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OECD/NRC BWR Turbine Trip Transient Benchmark as a Basis for Comprehensive Qualification and Studying Best Estimate Coupled Codes

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ABSTRACT

An OECD/NRC sponsored coupled-code benchmark has recently been initiated for a BWR turbine trip (TT) transient. Turbine trip transients in a BWR are pressurization events in which the coupling between core space-dependent neutronic phenomena and system dynamics plays an important role. In addition, the available real plant experimental data makes the proposed benchmark problem very valuable. Over the course of defining and coordinating the BWR TT benchmark, a systematic approach has been established to validate best-estimate coupled codes. This approach employs a multi-level methodology that not only allows for a consistent and comprehensive validation process but also contributes to the study of different numerical and computational aspects of coupled best-estimate simulations. The OECD/NRC BWR TT benchmark is presently being undertaken by a number of important companies and organizations from different countries. An international professional community has been established over the course of these benchmark activities, allowing in-depth discussions of coupled methodologies and their applications in order to determine additional requirements of the best-estimate calculations in analyzing reactivity transients. This paper provides an overview of the OECD/NRC BWR TT benchmark activities with emphasis on the discussion of the numerical and computational aspects of the benchmark. The benchmark team has also been involved on the technical side of the benchmark by analyzing different aspects and performing sensitivity studies of the different benchmark exercises. These studies allow the benchmark team to carry out a comparative in-depth analysis for the key parameters in evaluating rapid pressurization events. Selected results of such studies are presented and discussed in the paper. The identification and analysis of key parameters for a BWR TT transient by performing sensitivity studies allowed the benchmark team to assist the participants in the most efficient way.

1. INTRODUCTION

The Nuclear Energy Agency (NEA) of the Organization for Economic Co-operation and Development (OECD) has completed, under Nuclear Regulatory Commission (NRC) sponsorship, a PWR Main Steam Line Break (MSLB) Benchmark against coupled three-dimensional (3-D) neutron kinetics/thermal-hydraulic system codes [1]. Another OECD/NRC coupled-code benchmark has recently been initiated for a BWR turbine trip (TT) transient [2]. Turbine trip transients in a BWR are pressurization events in which the coupling between core space-dependent neutronic phenomena and system dynamics plays an important role. In addition, the available real plant experimental data makes the proposed benchmark problem very valuable [3]. Over the course of defining and coordinating the BWR TT benchmark a systematic approach has been established to validate best estimate coupled codes. This approach employs a multi-level methodology that not only allows for a consistent and comprehensive validation process but also contributes to the study of different numerical and computational aspects of coupled best-estimate simulations [4].

The OECD/NRC BWR TT benchmark is presently being undertaken by a number of important companies and organizations from different countries [3,4]. The benchmark team (The Pennsylvania State University (PSU) in cooperation with Exelon Nuclear and NEA/OECD) is responsible for coordinating the benchmark activities, answering the participants' questions and assisting them in developing their models, analyzing the submitted solutions and providing reports summarizing the results for each phase. A Frequently Asked Questions (FAQ) e-mail forum has been established via the OECD e-mail service. This forum has proved to be of invaluable help to the benchmark participants by providing rapid responses to questions as well as expediting the distribution of information. The benchmark materials can be accessed through benchmark web and ftp sites. Three international OECD/NRC BWR TT benchmark workshops have taken place during last two years – in Philadelphia, USA (November 2000, hosted by Exelon), in Villigen, Switzerland (hosted by Paul Scherrer Institute) and in Dresden, Germany (May 2002, hosted by Forschungszentrum Rossendorf). An international professional community has been established over the course of these benchmark activities, allowing in-depth discussions of coupled methodologies and their applications in order to determine additional requirements of the best-estimate calculations in analyzing reactivity transients.

2. DESCRIPTION OF ACTUAL WORK

Three-turbine trip (TT) transients at different power levels were performed at the Peach Bottom (PB)-2 Atomic Power Station (a GE BWR/4) prior to shutdown for re-fueling at the end of Cycle 2 in April 1977. The second test has been selected for the benchmark problem (it has the highest quality measured dataset) to investigate the effect of the pressurization transient (following the sudden closure of the turbine stop valve) on the neutron flux in the reactor core. The tests were designed to produce plant/core responses that approached the design basis conditions as closely as possible. The actual data were collected, including a compilation of reactor design and operating data for Cycles 1 and 2 and the plant transient experimental data. This transient was selected for this benchmark study because it is a dynamically complex event with reactor variables changing very rapidly, and it constitutes a good problem to test the coupled codes on both levels: neutronics/thermal-hydraulic coupling and core/plant system coupling. In the TT2 test, the thermal-hydraulic feedback alone (void feedback plays the major role while Doppler feedback plays a subordinate role) limited the power peak and initiated the power reduction. The reactor scram then inserted additional negative reactivity a completed the power reduction and eventual core shutdown. Figure 1 illustrates the measured time history data of the core fission power and total reactivity. This provides a unique opportunity for a comprehensive feedback testing and examination of capability of advanced codes to analyze complex transients with coupled core/plant interactions though comparison with actual experimental data.

Benchmark specifications, which are based on the PB-2 TT2 test data, were published by the NEA/OECD in a format similar to the PWR MSLB final specifications [2]. The benchmark specifications included a complete set of the initial and boundary conditions that are needed for the participants to perform the three exercises mentioned above. The specifications include: 1) Thermal-hydraulic plant data; 2) Neutron kinetics core specifications including core geometry, neutron modeling and composition map; 3) Macroscopic cross-section library which accounted for the core exposure distribution through the cross-sections. Separate consistent libraries are provided for 3-D and 1-D kinetics models.

