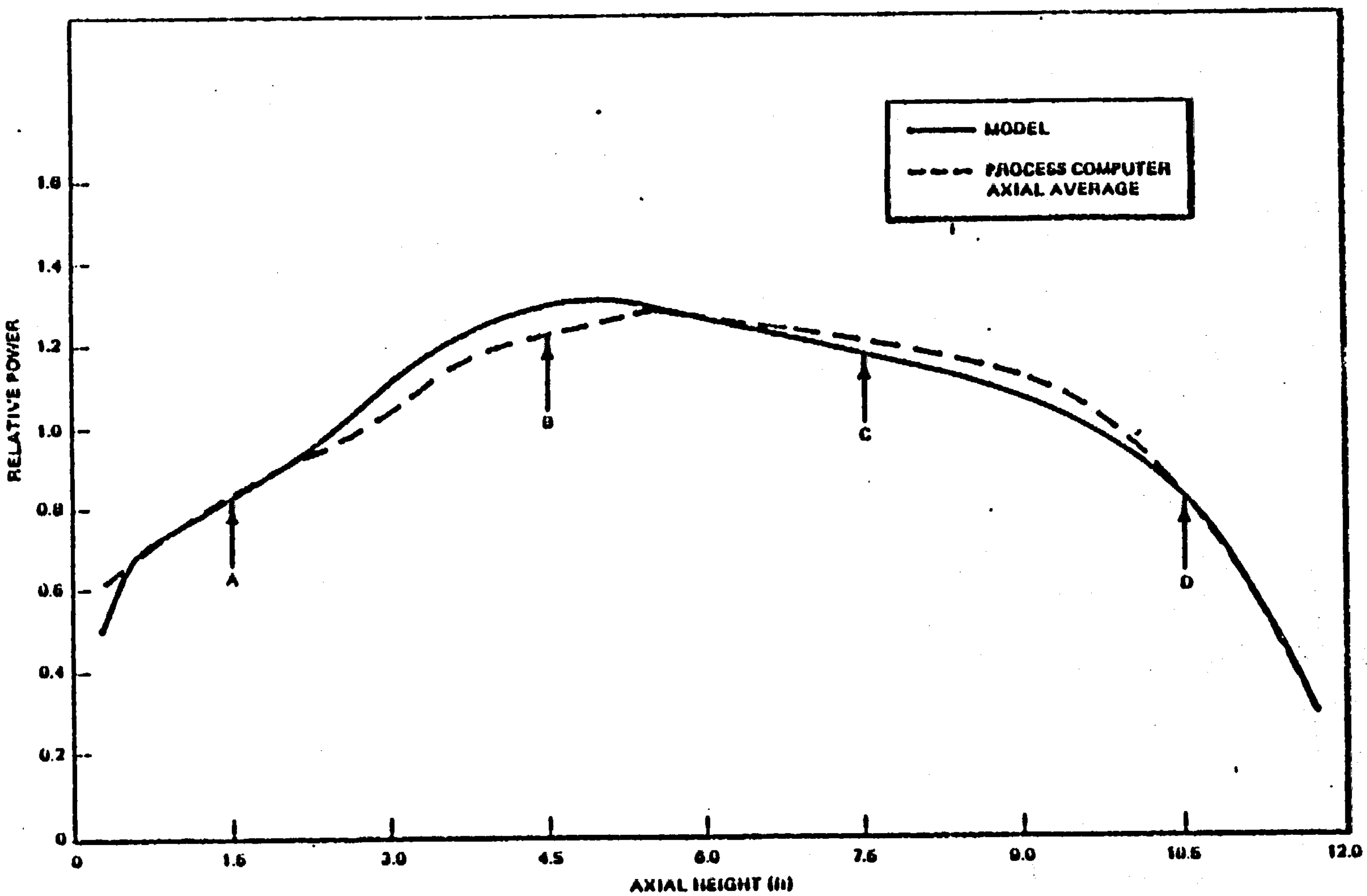

Figure 7 Average Axial Power KKM Test Conditions

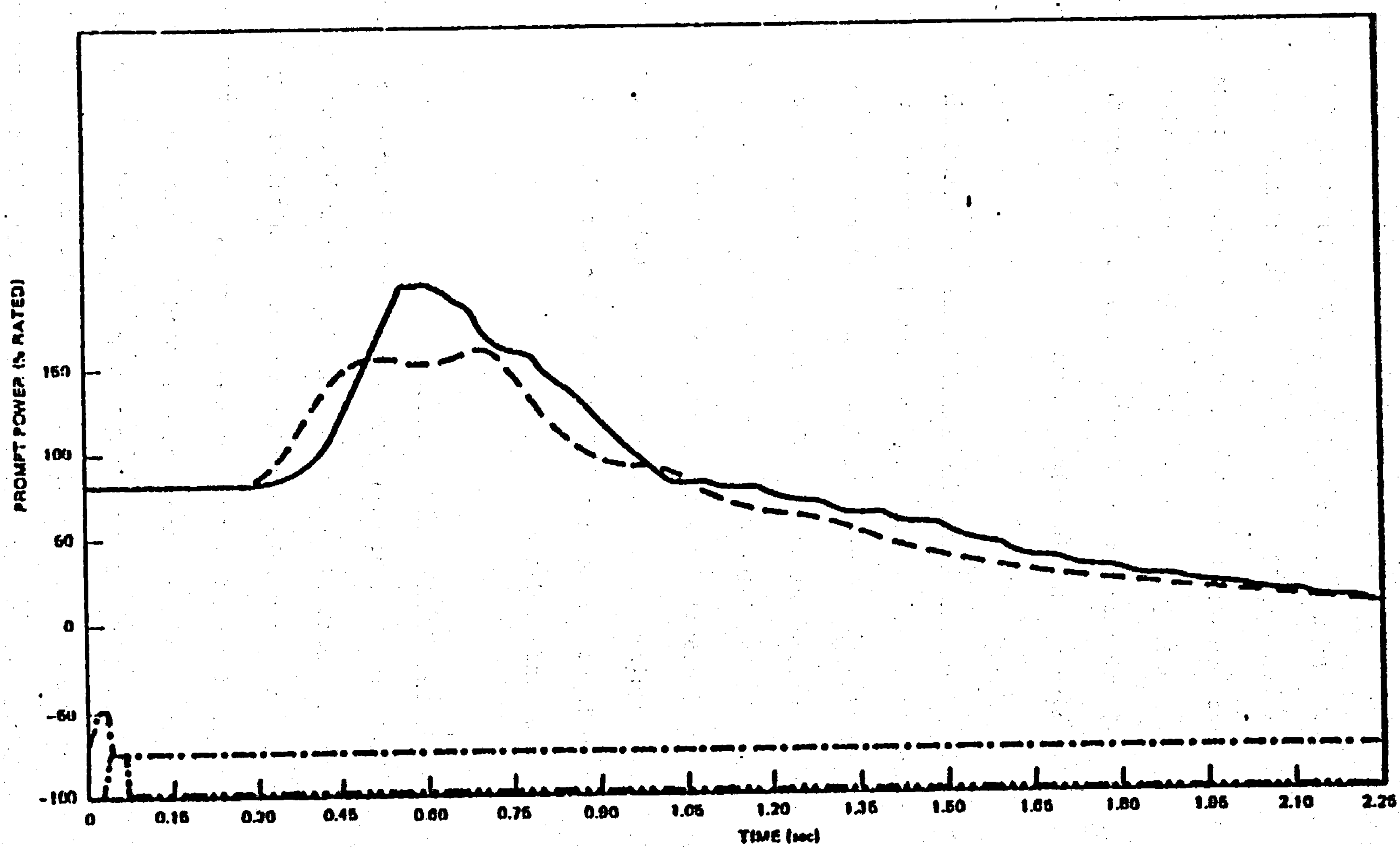

Figure 8 KKM Turbine Trip Prompt Neutron Power

Although there is conservatism of 9.1% in ΔCPR/ICPR, the difference in absolute values is small; i.e., 0.007 in terms of ΔCPR/ICPR. Since the value of ICPR was 1.279, the value of ΔCPR is approximately 0.01. According to General Electric the maximum practical accuracy in ΔCPR is 0.01 (Volume III). In addition, KKM transient is a relatively mild transient. Hence, we do not give any credit for conservatism in ΔCPR/ICPR prediction.

