Analysis of a rod withdrawal accident in a BWR with the neutronic-thermalhydraulic coupled code TRAC-BF1/VALKIN and TRACE/PARCS

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Abstract

The control rod withdrawal accident at hot zero power (HZP) is characterized by a single rod withdrawal from a core position with high reactivity worth, starting at criticality with a very low power level. The evolution consists basically of a continuous reactivity insertion. The main factor limiting the consequences of the accident is a mixed void-Doppler feedback in BWR. The peak power occurs while important power distribution changes take place in the core and also the rod extraction continues. To check the performance of the coupled codes TRAC-BF1/VALKIN and TRACE/PARCS against complex 3D neutronic transients, a rod withdrawal accident in COFRENTES NPP is simulated. This transient is a dynamically complex event, where neutron kinetics is coupled with thermal hydraulics in the reactor primary system, and reactor variables change very rapidly. TRAC-BF1/VALKIN code uses the best estimate TRAC-BF1 code to give account of the heat transfer and thermalhydraulic processes, and a 3D neutronic module. This module has two options, MODKIN that makes use of a modal method based on the assumption that the neutronic flux can be approximately expanded in terms of the dominant lambda modes associated with a static configuration of the core, and the NOKIN option that uses a one-step backward discretization of the neutron diffusion equation. TRACE is a code to study also transients in LWR reactors. This code used as a neutronic module the PARCS code.

KEYWORDS: 3D Neutronics/Thermalhydraulics, Reactivity Initiated Accident, Rod Withdrawal Accident, TRAC-BF1/VALKIN, TRACE/PARCS, 3D Coupled Codes

1. Introduction

Nuclear industries and licensing authorities need to be able to rely on the good performance of methods and computer programs used in safety analysis calculations. This is best achieved through validation and benchmarking.

With the implementation of advanced fuel management, margins to safety and licensing limits were frequently reduced. This lead to general development of advanced methods that reduce the

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level of conservation by implementing kinetics methods that more accurately capture spatial effects occurring during reactor transients. The performance of numerical methods need to be established over a realistic range of applications.

The simulated control rod withdrawal accident at zero power (HZP) is driven by the central control rod withdrawal from a core position with high reactivity worth, starting at criticality with a very low power level (10^{-9} times nominal value).The control rod bank are withdrawn, at 1 m/s. The evolution consists of a continuous reactivity insertion. The main factor limiting the consequences of the accident is the void feedback reactivity and the Doppler effect. The peak power occurs while important power distribution changes take place in the core and also while the rod extraction continues. In this accident, the maximum power is less important than its time integral. If the reactivity insertion rate is low, the heating of the fuel may be sufficient to have Doppler antireactivity, balancing the inserted reactivity while the power level is still under the trip level.

TRAC-BF1/VALKIN code [5] is a new time domain analysis code to study transients in LWR reactors. This code uses the best estimate code TRAC-BF1 to give account of the heat transfer and thermal-hydraulics processes, and a 3D neutronic module. This module has two options, the MODKIN option that makes use of a modal method based on the assumption that the neutronic flux can be approximately expanded in terms of the dominant lambda modes associated with a static configuration of the reactor core, and the NOKIN option that uses a one-step backward discretization of the neutron diffusion equation. The lambda modes are obtained using the Implicit Restarted Arnoldi approach or the Jacobi-Davidson algorithm. TRACE/PARCS is a code to study also transients in LWR reactors. This code used as a neutronic module the PARCS code (Purdue Advanced Reactor Core Simulator). PARCS is a three-dimensional reactor core simulator which solves the time-dependent, two-group neutron diffusion equation to predict the dynamic response of the reactor to external perturbations such as control rod movements or inlet coolant condition changes.