The benchmark team collects the analytical results for the three exercises from the participants and analyzes the results in a standard graphical form with a descriptive narrative summarizing the comparison among the participants' results, the reference calculation and test data. This benchmark involves both code-to-code and code-to-data comparative analyses on different levels: single values, 1-D distributions, 2-D distributions and time histories. The Automated Code Assessment Program (ACAP) [7] is being used to perform these analyses. Comparisons for selected parameters are presented and discussed below.

The chosen benchmark transient – the turbine trip transient – is a rapid pressurization event in BWR reactors, which is characterized by tightly coupled thermal-hydraulic/neutronics phenomena. The core spatial effects are dominated by the axial changes; however, at the end-of-cycle 2, a slight radial exposure non-symmetry exists, which introduces radial spatial effects in the initial steady state. This radial non-symmetry has negligible impact on the transient response since the radial power distribution remains fairly uniform throughout the pre-scram portion of the transient. Previous Exelon studies indicated that using a one-dimensional (1-D) neutron kinetics model has limitations (a "single channel" core model with average power and flow does not respond dynamically the same way as the average of all the channels – three-dimensional (3-D) average), and 1-D cross-sections are dependent on the core thermal-hydraulic model used for their generation (i.e. they are not universal and additional "normalization" procedures are needed to adjust these cross-sections for specific code application). The previous studies also demonstrated the major difficulty in validating coupled codes – it is difficult to determine if inaccurate thermal-hydraulic response is driving inaccurate neutronic response or vice versa. Similar relationships exist within the core/system interactions. In order to solve these problems a consistent and comprehensive benchmark approach has been developed – the benchmark consists of three separate exercises, two initial states and four transient scenarios.

The first exercise consists of performing a system thermal-hydraulic calculation (no neutron kinetics model involved – core power (or reactivity) is a fixed input) for the PB-2 TT transient. The purpose of the first exercise is to test the thermal-hydraulic system response and to initialize the participants' system models. This approach is used to develop/refine key sub-models including steam lines, steam bypass system, jet pumps, steam separators and upper down-comer region where a distinct steam-water interface exists. Fourteen participants representative of eight countries have submitted results for the first exercise. The single values (integral parameters) and 1-D axial distributions are analyzed using standard statistical methodology. For example, for the parameters for which the measured data is available the measured values are used as the reference. Such parameters are the core inlet enthalpy and core average pressure drop at the initial conditions of TT2 test. The measured value for the core inlet enthalpy is 1209.055 KJ/kg, and participants' results display a standard deviation of ±5.243 KJ/kg. The measured value of the core pressure drop is 0.113561 MPa and the calculated standard deviation of the participants' results is ±0.020 MPa. For the parameters for which measured data is not available (code to code comparisons) a mean solution is generated to serve as the reference. This can be seen in Figure 2, which shows the mean solution and standard deviations of the axial core average axial void fraction distribution. The participants' results for the time histories are analyzed using the

ACAP program with the recommended three different methods – D'Auria Fourier Transform Method, Mean Error Method and Continuous Wavelet Transform Method. Equal weight is given to each method in the calculation procedure of the normalized combined figure of merit for each participant. During the analysis of the preliminary results for the first exercise it became obvious that the sources of deviations in the participants' predictions stem from the differences in modeling of the two key parameters – pressure response and core flow response. The accurate prediction of the pressure response depends on the number of models used by a given code: steam line model (requires adequate nodalization and treatment of momentum effects), the steam bypass system model and the steam separator model (steam separator inlet inertia and non-equilibrium effects at steam-water surfaces must be properly treated). As one of the very important models for this transient, the steam bypass system should be modeled consistently to accurately predict bypass flow response. An example for the two code predictions (TRAC-BF1 and RETRAN) is provided in Figure 3, followed by the results of the same two codes for the core outlet pressure illustrated in Figure 4. The prediction of core flow response is sensitive to the jet pump model used (it must properly represent dynamic response of flow to pressure) and the core exit/separator region model (it must properly represent dynamic response of two-phase flow). In-depth discussions concerning the aforementioned modeling issues were carried out during the 2nd Benchmark Workshop [4], and additional information was provided to participants, which led to much better agreement in the final results.

The second exercise consists of performing coupled-core boundary conditions calculations. The purpose of the second exercise is to test and initiate the participants' core models. Thermal-hydraulic boundary conditions are provided to the participants from the benchmark team. The thermal-hydraulic core boundary conditions provided are the core inlet pressure, core exit pressure, core inlet temperature and core inlet flow. The second exercise consists of two options. Option 1 of the second exercise is to perform a coupled 3-D kinetics/T-H calculation for the reactor core using the PSU-provided boundary conditions at core inlet and exit. Option 2 of the second exercise is to perform coupled 1-D neutron kinetics and core thermal-hydraulics boundary conditions calculations. Three-dimensional and 1-D cross-section libraries are provided to the participants. The core inlet flow is provided in two formats: total core flow as a function of time and radially distributed flow as a function of time for thirty-three channels. In addition, the benchmark team provided the participants with normalized power vs. flow correlations for the different assembly types based on the detailed (each assembly represented by a thermal-hydraulic channel) modeling for the initial steady-state conditions. The studies performed by the benchmark team indicated that these correlations also apply reasonably well during the transient, which provided an opportunity for the participants to develop their own core-coupled spatial mesh overlays. An additional steady state was defined in the framework of the second exercise – hot zero power (HZP) state with fixed thermal-hydraulic feedback. This allows for "clean" initialization of the core neutronics models and cross-section modeling algorithms. The same methodologies are used for comparative analysis of the participants' results in the second exercise as in the first exercise. The difference is only in the fact that in the second exercise there are also (in addition to the single values, 1-D distributions and time histories) 2-D (core-averaged radial) distributions to be compared. The mean eigenvalue calculated for the HZP state is 0.99592. The mean core-averaged axial relative power distribution and standard deviations for this state is shown in Figure 5. The mean solution for eigenvalue for initial conditions of TT2 test is 1.0037, and the mean distribution and standard deviations for the core averaged axial relative power distribution for this state is shown in Figure 6. The mean core-averaged radial power distribution is shown in Figure 7. This figure illustrates the radial non-symmetry at the initial conditions, which is propagated into the transient. During the comparative analysis of the preliminary participants' results the following sources of modeling uncertainties were identified: core pressure drop (local losses and friction models), core bypass modeling and void feedback model (sub-cooled boiling and vapor slip). The fuel heat transfer (UO_2 conductivity and gap conductivity) and direct heating (2% to in-channel flow and 1.7 % to bypass flow) were specified. The scram initiation time and the speed of the rod insertion were also specified. Other important modeling issues were identified such as the impact of using assembly discontinuity factors (which are also provided to the participants in a similar table format as for the two group cross-sections), xenon correction (to account for the actual xenon concentration distribution at the initial steady-state conditions of the turbine trip test 2), number of thermal-hydraulic channels and spatial mapping schemes with the neutronics core model and bypass density correction in the cross-section feedback modeling (to account for the deviations of bypass density from the saturated value used in the cross-section homogenization – the cross-sections are generated by homogenizing the bypass region associated with the lattice).