4. Qualification Using Another Computer Code - Audit Calculations

Another important means of qualifying a code is to compare the results of calculations with the results obtained from another code. The two codes should be as independent as possible including the neutronic input parameters. The BNL-TWIGL (Reference 24) and RELAP-3B (Reference 25) codes are fully capable of analyzing these BWR turbine trip tests and satisfy the requirement of independence. The nuclear data base for deriving the input for the BNL-TWIGL/RELAP-3B codes also satisifies the requirement of independence.

a. Development of Calculational Method

A calculational method for the analysis of the turbine trip transients was developed at BNL using the RELAP3B and BNL-TWIGL computer codes. This method was developed under two NRC Technical Assistance Programs supplementing each other, and uses the codes in an iterative manner. The details of the method are presented in Reference 26. The RELAP3B code is used to perform the system transient analysis for the audit calculations. The BNL-TWIGL code is used to calculate the reactivity feedbacks and core power transient. The BNL-TWIGL code performs a space-time analysis of core neutronics and thermal hydraulics with feedback in two dimensions (reference 21).

The BNL-TWIGL code has a number of advantages over the ODYN code. The calculation can be performed with two neutron energy groups in two-dimensional (r,z) cylindrical geometry. It has the capability of allowing for five radial scram zones. Any important radial effects will, therefore, be calculated by BNL-TWIGL. The BNL-TWIGL code also has two disadvantages relative to the ODYN code. These disadvantages are: (1) the lack of a bypass flow channel, and (2) the independence of the Doppler reactivity with void fraction. Weighing these advantages and disadvantages of BNL-TWIGL relative to the ODYN code, it is our judgment that they will not adversely affect the comparison of the two codes for the turbine trip transient discussed herein.

The calculational method was developed using the Peach Bottom tests as a bench mark. Assuming the measured power history (power vs. time) in the core as input, RELAP3B calculates the system thermal-hydraulic parameters and provides the BNL-TWIGL code with the time dependent core inlet boundary conditions, i.e., pressure, flow and temperature variations with time. Then, the BNL-TWIGL code performs the space-time analysis of the core neutronics and thermal-hydraulics. The calculated power history is then compared with the measured power which was input to the RELAP3B code. If the differences are large the calculated power history is used in the RELAP3B code and the calculations are repeated until the power history calculated by the BNL-TWIGL code is in good agreement with the power history input to the RELAP3B code. This method was used for both the Peach Bottom tests and the licensing audit calculations (turbine trip without bypass transient).

b. <u>Peach Bottom Tests and Audit Calculations</u>

The RELAP3B/BNL-TWIGL calculational method described above was employed by BNL to analyze the Peach Bottom transient tests. The calculated power history agreed well with the measured power history. There is also good agreement between the other calculated and measured parameters. The core physics results that were obtained by BNL are presented in Reference 26. Reference 26, also discusses the geometric modeling, the neutron cross sections, and the initialization of the transients. These calculations confirmed the adequacy of the BNL-TWIGL/RELAP-3B modeling for computing BWR turbine trip transients. Figures 9 through 14 show samples of these agreements. "Revised BNL" curves in Figures 12 through 14 refer to a more detailed BNL model which will be explained subsequently.

Calculations performed with the ODYN code also agreed very well with the experimental results although the neutron flux predictions were slightly conservative. We believe that this can be attributed to the slightly higher ODYN pressure predictions during the transient. The power history calculated by BNL provides better agreement with the experimental data on a best estimate basis than the ODYN code predictions.

