UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 4 1981

(15)

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General Electric Company
175 Curtner Avenue
San Jose, California 95114

Dear Dr. Sherwood:

SUBJECT: ACCEPTANCE FOR REFERENCING GENERAL ELECTRIC LICENSING TOPICAL REPORT NEDO-24154/NEDE-24154P

The Nuclear Regulatory Commission has completed its review of the General Electric Company Licensing Topical Report NEDO-24154 Volumes I and II and NEDE-24154 Volume III entitled "Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors" and the supplemental information submitted by R. H. Buchholz, letter (MFN 155-80) dated September 5, 1980. This report describes the General Electric transient analysis code, ODYN. This code is to be used for transient analyses of the following eight transients:

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

Should Nuclear Regulatory Commission criteria or regulations change such that our conclusions as to the acceptability of the report are invalidated, General Electric and/or the applicants referencing the topical report will be expected to revise and resubmit their respective documentation or submit justification for the continued effective applicability of the topical report without revision of their respective documentation.

Sincerely,

Robert L. Tedesco, Assistant Director
for Licensing
Division of Licensing

Enclosure:
As Stated

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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June 1980

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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

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,

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 momentum 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

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

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

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.

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.

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,

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. <u>Core Thermal-Hydraulics Model</u>

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

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.

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

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.

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 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

The fuel and cladding conductivity and heat capacity are assumed to be temperature dependant. 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 independant 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

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 uncer-

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

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

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. Margin in Pressure Calculations

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.

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,

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

3. Qualification Using Integral Tests

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 (XXM) 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. Peach Bottom Tests