The third exercise consists of performing a coupled 3-D kinetics/T-H calculation for the core and 1-D thermal-hydraulics modeling for the balance of the plant and/or coupled 1-D kinetics with system thermal-hydraulic modeling. There are four scenarios – the best estimate scenario (the real test with available measured data) and three extreme versions (see Figure 10). The extreme scenarios were introduced to provide the opportunity to better test the coupling and feedback modeling: 1) Turbine trip without bypass system relief opening (increases the peak pressure, and thus, the power peak and provides enough pressurization for safety/relief valve opening); 2) Turbine trip without scram (produces secondary power peaks, which is of particular relevance for testing the coupled-code predictions); 3) Combined extreme scenario – turbine trip with bypass system relief failure and without reactor scram (it can be see from Figure 10 that this is a very challenging case for code-to-code comparisons). Since the benchmark-measured data also contain the LPRM (local power range monitor) measurements, the benchmark team provided the participants with the description of an appropriate algorithm to model LPRM response and the necessary associated data (microscopic detector cross-sections, flux factors, etc.). The deadline for submitting the results for Exercise 3 is the end of August 2002. So far eight participants from five countries have submitted their results.

3. RESULTS Of SENSITIVITY STUDIES

The benchmark team has also been involved on the technical side of the benchmark by analyzing different aspects and performing sensitivity studies of the different benchmark exercises. Two codes have been used for this purpose – the Exelon version of RETRAN [5] and TRAC-BF1/NEM [6]. These studies allow the benchmark team to carry out a comparative in-depth analysis for the key parameters in evaluating rapid pressurization events. One such sensitivity study concerns the steam separator inertia. Usually, the reactor vendor provides an effective inertia for the separators, which is based on the combination of measurements and calculations. The separator inertia effect in the simulated transient can be described as follows: the dynamic “pressure wave” from the stop valves enters the reactor vessel in the steam dome; the pressure wave has two paths to reach the core – through the vessel down-comer (solid water) and through the separators (water/steam mixture); the down-comer path has only a small attenuation (delay) and the jet pump inertia is the dominating characteristic; the steam separator path has higher attenuation (delay) due to the compressibility of the steam-water mixture; the core exit/separator region of the model must properly represent the dynamic response of two-phase flow. Inertial terms and pressure drop are very important. The sensitivity studies performed with RETRAN involved three simulations – nominal (base) case based on 50%/50% split of vendor provided inertia at separator entrance (inlet) and liquid exit; a ‘high inlet inertia’ case with 75%/25% distribution, and a ‘low inlet inertia’ case with 25%/75% distribution. The impact on core inlet flow response, core pressurization response and subsequently on core power response is evaluated – see Figures 9, 10, 11, and 12. The obtained results indicated the following tendencies:

- Higher inlet inertia increases the effective ‘resistance’ to the pressure wave through the separators. The higher ‘resistance’ through the separators results in a delay in the decrease of flow at the exit of the core and a larger increase in flow at the inlet relative to the base case.

The core inlet and exit flow oscillations become more pronounced. The pressure oscillations in reactor steam dome and core exit also become more pronounced. The peak value of first oscillation is greater. The frequency of the oscillations is reduced. The higher first pressure oscillation results in greater void collapse and higher peak reactivity and neutron flux.

- Lower inlet inertia decreases the effective ‘resistance’ to the pressure wave through the separators. The lower ‘resistance’ through the separators results in an acceleration in the decrease of flow at the exit of the core and a smaller increase in flow at the inlet to the core relative to the base case. The core inlet and exit flow oscillations become less pronounced. The pressure oscillations in reactor steam dome and core exit also become less pronounced. The peak value of first oscillation is lower. The frequency of the oscillations is increased. The lower first pressure oscillation results in smaller void collapse and lower peak reactivity and neutron flux.

The value of the separator inlet inertia appears to impact the ‘distribution’ of the energy from the pressure wave entering the reactor vessel (i.e., the portion of the pressure wave that is transmitted to the core through the down-comer versus the separators). This sensitivity study illustrates the importance/impact of key parameters in simulating rapid pressurization events in a BWR. It also illustrates the sensitivity of the neutron flux response to what appear to be small changes in the pressurization rate and magnitude.

4. CONCLUSIONS

The PB TT2 test has been analyzed previously elsewhere with different codes and models. [8,9] These analyses involved point kinetics and 1-D kinetics system simulations and 3-D kinetics/core thermal-hydraulics boundary conditions calculations. Each of the organizations performing these separate analyses generated their own point kinetics parameters, 1-D cross-sections and 3-D cross-section libraries. In this way it was not possible to compare directly the results of different organizations especially for the parameters where the measured data is not available. In most of the 3-D core boundary conditions analyses the cross-section functionalization for instantaneous dependencies was done either by using polynomial fitting procedure or the procedure using multilevel tables with base and partial cross-sections. In both cases the instantaneous cross-term effects (which are important for the transient analysis) are not modeled completely, which led to different degree of underestimation of the void feedback depending on the procedure used.