Peach Bottom 2 Turbine Trip Test 1, Initial Power

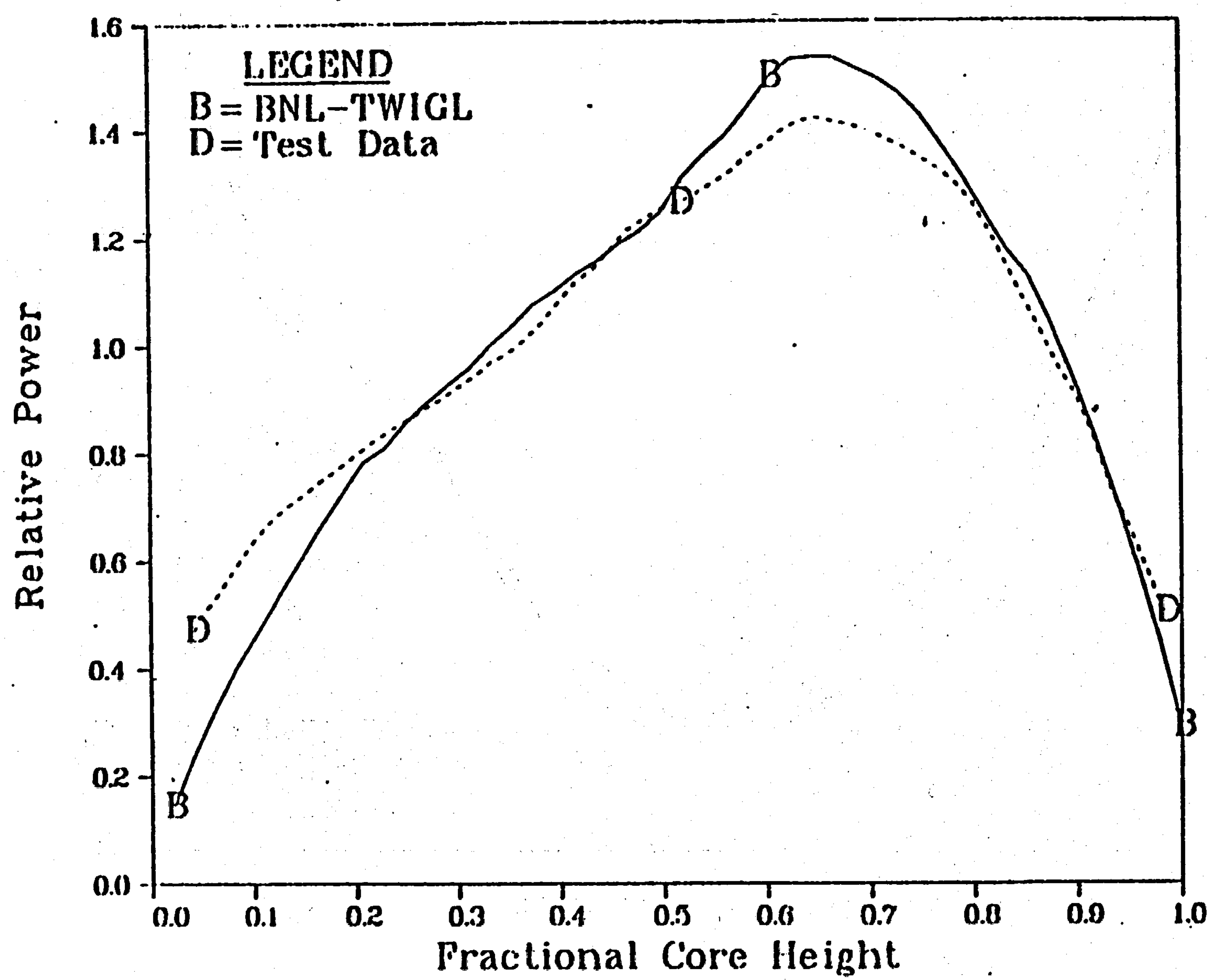

Figure 9

Peach Bottom 2 Turbine Trip Test 2, Initial Power

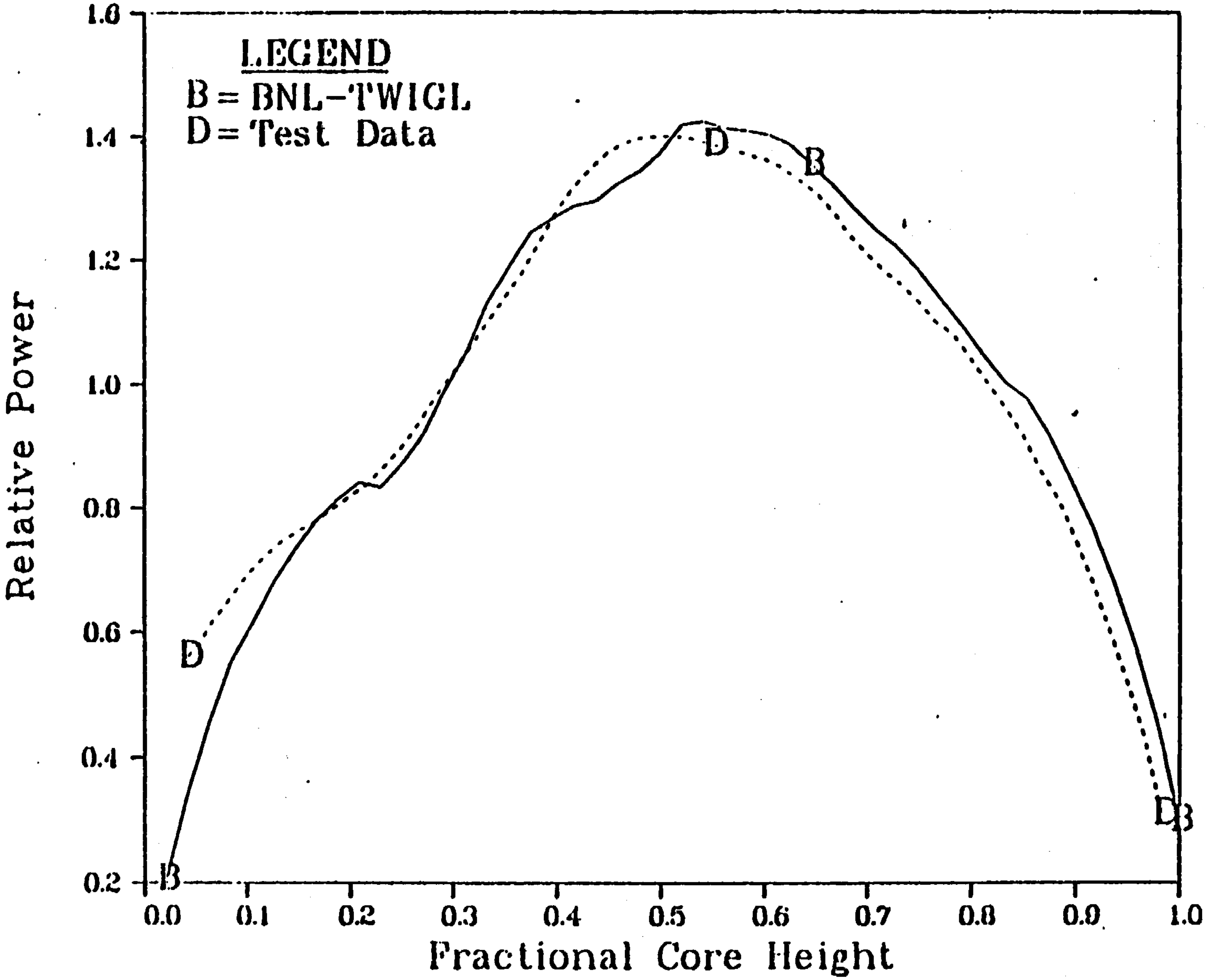

Figure 10

Peach Bottom 2 Turbine Trip Test 3, Initial Power

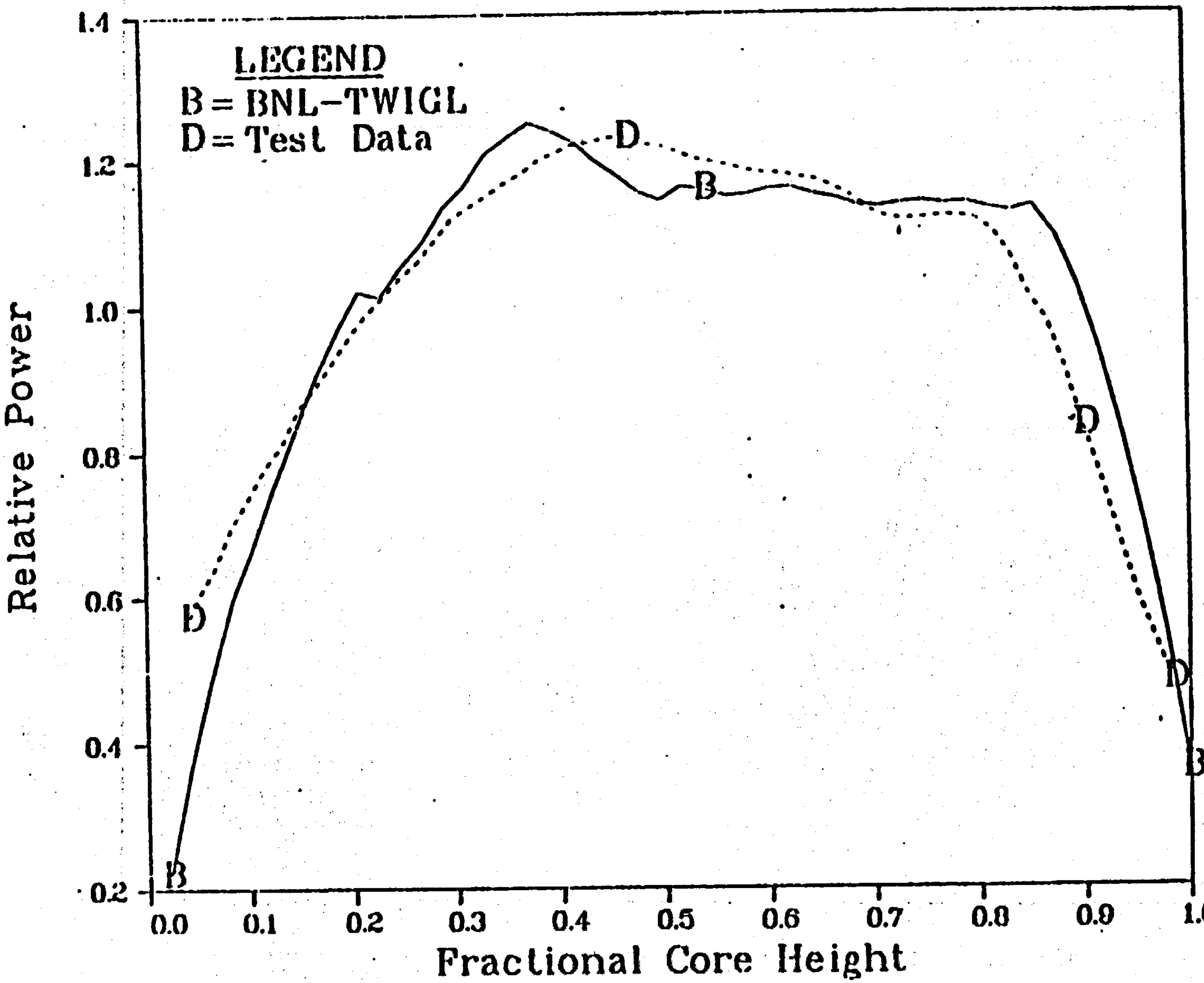

Figure II-63