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

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

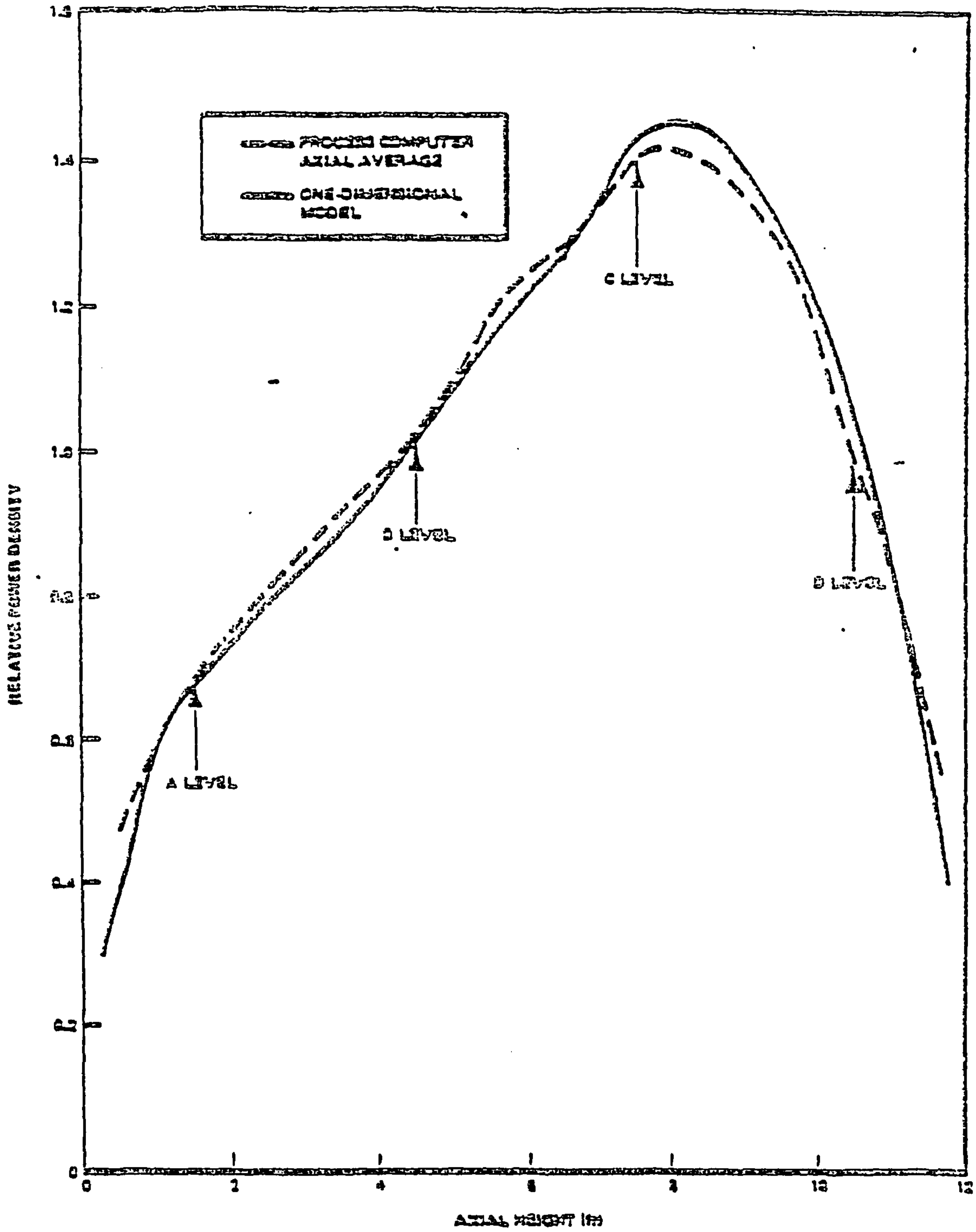

Figure 1 Axial Power Profile Turbine Trip 1

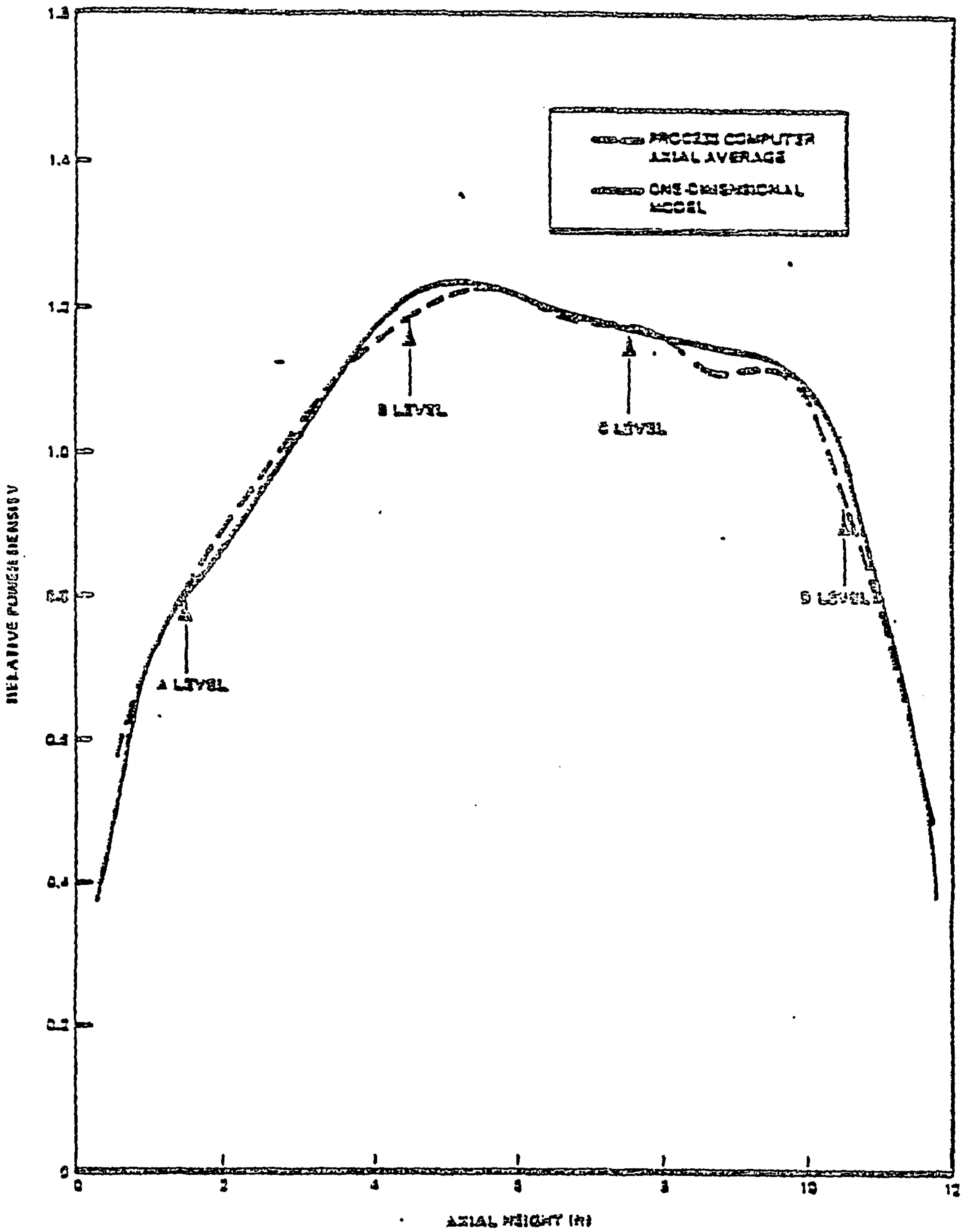

Figure 3 Axial Power Profile Turbine Trip 3

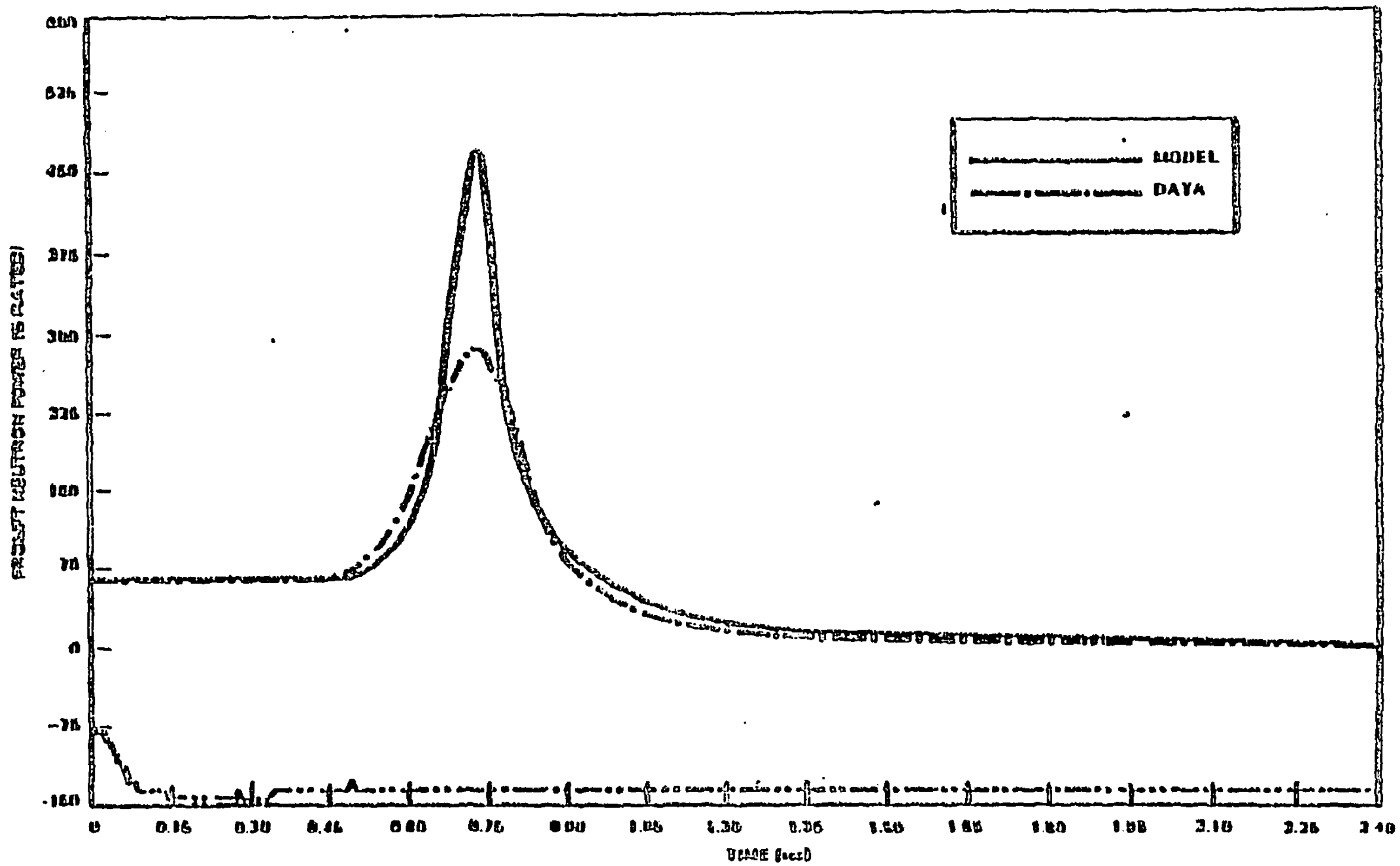

Figure 5 Peach Bottom-2 Turbine Trip 2 Prompt Neutron Power

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 30%. 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

General Electric has concluded that the steamline dome and vessel thermal hydraulic models simulate the overall core pressure rise rather 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 85% 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

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. KKM Test Comparison

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

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).

