Nuclear Energy & Infrastructure Systems Division - Energy Sciences & Technology Dept.

Building 130
Phone 631 344-2616
Fax 631 344-3957
bromley@bnl.gov

managed by Brookhaven Science Associates
for the U.S. Department of Energy

Memo

date: August 13, 2001

to: David J. Diamond

from: Blair P. Bromley

subject: Pulse Width Variation in a Rod Ejection Accident

1. Objectives

The work completed had the following objectives:

i) To compute pulse width (full width at half maximum- FWHM) of the power curves for the Three Mile Island Unit 1 Pressurized Water Reactor (TMI-1 PWR) core model in the event of a super-prompt-critical rod ejection accident (REA) at Hot Zero Power (HZP) in the central fuel assembly (Rod 7A) at both EOC and BOC for various control rod worths and delayed neutron fractions.

ii) To correlate the pulse width and maximum fuel pellet enthalpy change with control rod worth, and delayed neutron fractions.

iii) To compare enthalpy and pulse width results with those obtained by Russian colleagues at the Russian Research Center – Kurchatov Institute (RRC-KI) in Moscow.

iv) To make recommendations for future computational analyses of the REA in PWR's, and also for future fuel testing experiments.

2. Methodology

2.1 PARCS Reactor Dynamics Simulation Code

The PARCS (Purdue Advanced Reactor Core Simulator) code (Version v1.05) was used to simulate both the steady-state and transient reactor dynamics behavior of the TMI-1 PWR core model. The PARCS code is a three-dimensional, two-group diffusion model using nodal methods [1]. PARCS can be coupled with a thermal-hydraulics code such as RELAP5 to get a complete self-consistent simulation of the reactor core, or a simplified thermal-hydraulics model that is incorporated in the PARCS code can be used in a stand-alone mode. Although the stand-alone version of PARCS has a limited range of applicability and accuracy, it can be run more quickly than the PARCS code coupled with RELAP5, and it is used to obtain the simulation results shown in this memorandum.

2.2 Rod Ejection Accident Analysis

A rod ejection accident in the TMI-1 core was modeled on a transient basis with the PARCS code in the stand-alone mode. The PARCS code was first used to compute the initial steady-state neutron flux, power, and temperature distributions and the effective multiplication factor at hot zero power (2.772 kW_{th}, 1.0e-4 % of full power). The computed multiplication factor is used to normalize the neutron source distribution such that the initial reactivity will be zero ($k_{effective} = 1$, $\rho_{total} = 0$).

The PARCS code was then used to perform a 3-second transient simulation of the core neutronics and thermal-hydraulics behavior during and after the control rod in the central fuel assembly (Rod 7A) is withdrawn over a period of 100 milliseconds. This model simulates a rod ejection accident. The withdrawal rate of the central control rod is constant. The PARCS simulation was programmed to initiate a reactor trip when the power level reached 112% of the normal full power level (2772 MWth). Safety banks would begin insertion 400 milliseconds after the 112% power level was reached, and would be fully inserted in 2.2 seconds at a constant rate

The pulse width of the power transient in an REA is the full width half maximum (FWHM) value. The FWHM will not be symmetric since the time period between the half-maximum and the maximum after the power peak will be longer than before the peak due to the delayed negative effect of Doppler fuel and moderator feedback.

3. Observations and Discussion of Results

3.1 Sample Transient Runs at EOC and BOC

Transient simulations were performed with PARCS to evaluate the TMI-1 PWR core neutronics and thermal-hydraulics behavior in the event of a prompt critical REA from HZP at both EOC and BOC. Sample results of REA power transients at HZP for both EOC and BOC are shown in Figures 1 and 2. The ejected rod worth in the EOC and BOC cases are approximately $1.22 and $1.19, respectively.

3.2 Parametric Studies of Rod Worth and Delayed Neutron Fraction

A set of parametric studies were completed in which the multiplication factors on the macroscopic absorption and fission yield cross sections were adjusted to artificially change the worth of control rod 7A in the central fuel assembly. In addition to variation of rod worth, the delayed neutron fraction at EOC was varied from 70 to 120% of the nominal value. For each transient simulation of a given rod worth and delayed neutron fraction, there would be an associated pulse width and maximum increase in fuel pellet enthalpy.