The current OECD/NRC BWR TT benchmark is designed to provide a validation basis for the new generation best estimate codes – coupled 3-D kinetics system thermal-hydraulic codes. Based on the previous experience, three benchmark exercises were defined in order to develop and verify, in a consistent manner, the thermal-hydraulic system model, the coupled core model, and the coupled core/system modeling. The three defined exercises are also help to identify the key parameters for modeling a turbine trip transient. This in turn allows the evaluation of these key parameters, through the performance of sensitivity studies, which allow the benchmark team to assist the participants in the most efficient way. The introduction of the extreme scenarios of Exercise 3 contributes to the study of different numerical and computational aspects of coupled simulations. The participants use the cross-section library, generated by the benchmark team, which removes the uncertainties introduced with using different cross-section generation and modeling procedures. The defined benchmark cross-section modeling approach is a direct interpolation in multi-dimensional tables with complete representation of the instantaneous cross-section cross-term dependencies.

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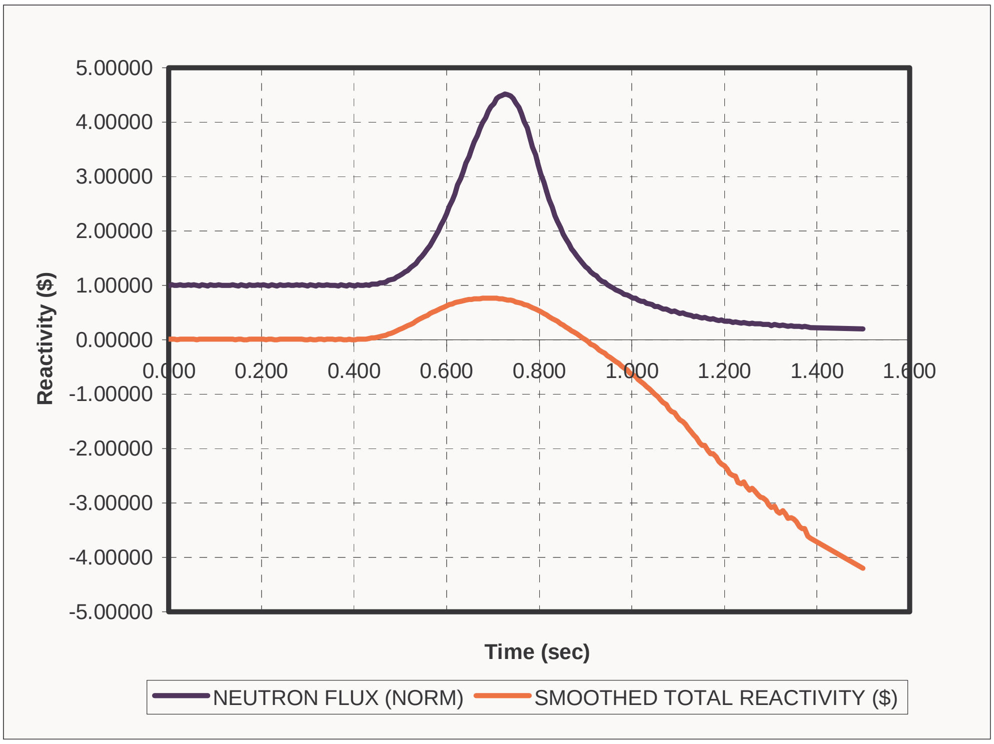

Figure 1. Neutron Flux and Reactivity Time Evolution for PB2 TT2

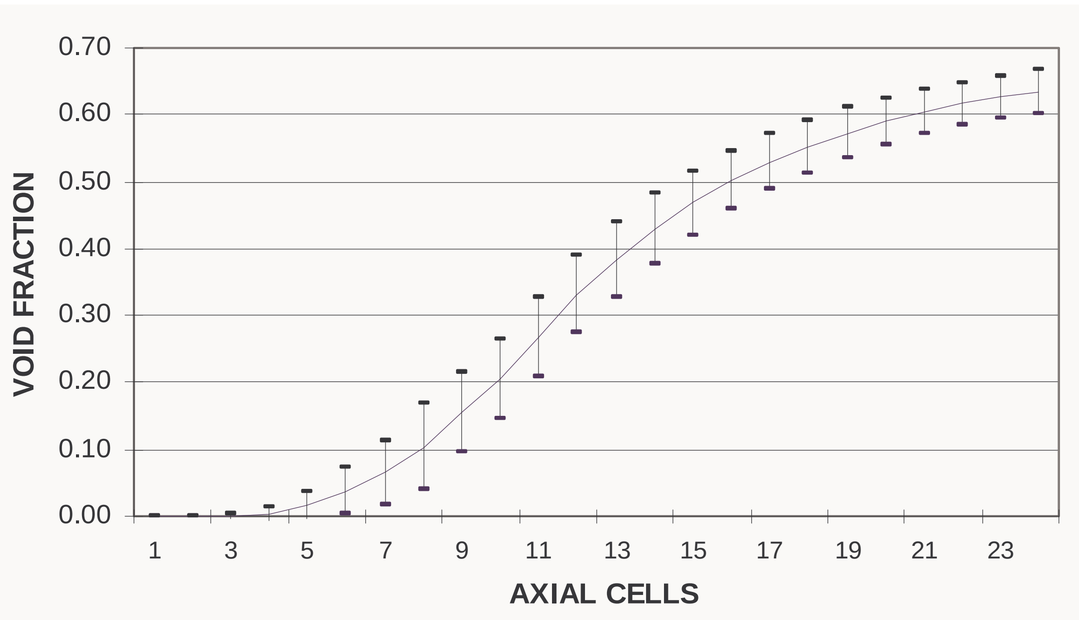

Figure 2. Exercise 1 – Core Average Axial Void Fraction Distribution – Mean Solution and Deviations