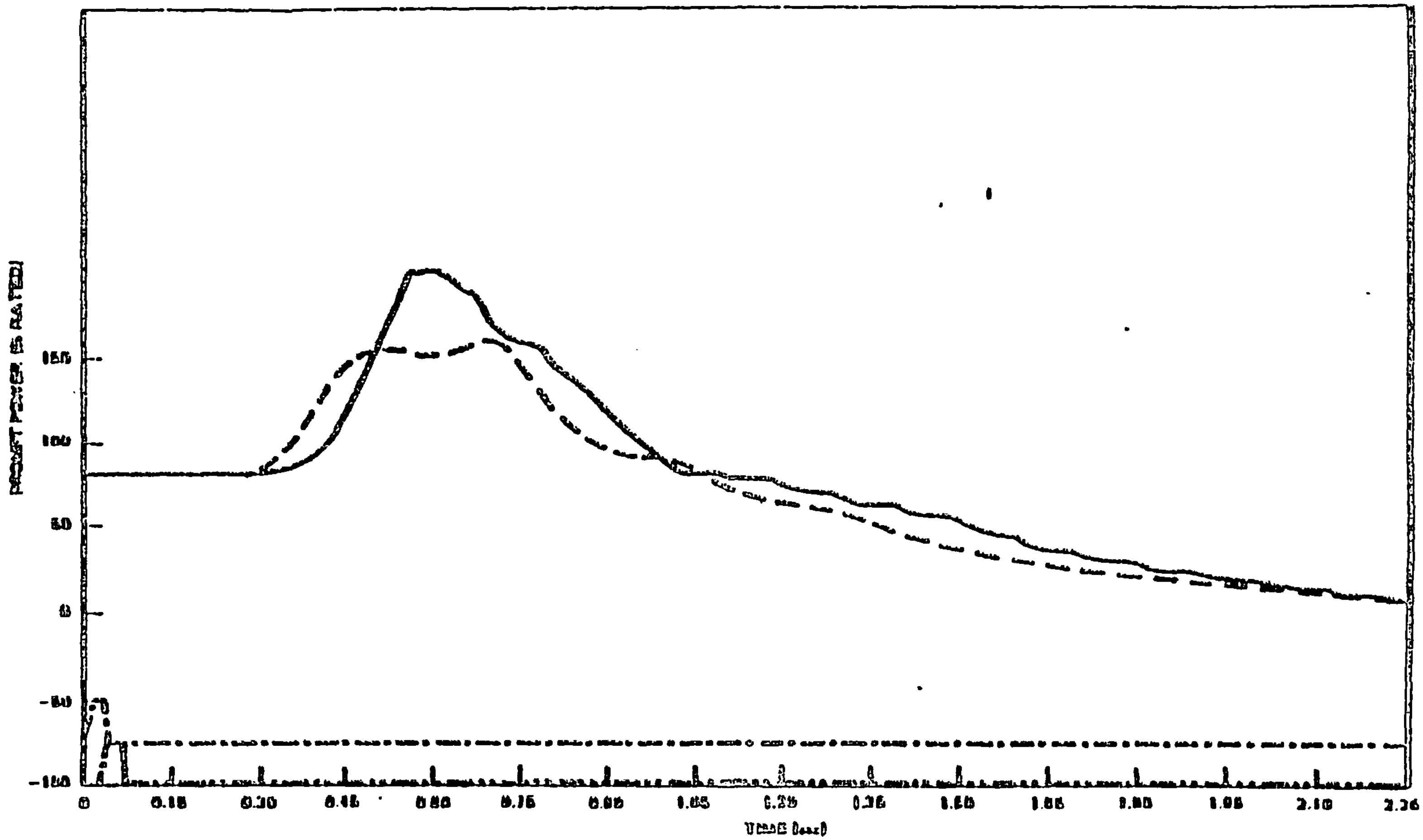

Figure 8 KKM Turbine Trip Prompt Neutron Power

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