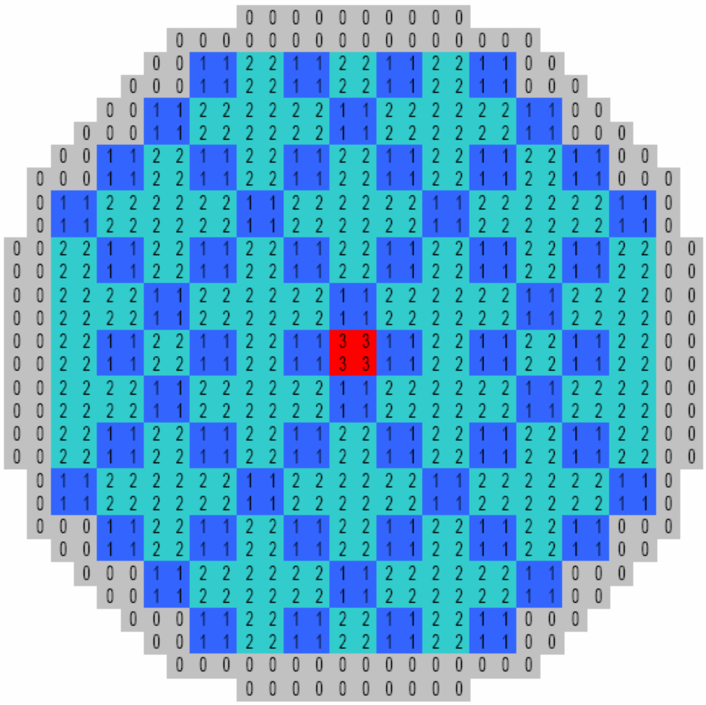
To check the performance of these coupled codes against complex 3D neutronic transients, using the cross-sections tables generated with the SIMTAB methodology [4] from SIMULATE to TRAC-BF1/VALKIN and PARCS, a rod withdrawal accident is simulated in Cofrentes NPP. The initial steady state of the reactor is a HZP, where the fuel temperature is 563.7 K and the moderator density is 731 kg/m^3. The initial power corresponds to 28.9 w, the nominal power is 3237 Mw. In this case, the transient is started by a fall of the central group of control rods, from 0 notches to up to 12 of 48 notches. The consequence of this failure is a rapid reactivity insertion along with an adverse core power distribution.

Cofrentes NPP is a General Electric BWR-6 design located in Valencia (Spain). It is in operation since 1985 and currently in its fifteenth cycle, loaded with a 624 fuel assembly mixed core. Owned and operated by Iberdrola, the original rated power has been successively increased up to 112%.

2. Description of the scenario

The initial steady state is a HZP where the moderator temperature is 563.71 K and the initial density is 731.15 kg/m^3 and the reactor power is 28.9 w. The fixed thermal-hydraulic variables should be equally distributed through the whole core. The transient is started by the fall of the central control rod. There are 177 control rods that can be grouped according to their initial insertion degree. Fig. 1 shows the initial position of the 3 assigned groups. Groups 2 and 3 are fully inserted at the initial time step, and group 1 (central control rod) is fully extracted. During the transient, group 3 is extracted 12 notches at 1 m/s.

Figure 1: Control rod groups assignment**.**



The simulations are made by a coupled 3-D kinetics/core thermal-hydraulic boundary conditions model. The developed TRACE and TRAC-BF1 model was built based on different thermal-hydraulic components. The core is modeled by 14 channels, bottom and top boundary conditions are specified in this model using the FILL and BREAK components, Fig. 2. The 14 thermal-hydraulic channels are coupled to the neutronic model according to the radial distribution shown in Fig. 3.

Radially, the core is divided into cells 15.24 cm, each corresponding to one fuel assembly plus a radial reflector. There are 624 fuel assemblies and 116 reflector assemblies. Axially, the core is divided into 27 layers (25 fuel layers plus top and bottom reflector) with a height of 15.24 cm each, being the total active core height 381 cm. There are 76 different assembly types. For the neutronic model, fuel assemblies are considered homogeneous having 1878 different compositions. The set of 3D cross sections are obtained from tables generated with the translator SIMTAB from SIMULATE to VALKIN code.

3. Numerical results and discussion

3.1 Steady state results

Initial steady state have been simulated with TRAC-BF1/VALKIN, TRACE/PARCS and SIMULATE. Tab. 1 shows the value of the effective multiplication factor for the different simulations.

In Fig. 4 we can see the reactor axial power distribution in relative power density. The steady state values are in good agreement with all the results obtained using the three codes.