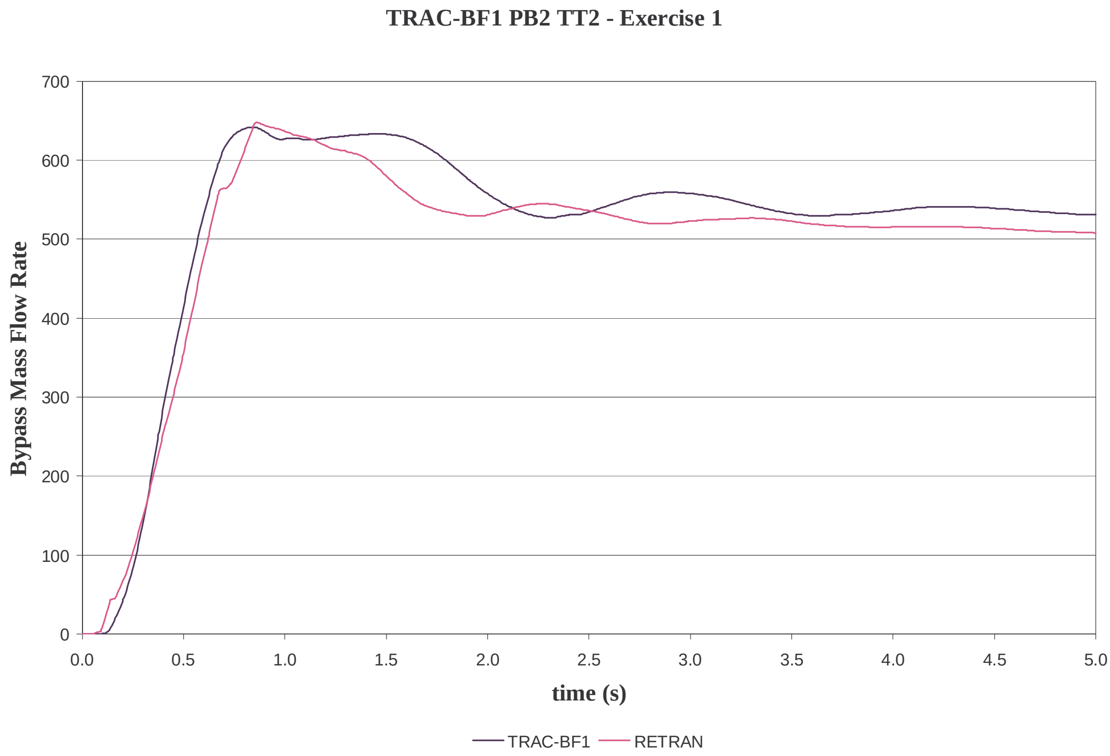

Figure 3. Bypass Mass Flow Rate Predictions for First Benchmark Exercise

Figure 4. Core Exit Pressure Predictions for the First Benchmark Exercise

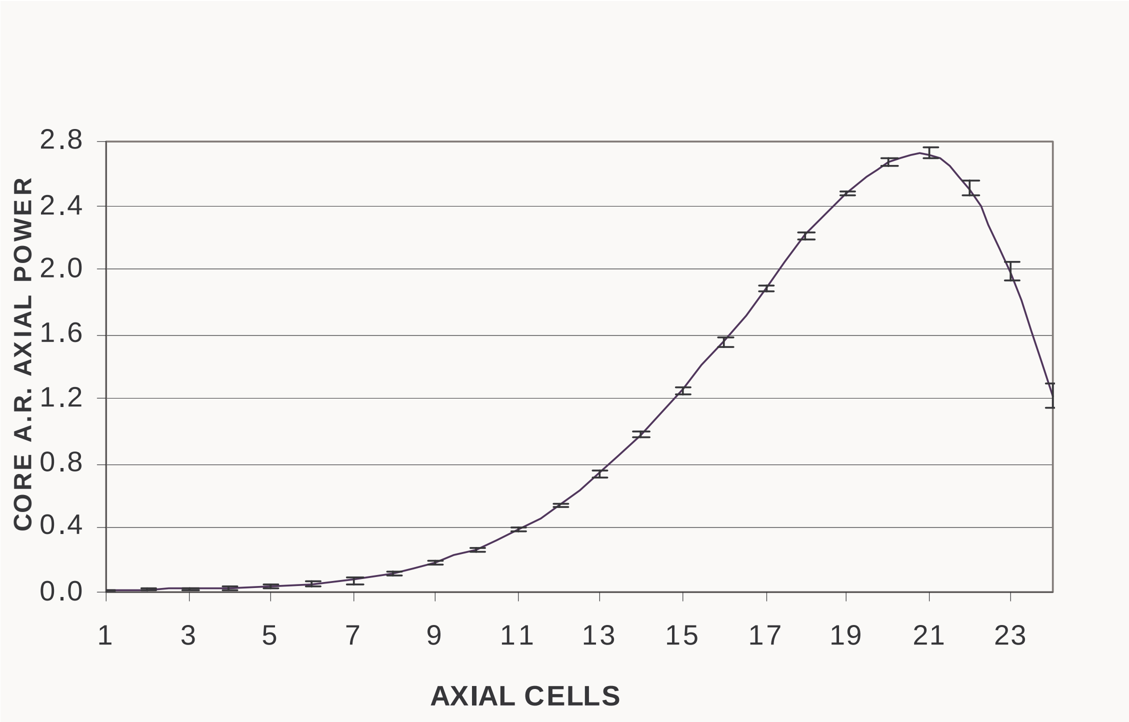

Figure 5. Exercise 2 - Core Average Relative Axial Power Distribution

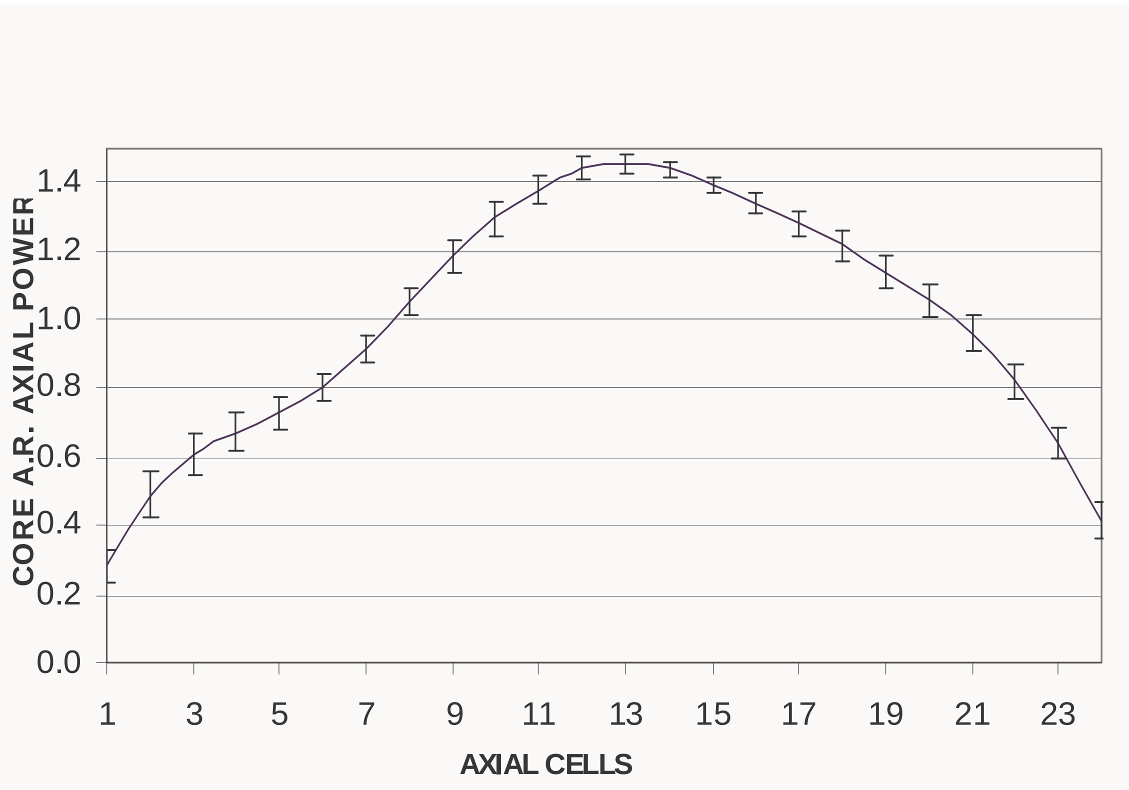