The variation of the FWHM power pulse widths for various REA events at EOC and BOC are shown in Figures 3 to 5. According to Figure 3, the pulse width tends to have an inverse relationship with the normalized rod worth, and goes down with increasing delayed neutron fraction. The pulse width is even lower at BOC than at EOC for a given rod worth. According to Figure 4, all the data points at EOC for various rod worths and delayed neutron fractions collapse onto a single curve when the pulse width is plotted against the absolute difference between the rod worth and the delayed neutron fraction. Since the maximum fuel pellet enthalpy rise is proportional to the absolute reactivity, it is no surprise that the pulse width also tends to have inversely proportional relationship with the fuel enthalpy rise. The pulse width for a given enthalpy rise tends to be slightly lower at BOC than at EOC. For an enthalpy rise of more than 40 cal/g, the FWHM pulse width is less than 20 ms.

3.3 Comparison with RRC-KI Results

Russian colleagues at the Russian Research Center – Kurchatov Institute have carried out a series of parametric studies of the REA in the TMI-1 PWR core model using their own 3-D core neutronics code, BARS, coupled with the RELAP-5 thermal-hydraulics code. The BARS code uses a Green's function approach to solve a multi-group (4 or 5 groups) diffusion model of the core on a pin-by-pin basis in the radial plane while a harmonic expansion is used to represent the flux in the axial direction. [3]

One sample set of calculations from the RRC-KI group [3] is shown in Table 1 along with PARCS data at EOC. For a comparable ejected rod worth (about $1.2), the BARS results for the maximum power and pulse width are very similar to those found with PARCS. The maximum power and pulse width computed by PARCS were approximately 387% and 63 ms respectively, while the same calculations by BARS were approximately 391% and 62.6 ms respectively. The differences between the BARS and PARCS calculations for the peak power and pulse width are approximately 1%, as is the rod worth. There is a greater discrepancy in calculation for the enthalpy increase. In the case of no reactor trip, a sample calculation done with PARCS shows an enthalpy increase of 16.6 cal/g at 2.5 seconds, while BARS/RELAP-5 shows a an enthalpy increase of 20.6 cal/g, a 20% difference. This discrepancy may be explained by the differences in both the neutronics and thermal-hydraulic models are different. The fuel properties data used in the heat transfer calculations may also be an explanation.