Peach Bottom 2, Turbine Trip, Test 1

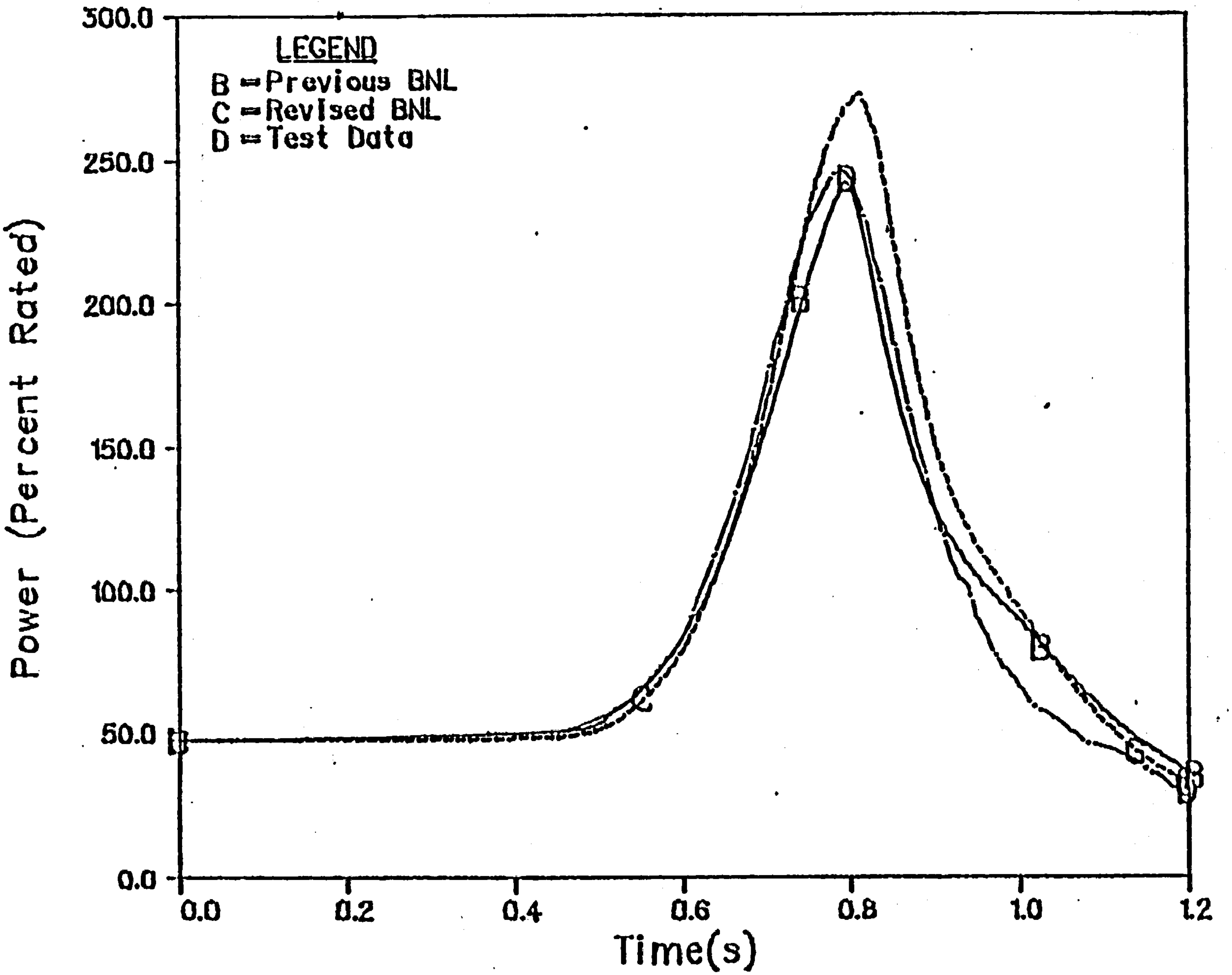

Figure 12

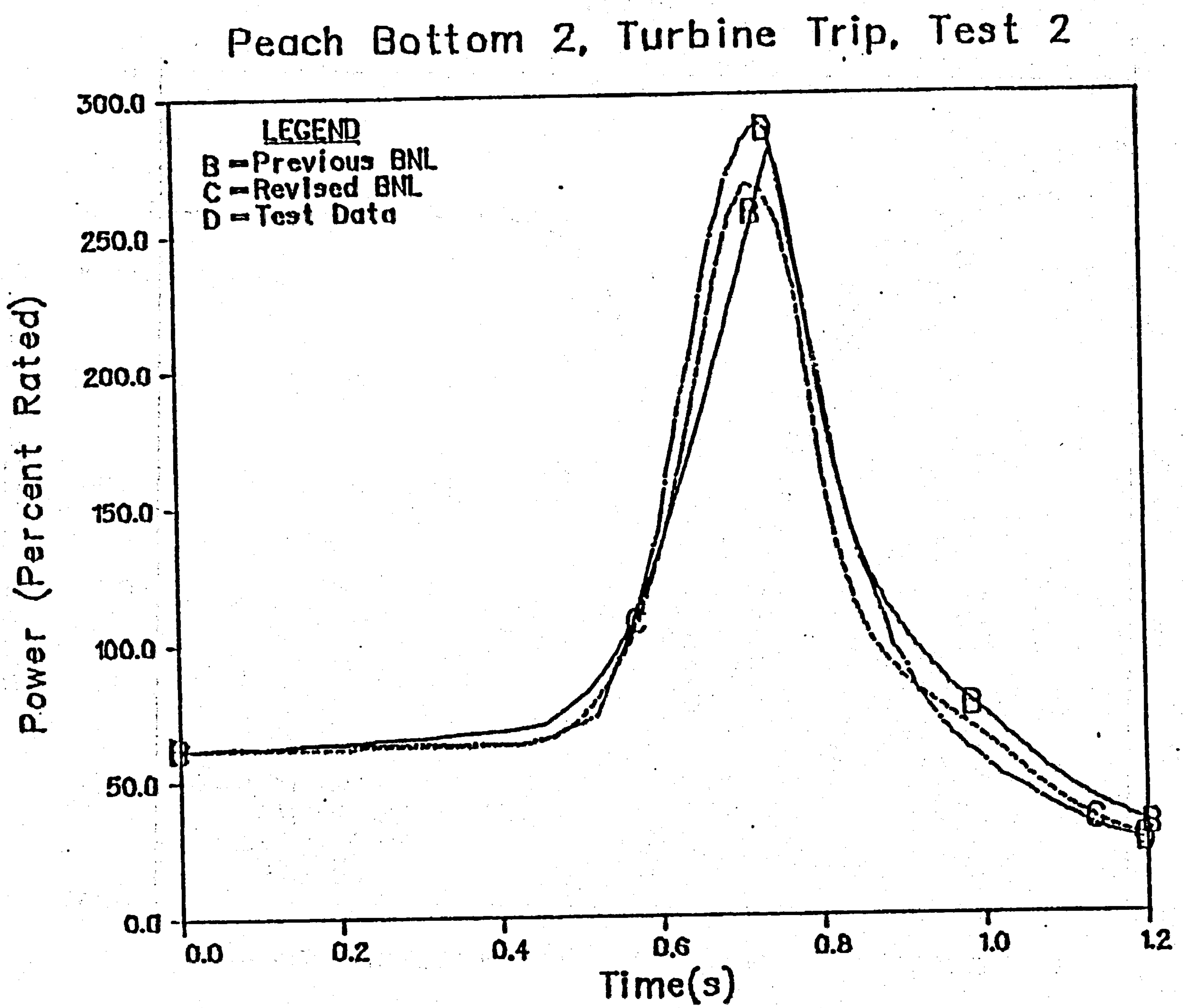
Peach Bottom 2, Turbine Trip, Test 2
LEGEND
B = Previous BNL
C = Revised BNL
D = Test Data
Power (Percent Rated)
300.0
250.0
200.0
150.0
100.0
50.0
0.0
0.0
0.2
0.4
0.6
0.8
1.0
1.2
Time(s)