Figure 2: TRACE and TRAC-BF1 model**.**

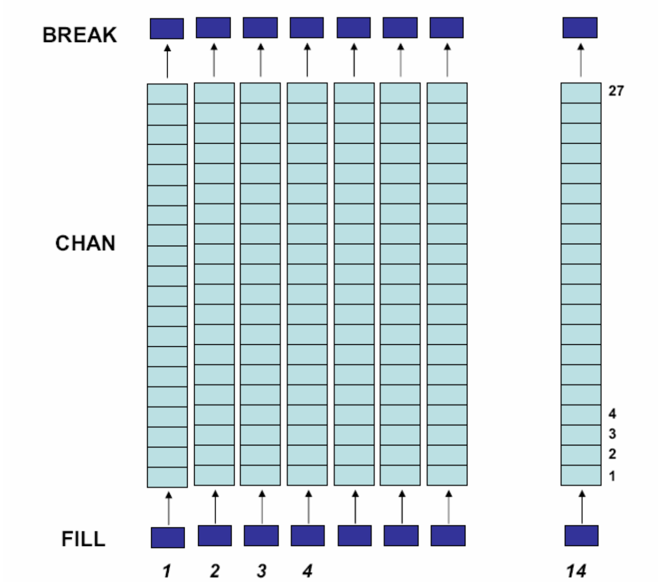


Figure 3: Radial channel mapping**.**

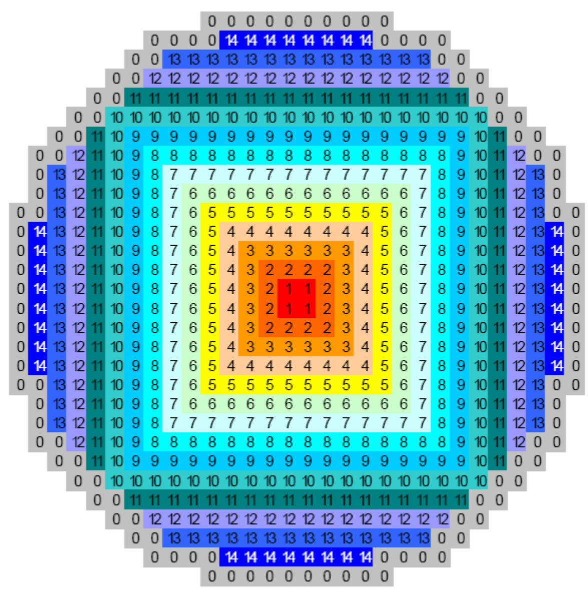
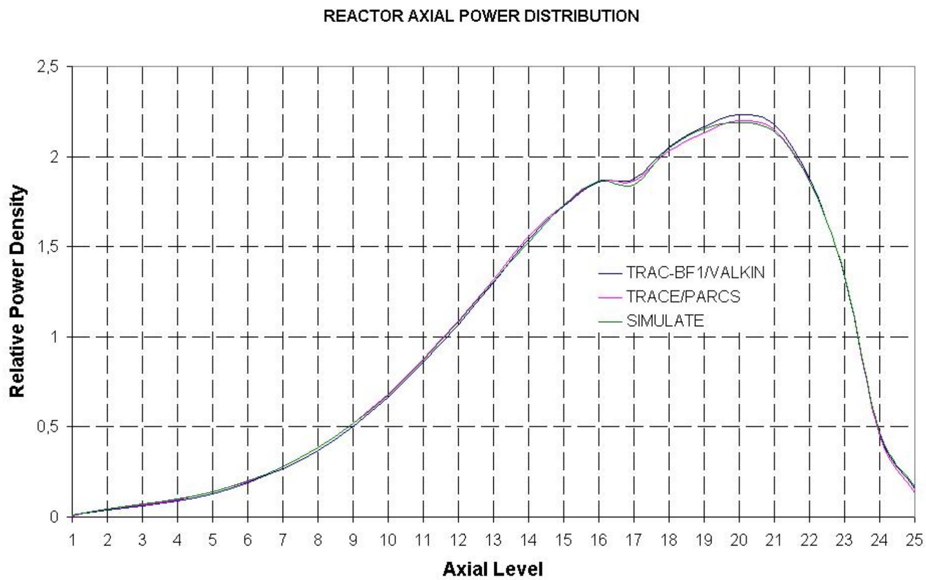


Table 1: Effective multiplication factor.

HZP	SIMULATE	TRAC-BF1/VALKIN	TRACE/PARCS
k_{eff}	0.9897	0.9824	0.9812

Figure 4: Steady state axial power profile.