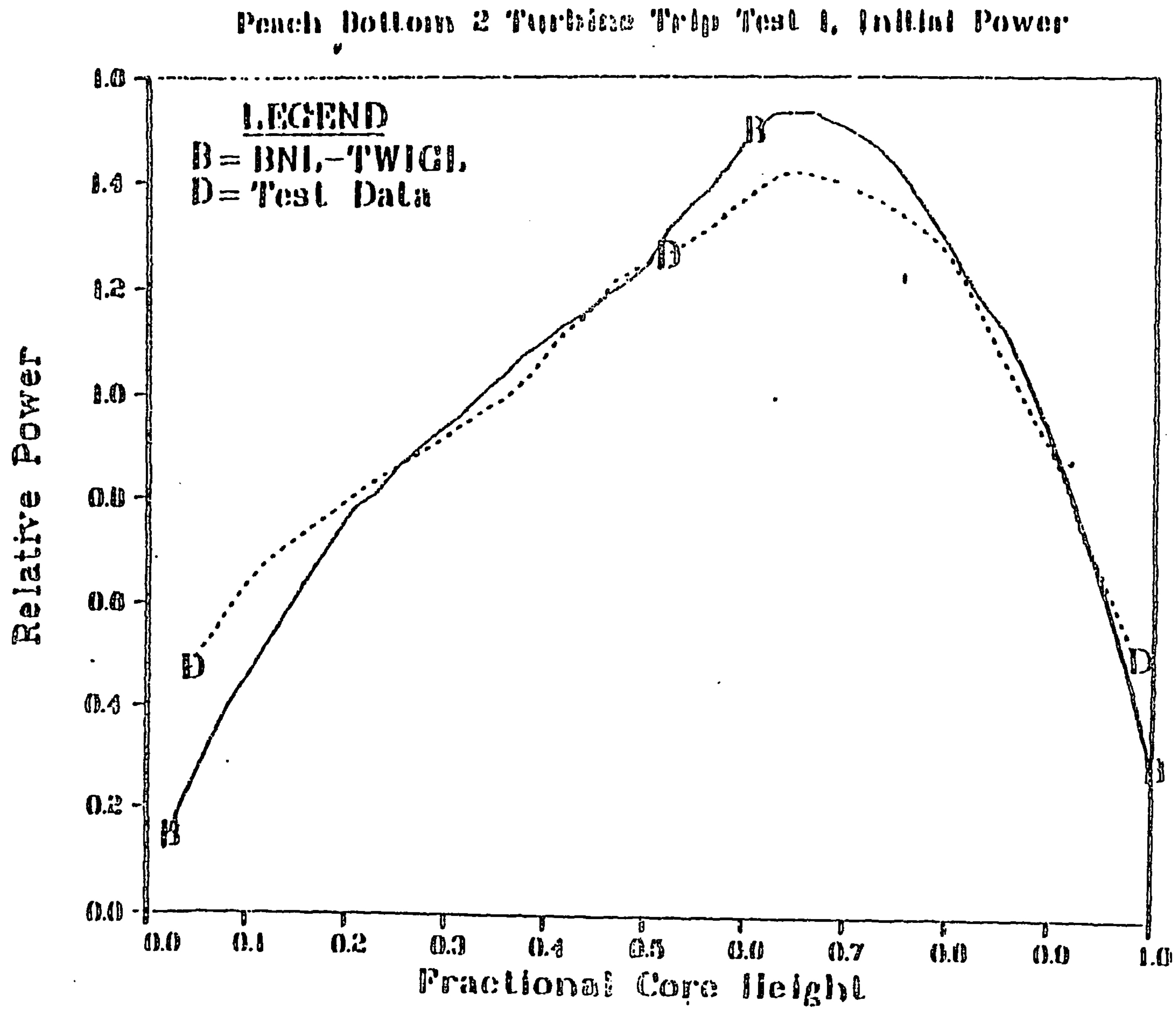

Figure 9

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Peach Bottom 2 Turbine Trip Test 3, Initial Power