Figure 13

Peach Bottom 2, Turbine Trip, Test 3

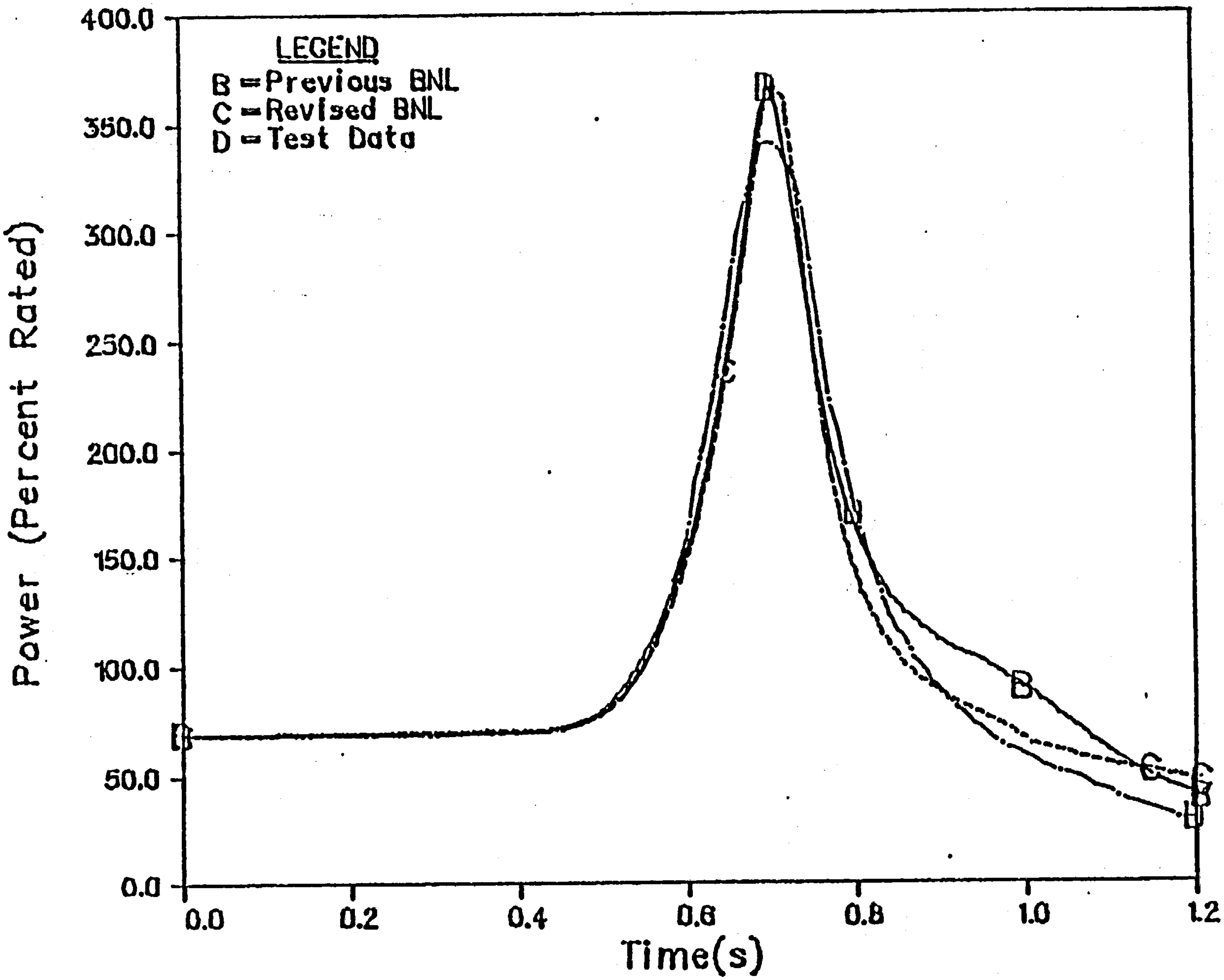

Figure 14

At a meeting on July 14, 1978 attended by GE and our consultants from BNL, a turbine trip without bypass transient (TTWOB) was defined for calculation by GE with the ODYN code and by BNL with the BNL-TWIGL/RELAP-3B codes. This TTWOB transient was for PB-2 at end-of-cycle 2 with the reactor at an all rods out condition and with a Haling core power distribution. The reactor trip was assumed to occur from the primary trip signal for this transient, i.e., the position of the turbine stop or control valve. All of the system input parameters were discussed and values were assigned. The reactor was assumed to be operating at a 104.5 percent of full rated power and at 100 percent of rated core flow.

The initial calculations by GE and BNL differed considerably. The total core power as a function of time calculated by BNL was about 60 percent greater in energy output although the initial rise and falloff of the power was about the same. The BNL calculation predicted a peak power of over 7 times the initial core power at about 0.9 seconds. The GE calculation resulted in a peak power of about 4 times the initial core power at about 1.0 second.

A GE evaluation of its calculation resulted in finding two significant errors that led to a new GE calculation. One of the errors was the steamline length. It had originally been input as 460 feet whereas the value should have been 400 feet. GE also found that one of its processing codes had improperly accounted for the Doppler reactivity feedback variation with void fraction. This new GE calculation resulted in a more severe transient than the earlier calculation. The new GE prediction of peak power was about 5.3 times the initial core power at about 0.9 seconds. This GE calculation had an earlier rise and an earlier fall-off of the total core power than the BNL calculation. The BNL calculation still predicted a greater energy output for the transient by about 20-25 percent. A comparison of the reactivity components showed that the void, scram, Doppler, and net reactivities as a function of time differed significantly between the BNL and new GE calculation. As an example, the BNL calculation resulted in a prompt critical calculation with a maximum net reactivity of over one dollar. The GE calculation resulted in a maximum net reactivity somewhat less than 0.8 dollars.

Since it was our expection that the BNL and GE calculations would be in better agreement, a meeting with General Electric at BNL was held to resolve differences between the calculations. This meeting was held from September 27 through September 29, 1978 with GE, BNL, and NRC in attendance, Reference 27. The main differences noted between the two calculations are listed below.

ITEM	REMARK
1. Relief valves	
(a) Set Points	GE values too large
(b) Delay time	BNL did not include
(c) Bank capacities	BNL did not use GE values
(d) Time constant for full flow	BNL did not include
2. Turbine inlet pressure	GE value too large
3. Steam separator modeling	
(a) Separator L/A	BNL value low
(b) Separator mass	BNL water inventory too low

ITEM (Cont.)	REMARK (Cont.)
4. Doppler reactivity feedback	BNL does not include variation with void fraction
5. Bypass heating effects	BNL does not include
6. Fuel gap conductance	BNL used variable value (400-500) whereas GE used a constant value of 1000

In addition, some of the BNL neutronic data were also compared to corresponding GE data. The beginning-of-cycle infinite multiplication factor (K_∞) was compared for a number of fuel types both with and without a control blade and as a function of void fraction. Only small differences were noted between the various sets of data. The initial axial K_∞ distribution was also compared and again only small differences were noted. It was also noted that the BNL control worth was about 15 percent larger than that of GE. These neutron cross section and K_∞ data as a function of fuel exposure and void fraction were later provided to the staff by GE for the two dominant fuel types in the PB-2 core (References 28 and 29).

Void reactivity coefficients extracted from the BNL and GE TTWOB calculations indicate that the BNL value at the start of the transient is larger in magnitude by about 14 percent than the GE value. This larger BNL void coefficient is consistent with the lattice physics data that is used. However, the neutron effective void correlation used by General Electric compensates for a large part of this difference (Reference 34).

A reanalysis of the turbine trip without bypass transient was performed by BNL. Both analyses were presented in Reference 33.

Some sensitivity studies were also performed. Some of the differences noted in the meeting did not change the BNL results substantially. However the difference in the separator inventories and the required renodalization around the separators to accommodate proper inventories inside and outside of the separators and inclusion of a separate flow path between the steam dome and outside of the separators (bulkwater), reduced the neutron power and the total energy output in the licensing basis transient (TTWOB). The total energy output, the integral of the neutron power, predicted by BNL during the transient was less than that in the second calculation performed by General Electric. However, the General Electric calculation still indicated an earlier rise and an earlier fall-off of the total core power than the BNL calculation. Figure 15 presents these two BNL calculations as well as the General Electric calculation.

The primary reason for the change in the energy outputs as well as disagreement in the shape of the neutron power transient was the new inlet core flow calculation by the BNL. The core inlet flow in the second BNL calculation was in closer agreement with the second GE calculation in that it exhibited similar oscillatory behavior. However, there were still some differences in amplitudes. It should be noted that the differences in flow variation during the transient between the two BNL calculations were within 15% of each other. Judging from the BNL studies, Reference 33, we conclude that the modeling of separators is significant in predicting the core inlet flow. We also note that the BNL modeling of the separators is still deficient in that the inertia term, L/A, does not depend on the