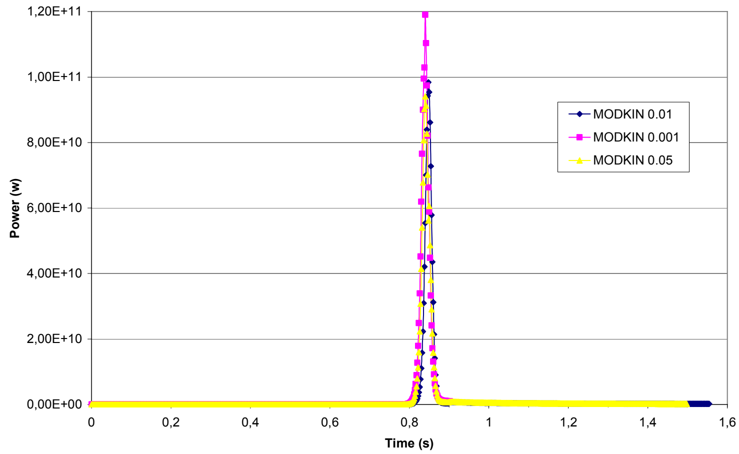


2.2 Transient results

From a subcritical initial state at zero power (28.9 w), the control rod bank 3 is withdrawn in 0.916 seconds. Basically the evolution consists of a continuous reactivity insertion. With a higher reactivity insertion rate, the transient produces a fast power burns. If the reactivity insertion rate is low, the heating of the fuel may be sufficient to have Doppler antireactivity effect, balancing the inserted reactivity. The consequence of this transient is a rapid reactivity insertion along with an adverse core power distribution. If this transient takes place, a fuel rod thermal transient that could cause departure from nucleate boiling could also occur, together with limited fuel damage. The transient is terminated by the Doppler effect caused by the increased fuel temperature but this occurs before conditions are reached that can be dangerous for the nuclear power plant safety. The transient with TRAC-BF1/VALKIN have been simulated using the MODKIN module with different updating times. The power peak depends highly on the updating time. This is because the simulated transient produces a strong variation of the thermalhydraulic and neutronic parameters, and these parameters must be updated as much as possible.

The results obtained with the MODKIN module are calculated using only the fundamental mode, and the diffusion constants and cross-sections are updated every 0.01 s, 0.05 s or 0.1 s (Fig. 5). The results differ and also the time of the power peak depending on the update time. The time of maximum fission power peak are very similar, but MODKIN with larger update time is always in advance, by a few milliseconds.

Figure 5: Reactor power evolution with TRAC/BF-1/VALKIN and different updating times.



We have to notice that the Doppler feedback model of the PARCS code is slightly different of that of the VALKIN code, because the feedback fuel temperatures are calculated approximately in PARCS as

$$T_f = (1-\alpha)\cdot T_{f_c} + \alpha \cdot T_{f_s}, \qquad (1)$$

while in VALKIN this temperatures are calculated assuming a linear power distribution in the fuel region.

Regarding the phenomenology, the Doppler feedback reactivity becomes important at time 0.8 seconds, being able to balance the control rod reactivity insertion around 0.835 seconds. In this time the neutron fission power reaches the maximum peak, the void fraction feedback reactivity is very low, but after the power peak, this contribution becomes more important due to the fact that the neutronic power effectively heats the coolant, initiating the boiling of the refrigerant. This gives a substantial increment of the void feedback reactivity reducing the neutron fission power.

The results with TRAC-BF1/VALKIN and the obtained with TRACE/PARCS with the option of Finite Difference Method (FDM) and time step size of 0.001 s are very similar.

To facilitate the reading we present, Fig. 6 and Fig. 7, only results of TRACE/PARCS, analyzing and comparing the different kernel options of PARCS code (FDM versus HYBRID) and the effect of the time step size.

In Fig. 8 and 9, we show the Doppler, coolant outlet and fuel centerline temperatures, comparing the results for two time step sizes. In Fig. 10, the enthalpy rise comparison is presented. We can see that the influence time step size is very important for this type of transients. Also, the neutronics of VALKIN is more robust, but it requires more CPU time. PARCS gives different results depending on the kernel option and the time step size, providing more accurate results with the FDM option and a time step size of 0.001 s.

Finally, in Fig. 11 we show the reactivity evolution with PARCS. In the first part, the reactivity increases due to the rod withdrawal until the Doppler and void feedback balance the reactivity. At time higher than 0.9 s the void feedback becomes more and more important, being higher than the Doppler feedback at 1.4 s.

Figure 6: Reactor power evolution with TRAC/PARCS FDM and different time steps.

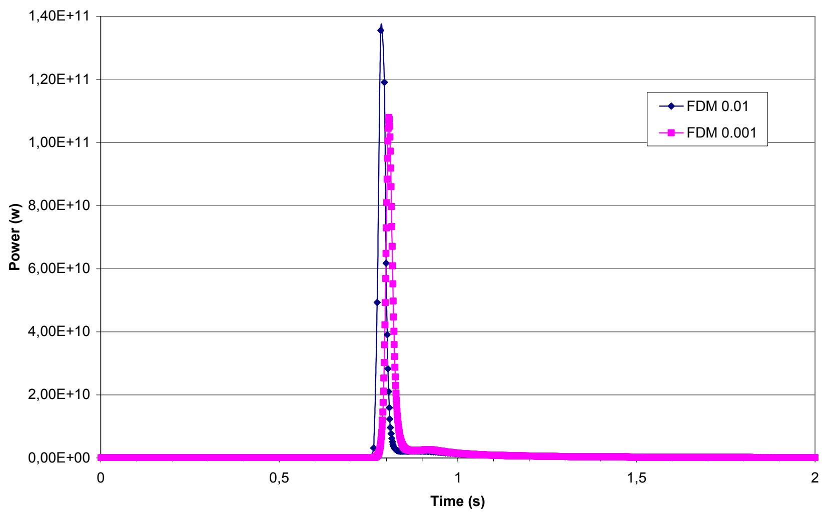


Figure 7: Reactor power evolution with TRACE/PARCS HYBRID and different time steps.