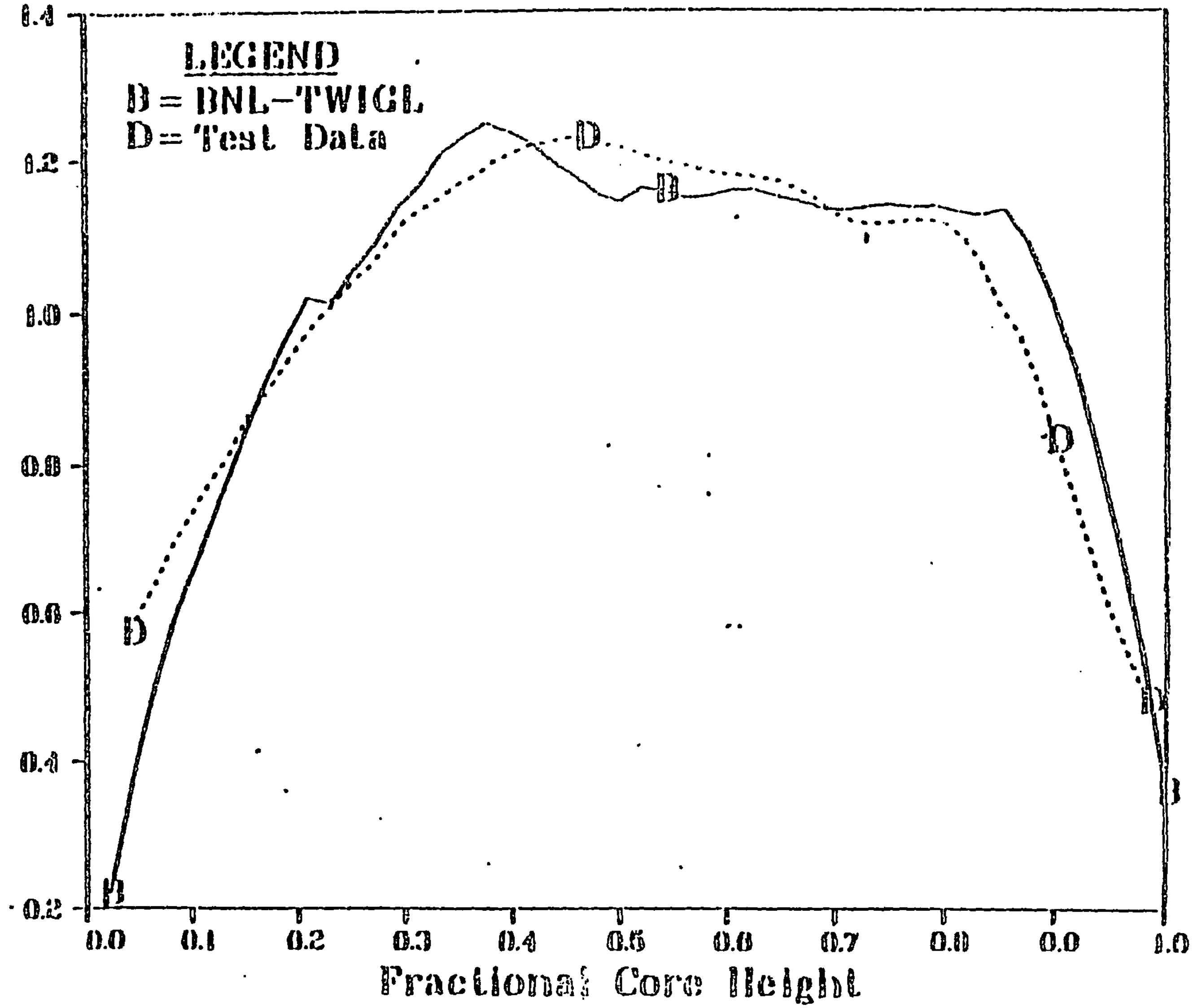
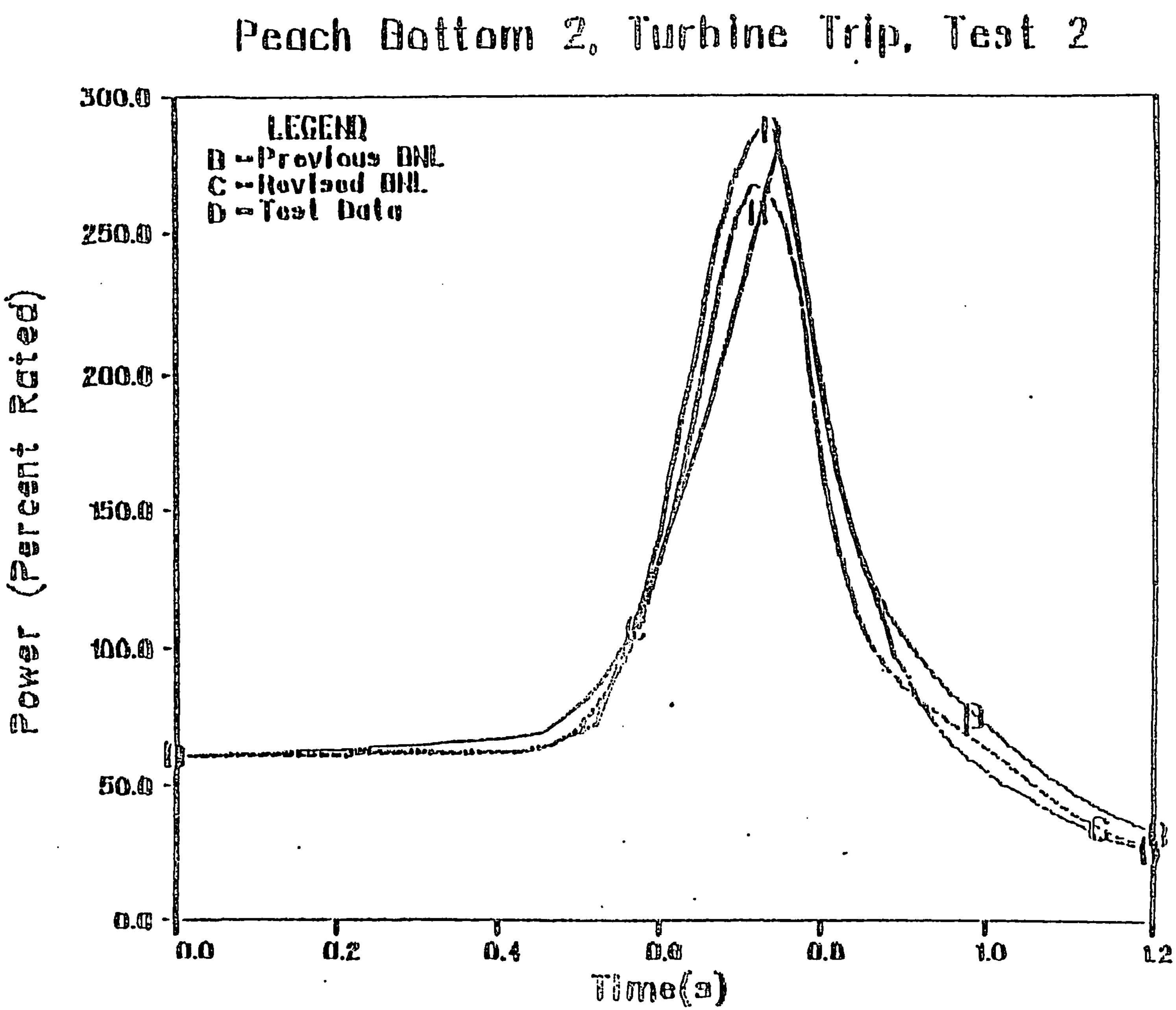