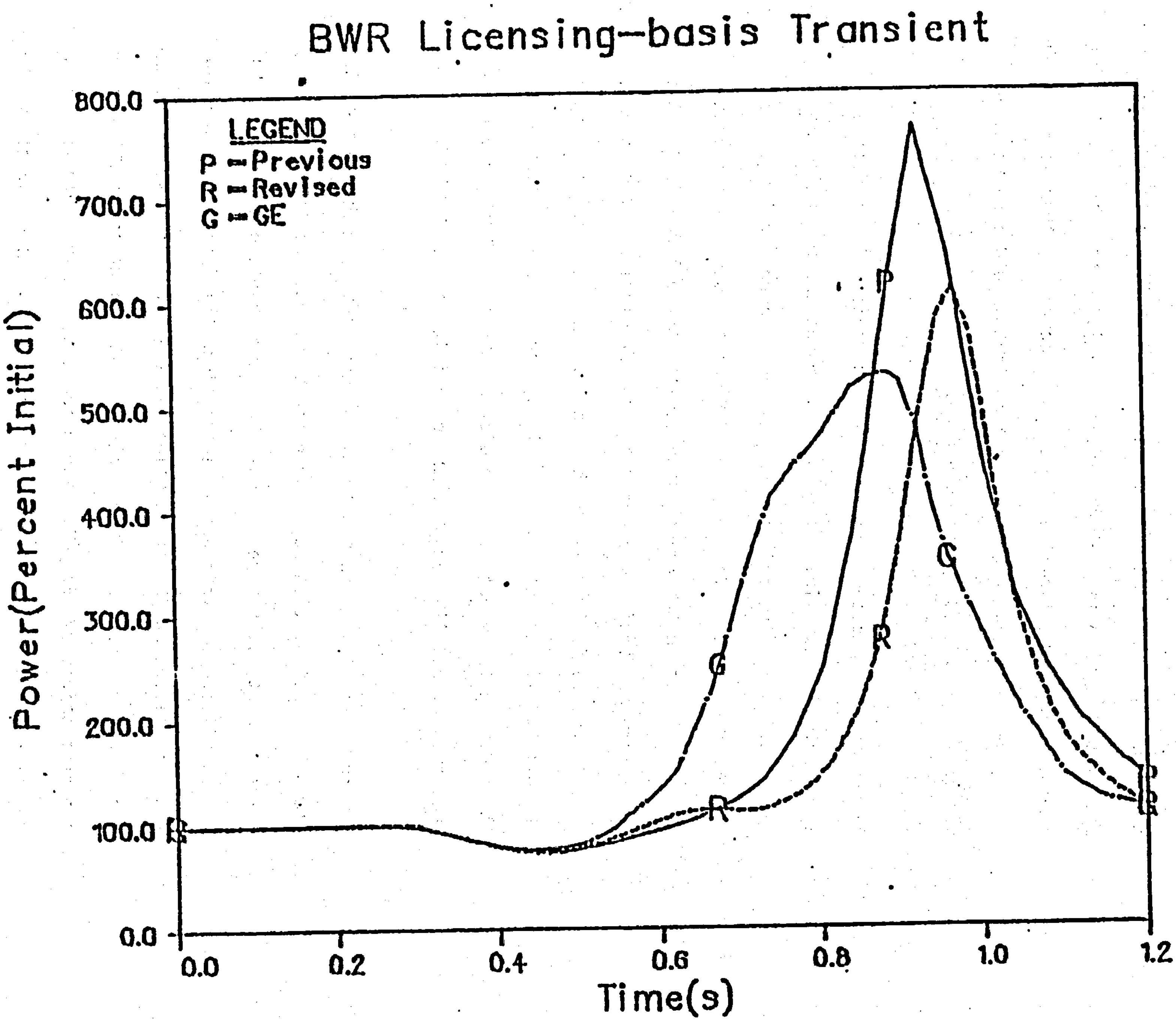

Figure 15

II-71

quality at the entrance of the separators. The ODYN code contains the modeling of the L/A term which derived its basis from experimental data obtained from separators. Hence, we judge that the ODYN model should predict the core inlet flow more accurately than does the BNL model, and that the neutron power transient should also be more accurately predicted by the ODYN code.

Reference 33 indicates that previous and revised BNL models predict almost similar core inlet flow variations at the beginning of the Peach Bottom turbine trip tests. However, there are large differences between predictions after 0.8 sec. Since the power peaks occur before 0.8 sec in the Peach Bottom tests, these large differences in inlet flow predictions do not alter the predictions of neutron powers as illustrated in Figures 12 through 14. However, as shown in the analysis of licensing basis transient, the separator inertia term and its modeling is important in predicting the transient behavior (or amplitude of oscillations) of the core inlet flow.

Reference 33 also indicated that the heat flux predictions in the second BNL calculation were lower for the portion of the transient where highest ΔCPRs were expected to occur than those calculated by General Electric. This was expected since the integral of neutron flux in the second BNL calculation was smaller than that calculated by General Electric. BNL did not perform ΔCPR calculations. However, the predictions of heat flux would suggest, and we would expect that the second BNL calculations would produce less ΔCPR than that calculated by General Electric.

BNL did not report heat fluxes in their first calculations. However, the integral of the neutron flux was almost the same as that calculated by General Electric. Hence, we expect similar severity for ΔCPR values if ΔCPR calculations were made using the first calculations performed by BNL.

A third analysis of the TTWOB transient was performed by BNL using GE calculated values of the core exit pressure and core inlet flow. The BNL-TWIGL calculation now predicted a transient with similar initial power rise and fall-off characteristics as the GE ODYN calculation although the peak power was higher in the BNL calculation. A sensitivity calculation with a 5 percent change in the void reactivity feedback resulted in a BNL-TWIGL power transient that compared very well with the corresponding GE results. The change of 5% in void reactivity or the coefficient is well within the calculated uncertainty of 11% as presented in Section II.A.6.

These audit calculations established the fact that core inlet flow is a very sensitive parameter. The core inlet flow measurements (i.e., the jet pump Δp measurements) in the Peach Bottom tests contained some errors. In Section II.B.3.a, where qualification of the ODYN code using Peach Bottom tests was evaluated, many comparisons between various parameters (such as pressure and neutron flux) were made and these parameters were always found to be conservative relative to data. However, despite these conservative features, the calculated values of ΔCPR cannot be considered as conservative. This was also pointed out in Section II.B.3.a. Based on the information submitted, it is our judgment that there is some nonconservatism in core inlet flow calculation in the ODYN code overcoming all other conservatisms.

We conclude that, although a precise audit of the GE ODYN code was not obtained by the BNL-TWIGL/RELAP-3B codes for this licensing basis TTWOB transient, the analyses performed by BNL provided us with valuable insights concerning this transient. We conclude that the primary reason for the disagreement of the BNL-TWIGL/RELAP-3B TTWOB results with the GE results is due to core inlet flow differences. Judging from the audit calculations as well as Peach Bottom test results, we also conclude that differences in predictions on the order of 20% in prediction of peak neutron flux can be expected using different computer codes which represent the state of the art.

5. <u>Summary of Code Qualification</u>

In summary, we find that the ODYN is a best estimate code containing models developed from first principles and provides good predictions of existing experimental data. The experimental data were obtained from separate effects and integral plant tests. The separate effects tests include core power measurements from various plants and heated tube tests to verify the void fraction model. Integral plant tests were performed at Peach Bottom Unit 2 and KKM. Comparison of the test data and calculations indicates that the agreement is within the uncertainties calculated in Section A. We find that the ΔCPR predictions from the ODYN and SCAT codes are neither conservative nor nonconservative. They predict the available data well.

The ODYN-SCAT prediction of the three Peach Bottom transient tests and one KKM transient test demonstrated a 2σ uncertainty of approximately 37% of ΔCPR/ICPR at a 95% confidence level. We have determined this using χ^2 distribution. No credit was given for measurement errors. This results in a 2σ ΔCPR/ICPR uncertainty of 0.068 for a transient which degrades the CPR from an initial value of 1.30 to the limit of 1.06. Since these tests represent a very limited data base, it is likely that the 2σ uncertainty can be reduced significantly by the acquisition of additional test data for comparison to code predictions. Hence, we recommend that additional integral plant tests be performed to qualify the code with a higher confidence.

C. EVALUATION OF THE MARGIN

The ODYN statistical analysis was performed by General Electric at our request in order to provide a quantitive basis for determining if the ODYN licensing basis contains an acceptable level of conservatism. Two quantities were calculated in this analysis: the probability of the expected ΔCPR exceeding the licensing basis ΔCPR; and the probability of exceeding the thermal-hydraulic design basis (i.e., probability of exceeding 0.1% of fuel in Boiling Transition).

The ODYN Code is intended to be used to calculate the change in Critical Power Ratio (CPR) during rapid pressurization transients such as the loss of load and feedwater controller failure transients. This information is used in combination with the General Electric Thermal Analysis Basis (GETAB) CPR safety limit to establish the operating limit CPR. GETAB is a statistical analysis which determines the value of CPR which corresponds to 0.1% of the fuel rods in Boiling Transition. The GETAB analysis considers the effects of uncertaint es on input parameters such as power, coolant temperature and flow as well as uncertainties in the GEXL correlation.

Uncertainties in the ODYN Code need to be considered since these will affect the probability of exceeding the thermal-hydraulic design basis. One method of accounting for the effects of ODYN code uncertainties is to include uncertainties directly into the GETAB statistical analysis. A second method is to assure that the ODYN licensing calculation gives a sufficiently conservative value of ΔCPR to assure that the thermal-hydraulic design basis is not exceeded. GE has chosen to use the second method in demonstrating the acceptability of the ODYN licensing basis.

We have determined that this approach is acceptable in principle. In addition, we have determined that an acceptable level of conservatism for the ODYN licensing basis corresponds to a 5% probability of exceeding the thermal-hydraulic design basis.