Figure 6. Exercise 2 - Normalized Transient Fission Power Response

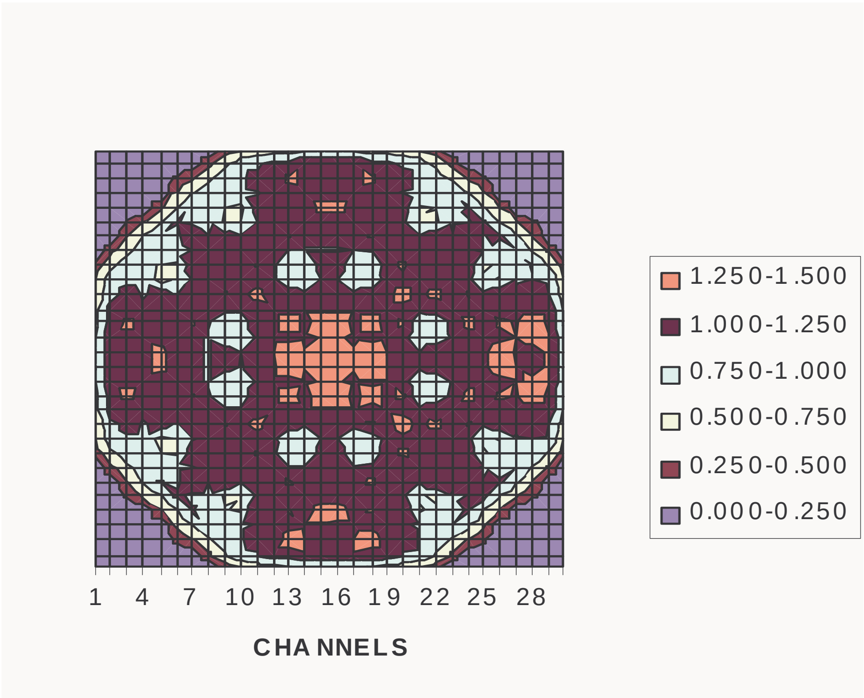

Figure 7. Two-dimensional normalized power distribution for initial steady state

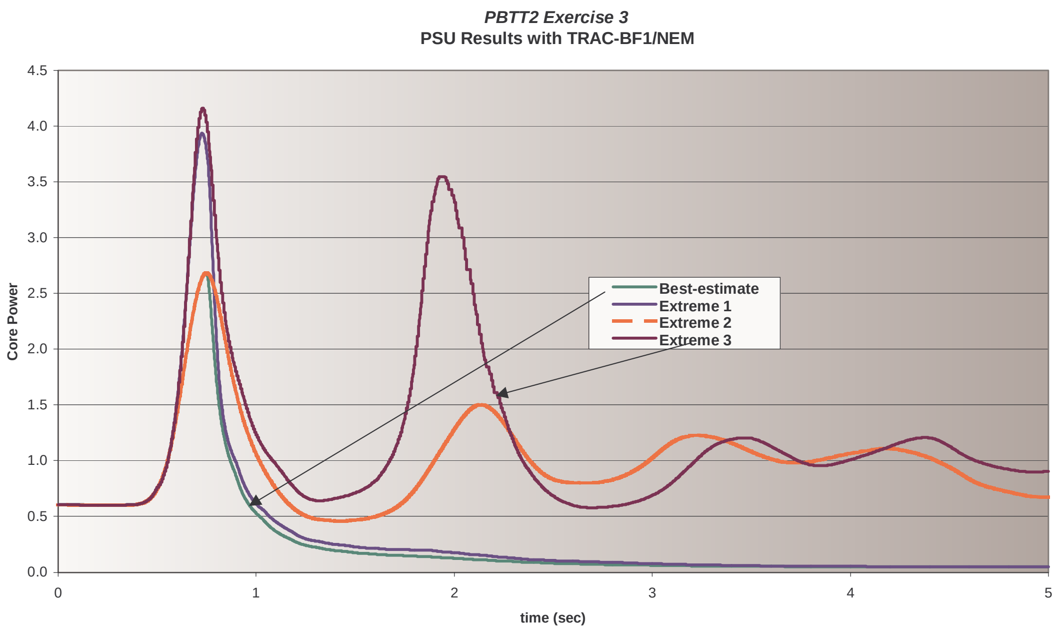

Figure 8. Relative core power response time evolution for different benchmark scenarios