Figure 11

II-53



Peach Bottom 2, Turbine Trip, Test 2
LEGEND
B - Previous DNL
C - Revised DNL
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

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 450 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

The ODYN-SCAT prediction of the three Peach Bottom transient tests and one KKM transient test demonstrated a 2σ uncertainty of approximately 37% of $\Delta CPR/ICPR$ 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 $\Delta CPR/ICPR$ 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.

. 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

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 and 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

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 improved 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.

B. Statistical Approach for Reduction of Margin Penalty

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

value of $\Delta CPR/ICPR$ 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 $\Delta CPR/ICPR$ 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 $\pm$ 0.068 in $\Delta CPR/ICPR$ 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 $\Delta CPR/ICPR$ at a 95% confidence level).

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

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[a] 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.

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[a] 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 IV

INPUT PARAMETERS SENSITIVE FOR THE ANALYSES

1. CRD scram speed - at technical 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).

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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17. Rouhani, S. Z., "Void Measurements in the Region of Subcooled and Low Quality Boiling," Symposium on Two-Phase Flow, University of Exeter, Devon, England, June 1965.

18. Rouhani, S. Z., "Void Measurements in the Region of Subcooled and Low-Quality Boiling," Park II, AE-RTL-788, Aktiebolaget Atomenergi, Studsvik, Sweden, April, 1966.

19. NEDO-20913-P, "Lattice Physics Methods," C. L. Martin, June 1976.

20. "GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods," General Electric Report NEDC-20181, November, 1973.

21. D. F. Ross (NRC) letter to E. D. Fuller (GE) dated June 2, 1978.

22. EPRI NP-564, "Transient and Stability Tests at Peach Bottom Atomic Power Station Unit 2 at End of Cycle 2," L. A. Carmichael and R. O. Niemi, June 1978.

23. Letter from K. W. Cook to R. L. Tedesco, MFN 3137-78, "Transmittal of Responses to Round 2 Questions on the ODYN Transient Model," dated Dec. 13, 1978.

24. BNL-NUREG-21925, "BNL-TWIGL, A Program for Calculating Rapid LWR Core Transients," D. J. Diamond, Ed., October 1976.

25. BNL-NUREG-22011, "User's Manual for RELAP-3B-MOD 110: A Reactor System Transient Code," 1977.

26. BNL-NUREG-24903, "Core Analysis of Peach Bottom-2 Turbine Trip Tests," H. S. Cheng, D. J. Diamond, September 1978.

27. Memo from F. Odar to Z. R. Rosztoczy, "Meeting with General Electric at Brookhaven National Laboratory," October 17, 1978.

28. Letter from K. W. Cook of GE to Frank Schroeder of NRC, October 10, 1978 on Potential Differences between GE and BNL Models.

29. Letter from K. W. Cook of GE to Frank Schroeder of NRC, October 26, 1978, on Transmittal of Exposure Dependent Data.

30. Transcript of ACRS hearings held on March 19-20, 1979, Los Angeles, California.

31. Letter from K. W. Cook of GE to R. L. Tedesco, Clarification of ODYN Model Uncertainties, MFN 123-79, dated April 30, 1979.

32. Letter from K. W. Cook of GE to R. P. Denise, Additional Void Fraction Information Requested for ODYN Review, MFN-219-79, August 27, 1979.