General Electric has provided statistical analyses of the loss of load (Turbine Trip) and feedwater controller failure transients. These analyses use Monte Carlo calculations to predict ΔCPR with a second order response surface which simulates ODYN calculations. The input parameters in the response surface are: initial power; control rod drive (CRD) speed; exposure index (a measure of axial power shape effects): ODYN code uncertainty; and response surface fitting uncertainties.

The response surface was generated through a regression analysis of ΔCPR calculations performed using input from the ODYN Code. The accuracy of the response surface was tested by General Electric by comparing the results of ODYN calculations to the results of response surface calculations. The accuracy checks were done for 15 load rejection transients for a BWR/3 EOC-6 and for 15 load rejection transients for a BWR/4 EOC-4. These comparisons showed good agreement between the two methods. In addition, a regressional fitting error was developed from these comparisons and this fitting error was added to the response surface. This error was found to have a range of one standard deviation values of 0.0076 ΔCPR/ICPR to 0.0126 ΔCPR/ICPR depending on the plant type, the time in cycle, and the transient of interest. This range of errors is three to four times smaller than General Electric's estimate of the ODYN code uncertainty (0.031 ΔCPR/ICPR) or our estimate of (0.044 ΔCPR/ICPR) - see Table I. This indicates that the response surface is a faithful reproduction of the ODYN calculational results and that the response surface can be used to establish the effect of ODYN code uncertainties on the probability of exceeding the thermal-hydraulic design basis.

The distribution functions of each of the input variables (initial power, CRD speed, exposure index, and code uncertainty) were reviewed. The uncertainties of the ODYN code are discussed extensively in the code review section of this report and will not be repeated here. The uncertainty on initial power level used by GE was + 2%. We requested additional information to substantiate this value and were given extensive information on the various elements in the plant energy balance and the uncertainties associated with each of these elements. The elements of the energy balance were checked against the ASME standard for determining energy output from a nuclear plant, "ASME Performance Test Codes, Test Code for Nuclear Steam Supply Systems (PTC 32.1-1969)." In addition, the uncertainty values for each element were reviewed and found to be reasonable. We have concluded that the 2% uncertainty (at one standard deviation) is an acceptable value for power measurement uncertainty.

In support of the assumed distribution of CRD speeds, General Electric has provided the results of tests from 13 operating BWRs. The total data base includes 3,985 individual CRD scrams. The information was presented in considerable detail, including the mean values and standard deviations of the times to 5%, 20%, 50% and 90% insertion for various plant types and for full core and partial core scram tests. An extensive and convincing statistical analysis of the data was also presented. Each data set was tested to determine if it could be tested as part of a larger data set; and only those data set which were found to be statistically alike, at a high confidence level, were treated together. Statistical tests were performed by General Electric to determine the significance of: variations among BWR designs; variations between full core tests and partial core tests; variations among operating plants; variations among scram tests; and variations among individual drives within scram tests. We conclude that these CRD scram tests are indicative of past operating experience and that the mean values and standard deviations of CRD speed can be chosen for the statistical analysis of the ODYN code. However, we cannot conclude that these CRD scram tests will be indicative of future reactor scram speed performance. In addition, it is also necessary to demonstrate that the scram characteristics of an individual reactor to be licensed can be represented by the distribution used in the analysis. The scram characteristics of an individual reactor should belong to the same population. General Electric should provide an assurance or appropriate modifications to Technical Specifications to demonstrate that the scram characteristics indeed belong to the same population or can be represented by the same distribution. General Electric should also assess the impact of the use of best estimate distributions on providing this assurance.

The transient response to rapid overpressure events is dependent on the core average axial power distribution and axial exposure distribution since these strongly influence both the void a:d control rod reactivity feedback. General Electric has defined Exposure Index as a measure of the axial exposure distribution. Exposure Index indicates the extent to which an actual axial exposure distribution differs from the ideal, design axial exposure distribution (Haling distribution). ODYN licensing calculations use the Haling distribution as input. General Electric proposed to show the conservatism associated with this assumption by establishing that the axial exposure distributions actually encountered during operation are more favorable than the Haling distribution. This conservatism was quantified as part of the overall ODYN statistical analysis by including Exposure Index as one of the input variables in the response surface.

To establish a basis for the expected distribution of Exposure Indices, General Electric presented data from 11 operating reactors at end of cycle conditions and 15 data points for 5 operating reactors at mid-cycle conditions. In response to a request for additional data on observed Exposure Index, General Electric provided 8 additional data points. Because of the limited number of data points and the large scatter in the data we were led to question the assumption that the data was normally distributed. The individual data points obtained from General Electric were subjected to the W-test for normality by the NRC, Applied Statistics Branch. This test indicated that there was not a sufficient reason to reject the assumption that the data were normally distributed. Based on this information, the inclusion of Exposure Index in the statistical analysis as a normally distributed variable is acceptable. As in the case with the use of measured CRD speeds, the implications of using best estimate values of Exposure Index based on past operating experience and the associated need for assurance and modifications to Technical Specifications to demonstrate continued acceptable performance were not addressed. Since we cannot determine appropriate modifications necessary for demonstration of the conservatism due to inability to operate at Haling power shape for each reactor to be licensed, we find that the use of the variation of power shape from that of a Haling shape in the statistical analysis is not appropriate.

General Electric has performed the statistical analysis using several different sets of assumptions relative to the response surface input parameters. The probability of exceeding the ODYN licensing basis ΔCPR was calculated for each case. This corresponds to the probability of exceeding the GETAB CPR safety limit. The probability of exceeding the criteria of 0.1% of fuel rods in Boiling Transition was also calculated for some of these cases. Since General Electric has proposed that the safety limit for BWRs be based on the GETAB CPR safety limit, it is appropriate to use the probability of exceeding this value as the basis for accepting the proposed licensing method. As stated previously, we have determined that a 5% probability of exceeding the GETAB CPR limit is acceptable. Unless the safety limit for BWRs is redefined by General Electric and reevaluated by the staff, the use of the probability of exceeding 0.1% of fuel in boiling transition is not an appropriate basis for judging the acceptability of the ODYN licensing basis.

In conclusion, we recommend that General Electric reperform the statistical analysis to demonstrate the appropriateness of the margin to the GETAB limit. This statistical analysis should not take credit for conservatism in the Haling power distribution. It may take credit for distribution in scram speeds if General Electric demonstrates that the distribution used in the analysis is applicable to the plant specific case. The analysis should also be performed using the code uncertainties as revised by the staff (± 0.068 ΔCPR/ICPR) which was based on the plant test data. General Electric may wish to convolute additional variables in the statistical analysis if assurance for conservatism for each specific application is provided.

III. STAFF POSITION

We stated our position on the ODYN code and its application in Reference 35. The following is a statment of that position.

I. ΔCPR Calculations

The analysis for ΔCPR must be performed in accordance with either approach A or approach B.

A. ΔCPR Calculations with Margin Penalty

This approach is comprised of the three step calculation which follows:

1. Perform ΔCPR calculations using the ODYN and the improve.. SCAT (Reference 36) codes for the transients in Table III and using the input parameters in the manner proposed in pages 3-1 through 3-4 of NEDE-24154-P. The sensitive input parameters are listed in Table IV.

2. Determine ICPR (operating initial critical power ratio) by adding ΔCPR calculated in step 1 above to the GETAB safety limit. Calculate ΔCPR/ICPR.

3. Determine the new value of ICPR by adding 0.044 to the value of ΔCPR/ICPR calculated in step 2 above. Apply this margin to Chapter 15 analysis of the FSARs submitted for OLs, and CPs and to reloads.

The margin of 0.044 is obtained from consideration of uncertainties in components listed in Table I.

A sample calculation is presented below:

Step 1

Assume that ΔCPR calculations using the ODYN licensing basis have been performed and the result is

$$\Delta CPR_c = .14$$

where the subscript c refers to calculations.