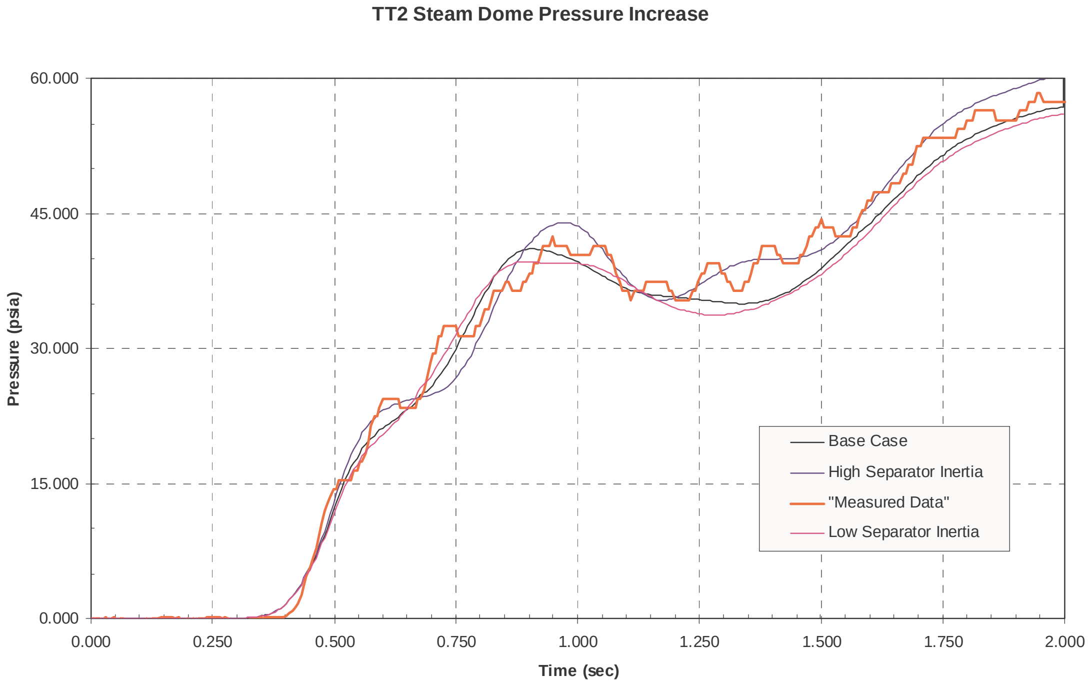

Figure 9. Time Evolution of Steam Dome Pressure Response for Different Cases

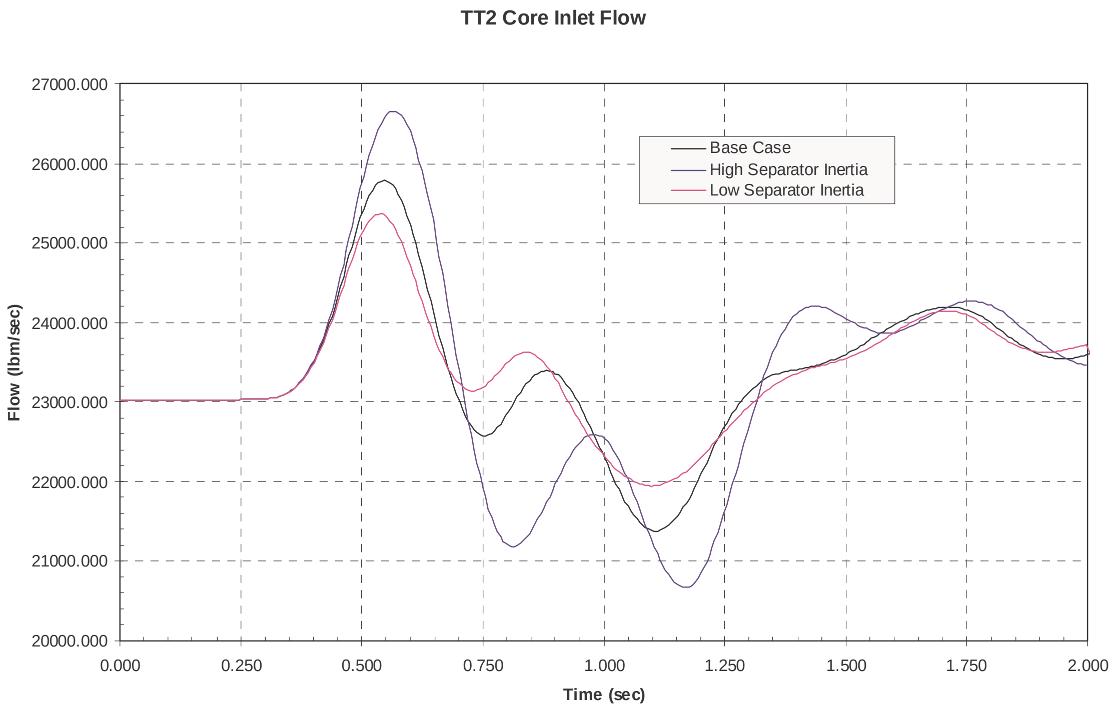

Figure 10. Time Evolution of Core Inlet Flow Response for Different Cases