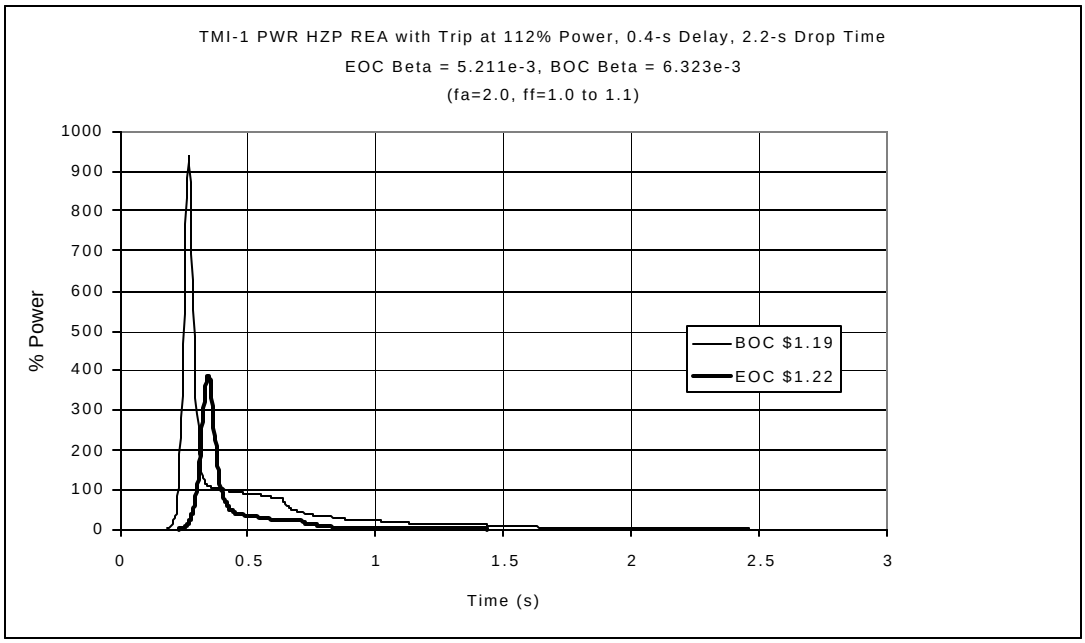

Figure 1: Power Transients for REA at HZP in TMI-1 PWR

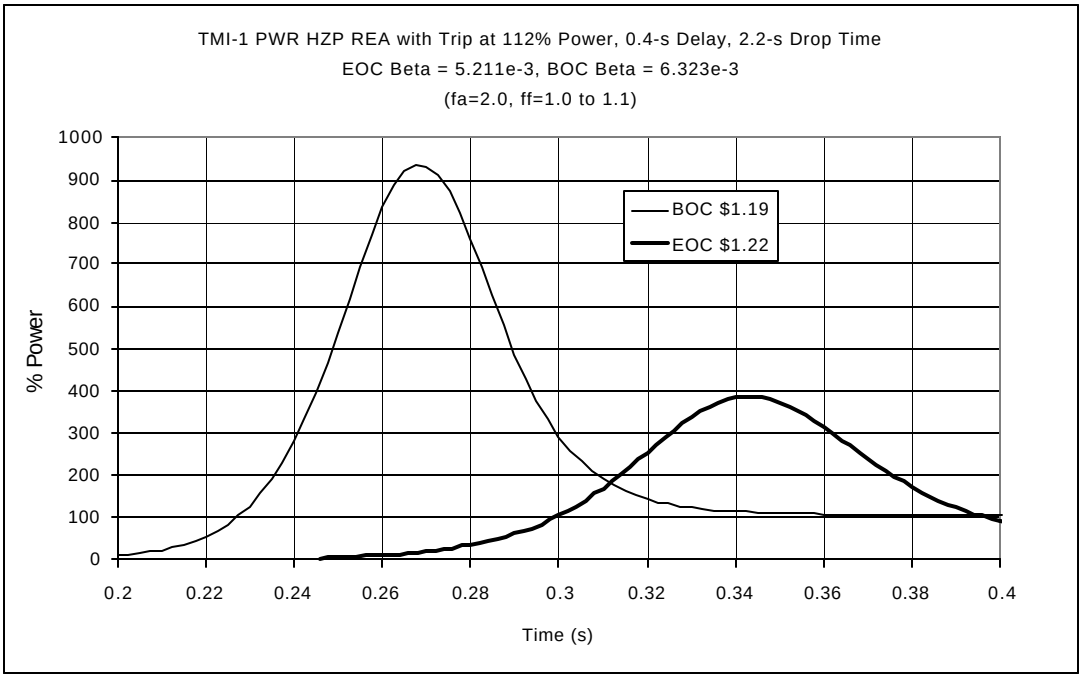

Figure 2: Power Transients for REA at HZP in TMI-1 PWR

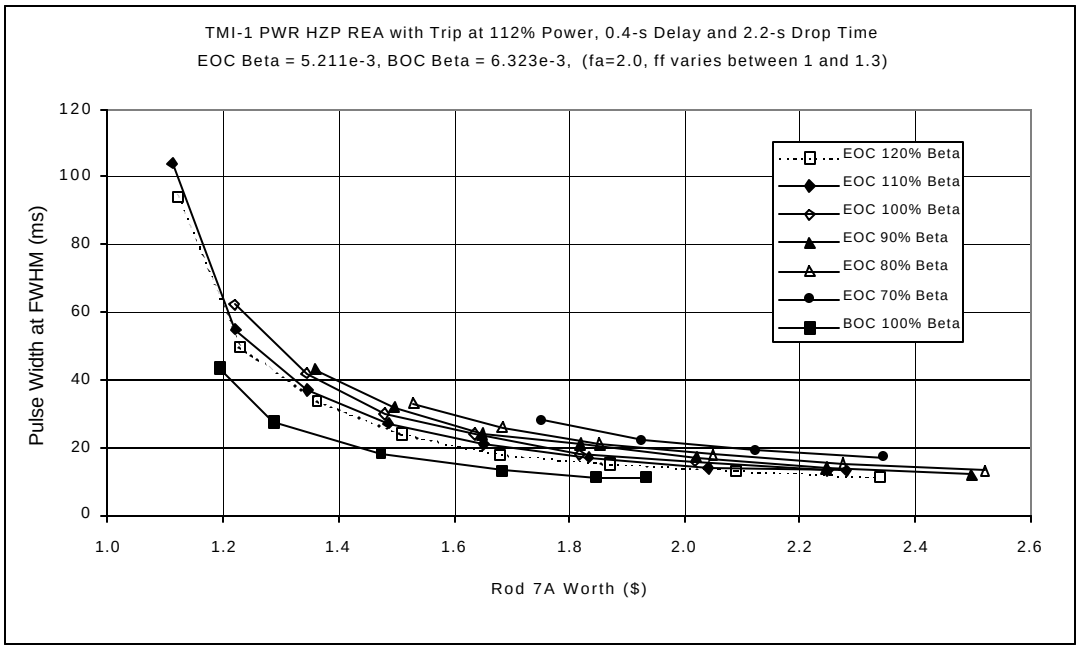

Figure 3: Pulse Width at FWHM for REA at HZP in TMI-1 PWR

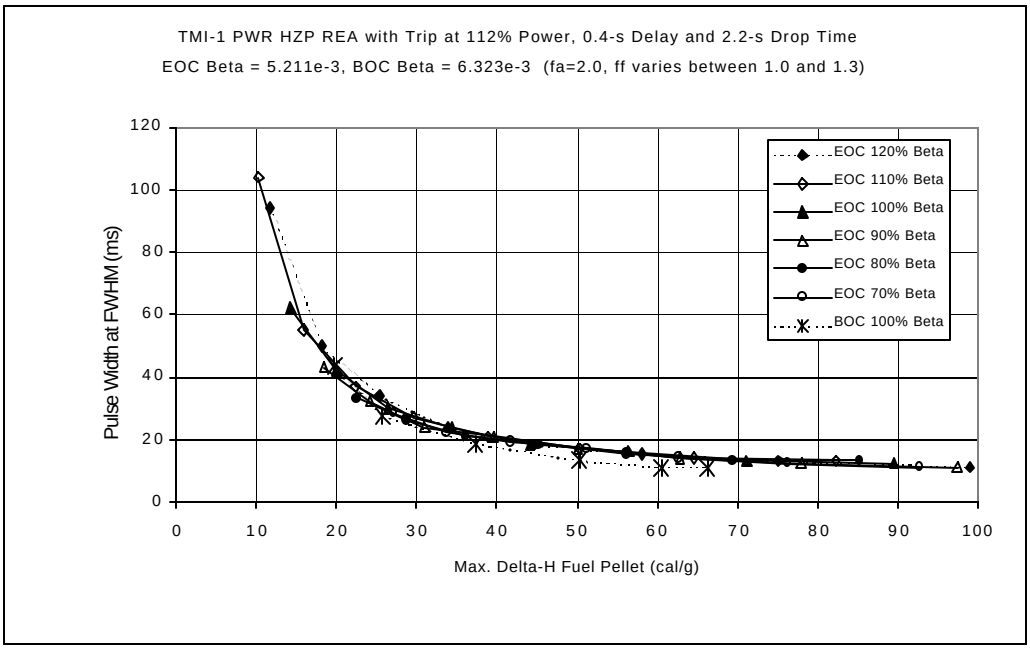

Figure 4: Pulse Width at FWHM for REA at HZP in TMI-1 PWR

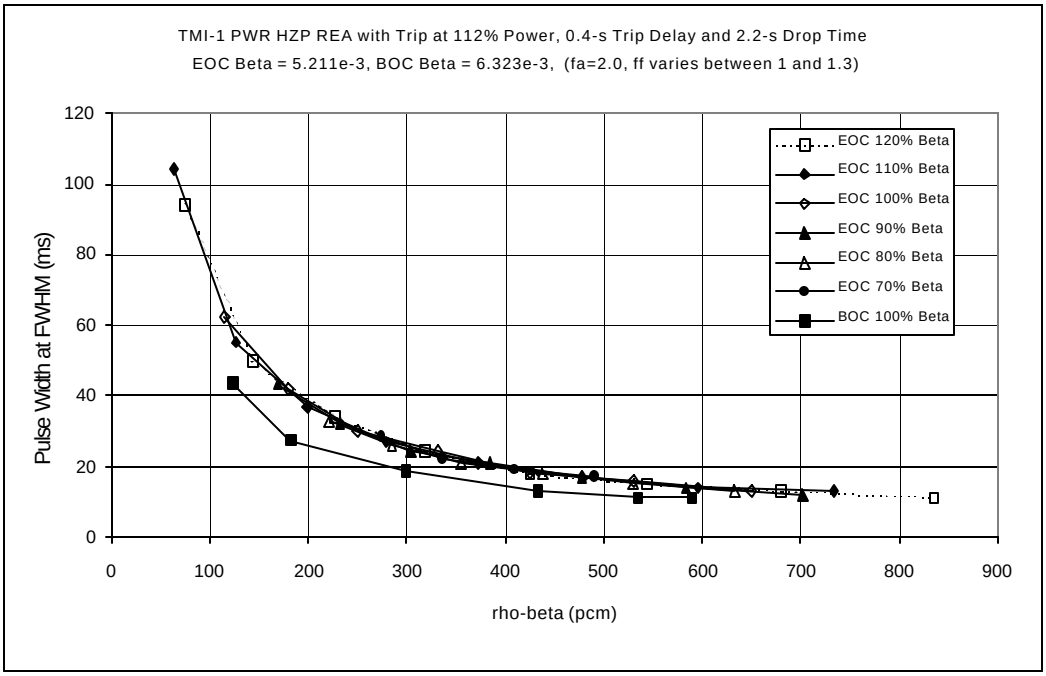

Figure 5: Pulse Width at FWHM for REA at HZP in TMI-1 PWR

Table 1: Comparison of BNL and RRC-KI Calculations for Pulse Width in REA for TMI-1 Core

Research Laboratory	BNL	RRC-KI
Computer Code Model	PARCS-SA	BARS/RELAP-5
Fuel Cycle Burn-up	EOC	EOC
Ejected Rod Worth ($)	1.223	1.209
Delayed Neutron Fraction	5.211e-3	5.211e-3
Pellet Power Shape	Uniform	Uniform
% Energy in Moderator	0.0	0.0
Trip	No Trip	No Trip
Peak Power (%)	**387**	**391**
Pulse Width (ms)	**63**	**62.6**
Max. Delta-H (cal/g)	16.8	NA
Delta-H at 0.785 s (cal/g)	14.7	17.5
Delta-H at 2.5 s (cal/g)	16.6	20.6

4. Conclusions

The PARCS Stand-alone code was used to carry out combined neutronics / thermal-hydraulic, transient three-dimensional simulations of rod ejection accidents in a model of the core of the TMI-1 PWR starting at hot zero power (HZP) at both end-of-cycle (EOC) and beginning-of-cycle (BOC). A series of parametric studies were performed in which the worth of the central control rod was artificially changed by multiplication factors on the absorption and fission yield cross sections. This procedure permitted control rod worths in excess of prompt critical ($\rho > \$1.0$). Simulations were also carried out for the EOC case where the core-averaged delayed neutron fraction was artificially increased or decreased as well.

The pulse widths of the power transients in the REA simulations by PARCS were shown to have inverse relationships with both the rod worth and maximum fuel pellet enthalpy. In congruence to the higher fuel pellet enthalpy rise at BOC, the pulse width at BOC was shorter than at EOC for a given rod worth due to the higher fissile fuel content. For rod worths varying between $1.2 and $2.2, the pulse width varied between 70 ms and 10 ms.

A sample comparison of the PARCS Stand-alone REA calculations done at BNL with the BARS/RELAP-5 calculations done by Russian colleagues at the Kurchatov Institute for the same TMI-1 PWR core model shows good agreement for the pulse width and maximum power for a given control rod worth. The PARCS calculation for the maximum fuel enthalpy increase is less than the value computed by BARS/RELAP-5 by about 20%. This discrepancy might be explained by the different methodologies for solving the neutron flux and power distributions, and in particular the fuel pellet temperature distributions.

If future experimental tests are performed to evaluate the failure limits of PWR fuel pins at various levels of burn-up in the event of an REA or any similar reactor transient, then it is suggested that these experimental tests should have pulse widths that are similar to those observed in the present set of PARCS simulation studies for a given maximum fuel pellet enthalpy rise. For example, an experimental test for an fuel pellet enthalpy change of 40 cal/g should have a pulse width of approximately 17 to 20 ms.

5. References

[1] Joo, Han G., et al., "PARCS: A Multi-Dimensional Two-Group Reactor Kinetics Code Based on the Non-linear Analytic Nodal Method," PU/NE-98-26, Purdue University, School of Nuclear Engineering, September, (1998).

[2] Ivanov, Kostadin N., et al., "PWR Main Steam Line Break (MSLB) Benchmark; Volume I: Final Specifications," NEA/NSC/DOC(99)8, U.S. Nuclear Regulatory Commission and OECD Nuclear Energy Agency, April (1999).

[3] Avvakumov, Alexander V., Private Communication to David Diamond, July 4, (2001).