Step 2

Calculate ICPR based on the calculations.

$$ICPR_c = 1.06 + .14 = 1.20$$

where the GETAB limit is 1.06.

$$\frac{\Delta CPR_c}{ICRP_c} = \frac{.14}{1.20} = .117$$

Step 3

$$\frac{\Delta CPR_{new}}{ICPR_{new}} = .117 + 0.044 = .161$$

$$\Delta CPR_{new} = \frac{\frac{\Delta CPR_{new}}{ICPR_{new}}}{\frac{\Delta CPR_c}{ICPR_c}} \quad \Delta CPR_c = \frac{.161}{.117} \quad .14 = .192$$

$$ICPR_{new} = 1.06 + .19 = 1.25$$

B. <u>Statistical Approach for Reduction of Margin Penalty</u>

General Electric assessed the probability of the ΔCPR during a limiting transient exceeding the ΔCPR calculated for the proposed licensing basis transient (NEDE-25154-P response to question 4). The General Electric study demonstrated that this probability, based on operating data over several fuel cycles from a group of plants, is very low. The key parameters in the study are scram speed, power level, power distribution, and an estimate of ODYN uncertainties. The proposed approach utilizes the conservatism inherent in the statistical deviation of the actual operating conditions from the limiting conditions assumed for the first three parameters in licensing basis calculations to compensate for potential non-conservatisms from the ODYN uncertainties.

The staff has concluded that the use of end-of-cycle power distributions from multi-cycles for several reactors to obtain credit for margin conservatisms relative to Haling power distribution is not appropriate. There is no assurance that the end-of-cycle power distribution conservatisms obtained from operating reactor history are representative of the end-of-cycle conditions which will exist for the specific core. We have also concluded that scram speed data used in the GE statistical assessment must be proved applicable to specific license and reload applications. In order to take credit for conservatism in the scram speed performance for reloads, it must be demonstrated that there is insufficient reason to reject the plant-specific scram speed as being within the distribution assumed in the statistical analysis. For CP and OL, the scram speed distribution for the specific plant must be demonstrated consistent with those used in the statistical approach. Similar design and prototypic performance characteristics coupled with appropriate technical specifications on scram speed performance could provide acceptable evidence of the applicability of the data base.

Statistical convolution of the power measurement uncertainties to take credit for full power operation at a power level value below that used in licensing calculations is acceptable to the staff. However, plant specific procedures to operate within the licensing limit must be taken into account in these calculations.

The code uncertainty penalty (0.044 in ΔCPR/ICPR) applied to the licensing calculations described in (A) does not account for unknown contributors. Past experience has shown that additional margin in safety calculations is often needed to compensate for unknown non-conservatisms in licensing calculations due to code errors or other factors. The ODYN prediction of three Peach Bottom transient tests and one KKI transient test demonstrated a 2σ uncertainty of approximately 37% of ΔCPR/ICPR at a 95% confidence level. This was determined using χ^2 distribution. No credit was given for measurement errors. This results in a 2σ ΔCPR/ICPR uncertainty of 0.068 for a transient which degrades the CPR from an initial value of 1.30 to the limit of 1.06. Since these tests represent a very limited data base, it is likely that the 2σ uncertainty can be reduced significantly by the acquisition of additional test data for comparison to code predictions. Therefore, the magnitude of the code uncertainty used in the statistical convolution may be reduced to a value consistent with the 2σ

value of ΔCPR/ICPR uncertainty at a 95% confidence level when such a reduction can be justified by additional transient test data.

In summary, the staff has concluded that the statistical approach to compensate for potential non-conversatisms from the ODYN uncertainties is acceptable with the following limitations.

1. Power distribution conservatisms should be excluded.

2. Scram speed conservatisms must be demonstrated to be applicable to plant specific cases.

3. Calculations should be performed using a code uncertainty value which is 37% of the ΔCPR/ICPR for a limiting transient to account for code uncertainties, including unknown contributors (e.g., code errors), based on the approved transient test data base. This results in a value of ± 0.068 in ΔCPR/ICPR uncertainty for a transient extending over a CPR range of 1.30 to 1.06.

4. The transient test data base must be expanded and submitted for staff review to justify any reduction in the value of ODYN Code uncertainty (2σ value of ΔCPR/ICPR at a 95% confidence level).

5. A new statistical analysis conforming with these limitations must be provided.

An acceptable licensing basis using the Option B statistical approach is a 95/95 ΔCPR/ICPR for the limiting event. This can be established in one of two ways:

a. Option B can be applied on a plant-specific basis - i.e., statistical analyses performed on a particular plant to determine its 95/95 ΔCRR/ICPR. The statistical analysis procedures to be used are those defined in the ODYN Licensing Topical Report (LTR), Volume 3, except for the modifications required by the NRC in Reference 35.

b. Option B can be applied on a generic basis. This involves the establishment of generic ΔCPR/ICPR adjustment factors for groupings of similar-type plants (the groupings used in the ODYN LTR are considered to be an acceptable matrix) which can then be applied to the plant-specific ΔCPR/ICPR calculations from the ODYN LTR deterministic approach to derive the estimated 95/95 values. Each plant group and transient type correction factor is based on an analysis of a typical plant in that group (e.g., BWR 2/3, 4/5, and 6), in which the differences between the 95/95 ΔCPR/ICPR calculated per the ODYN LTR deterministic approach is determined for a specific transient (e.g., load rejection without bypass). The difference, which may be positive or negative, is designated the plant group adjustment factor for that transient. The generic ΔCPR/ICPR adjustment factors established for the various plant groupings must be submitted to the NRC for review.

II. PRESSURE CALCULATIONS

Calculations should be performed for the Main Steam Isolation Value closure event with position switch scram failure using the values listed in Table I* as per staff evaluation to arrive at the overall code uncertainty in pressure calculation. Add this uncertainty to the ODYN calculated pressure for this event in OL, CP and reload applications. If General Electric can demonstrate that this uncertainty is very small (e.g., by a factor of 10 or more) relative to the bias in determining ASME Vessel Overpressure limit, no addition of uncertainty to the calculations of pressure is needed.

*We note that there is an error in Enclosure 2 of Reference 35. The bounding values of the drift flux parameters should have been in conformance with Table I as per staff evaluation.

TABLE III

TRANSIENTS TO BE ANALYZED USING THE ODYN CODE

A. For Thermal Limit Evaluation

Event	Thermally Limiting or Near Limiting (Typically)
1. Feedwater Controller Failure - Maximum Demand	X
2. Pressure Regulator Failure - Closed	
3. Generator Load Rejection	X
4. Turbine Trip	X
5. Main Steamline Isolation Valve Closures	
6. Loss of Condenser Vacuum	
7. Loss of Auxiliary Power - All Grid Connections	X

B. For ASME Vessel Overpressure Protection

	Pressure Limiting
1. MSIV Closure with Position Switch Scram Failure (i.e., MSIV Flux Scram)	X

TABLE IV

INPUT PARAMETERS SENSITIVE FOR THE ANALYSES

1. CRD scram speed - at tehcnical specification limit.
2. Scram setpoints - at technical specification limits.
3. Protection system logic delays - at equipment specification limits.
4. Relief valve capacities - minimum specified.
5. Relief valve setpoints and response - all valves at specified upper limits of setpoints and slowest specified response.
6. Pressure drop from vessel to relief valves - maximum value.
7. Steamline and vessel geometry - plant-unique values.
8. Initial power and steam flow - maximum plant capability.
9. Initial pressure and core flow - design values at maximum plant capability.
10. Core exposure/power distribution - consistent with Haling mode of operation.
11. Feedwater conditions - maximum temperature (maximum core average void content).

III. Other Limitations

1. Listing of important input variables such as listed in Table IV and initial plant parameters including but not limited to control system characteristics as depicted in Figures 4-13 through 4-16 of NEDO-24154, Vol. 1, but with numerical values provided should be provided with each submittal. The initial control system characteristics, including the model used in the selection of initial settings, shall be defined and substantiated in terms of the design basis for each control system of the plant. We understand that neutronic parameters which were originally obtained from the GE 3-D Core Simulator and collapsed to provide input to the ODYN code, are best estimate. If there is a significant change in this calculational method altering the input parameters, General Electric should submit the new procedure to NRC for its evaluation. The code uncertainty value of $\pm$ 0.068 ΔCPR/ICPR based on the Peach Bottom and KKM test data includes uncertainties in this calculational method since this method was used in comparison of test data with code predictions. Hence, any significant change in this procedure will change the code uncertainty.

2. A minimum of eight nodes should be used to represent the steam line. However, the maximum length of any node should not be more than 100 ft.

3. The code cannot predict accurately core inlet flow oscillations with frequencies above 5 Hz. Although we do not expect any inlet flow oscillation above frequency of 5 Hz for the transients listed in Table III, General Electric should verify that the harmonic components above 5 Hz are indeed very small if very rapid variations of flow in these transients are predicted.

4. The transients listed in Table III are short term licensing transients. If the code is intended to be used for long term transients or different types of overpressurization transients such as ATWS, appropriate modifications should be made.

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