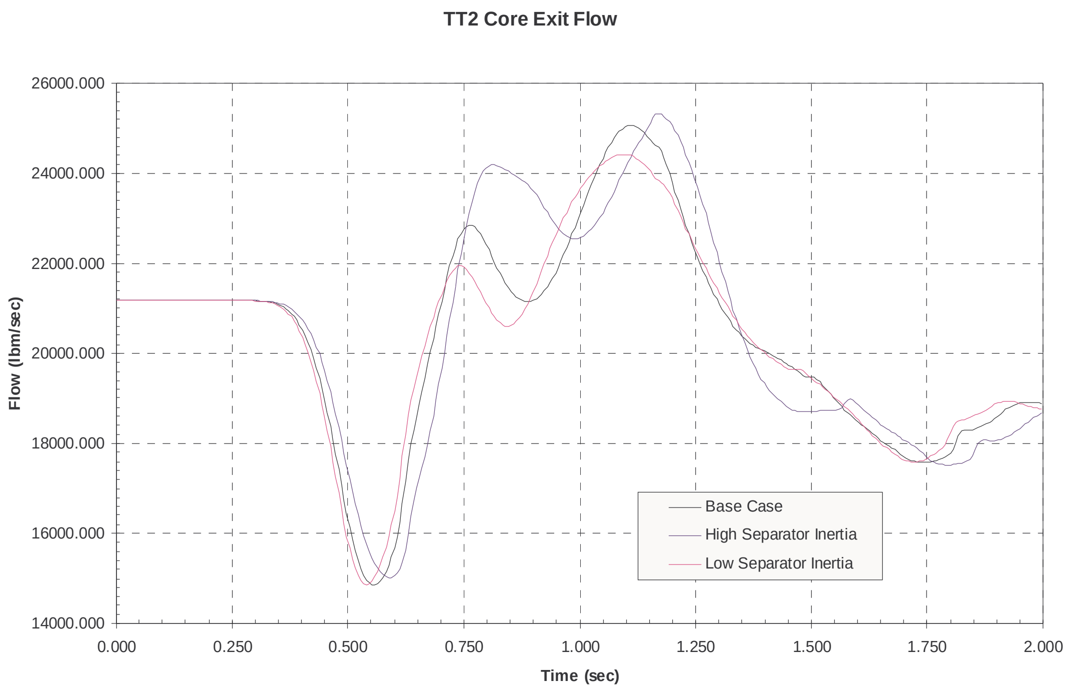

Figure 11. Time Evolution of Core Exit Flow Response for Different Cases

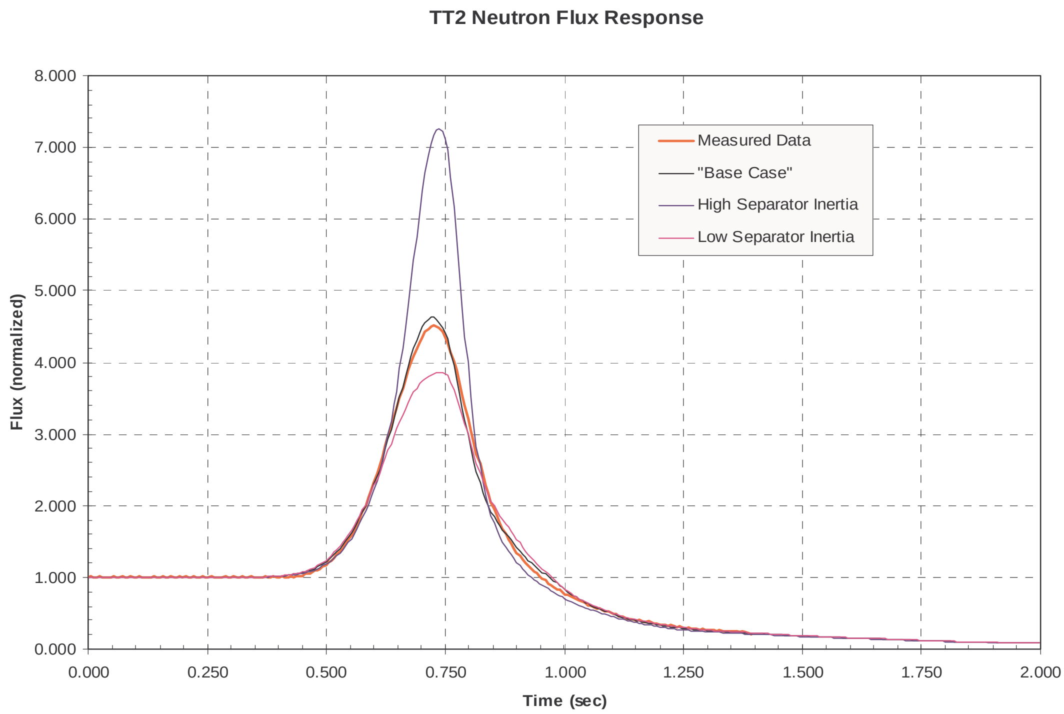

Figure 12. Time Evolution of Neutron Flux Response for Different Cases