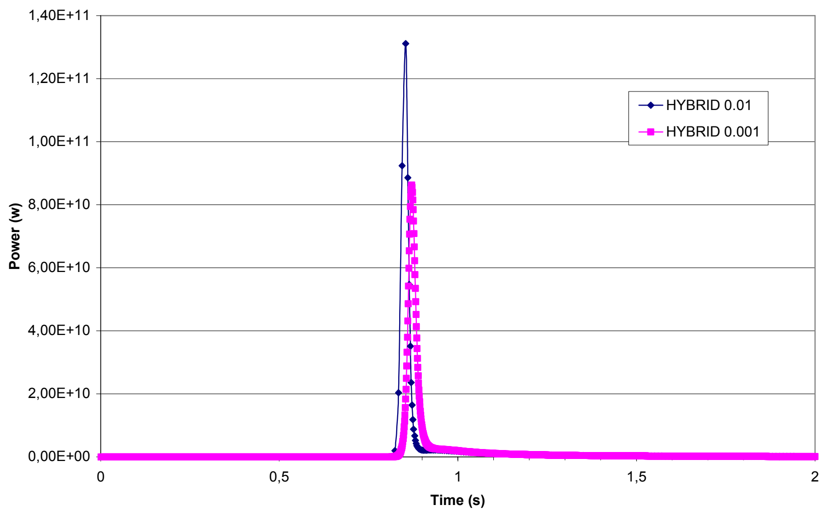


Figure 8: Temperatures with TRACE/PARCS FDM 0.01.

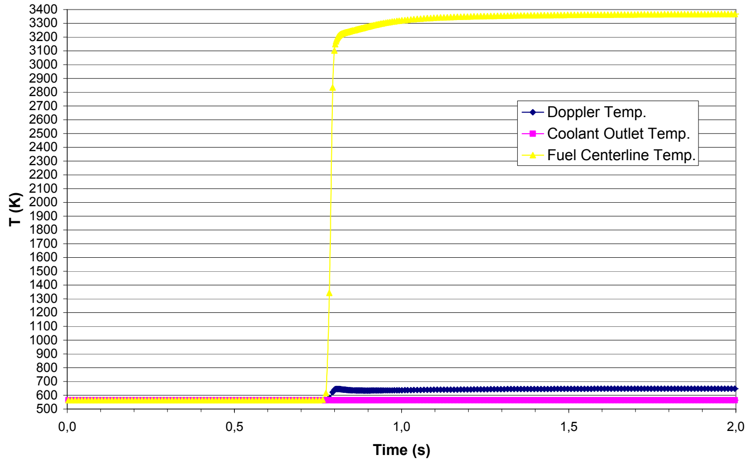


Figure 9: Temperatures with TRACE/PARCS FDM 0.001.

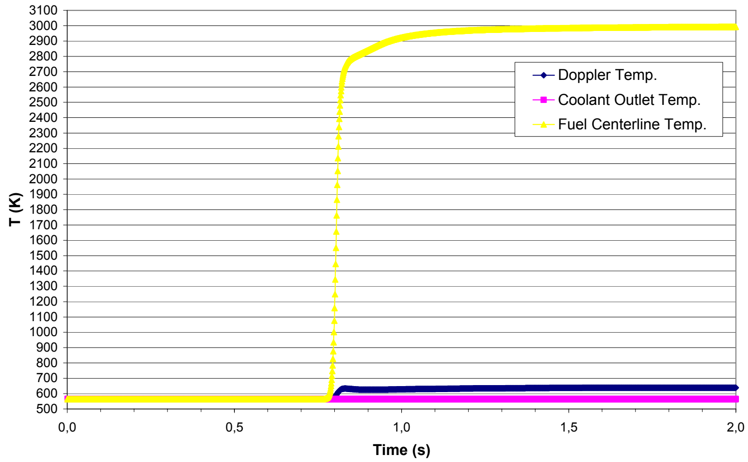


Figure 10: Enthalpy rise with TRACE/PARCS FDM.

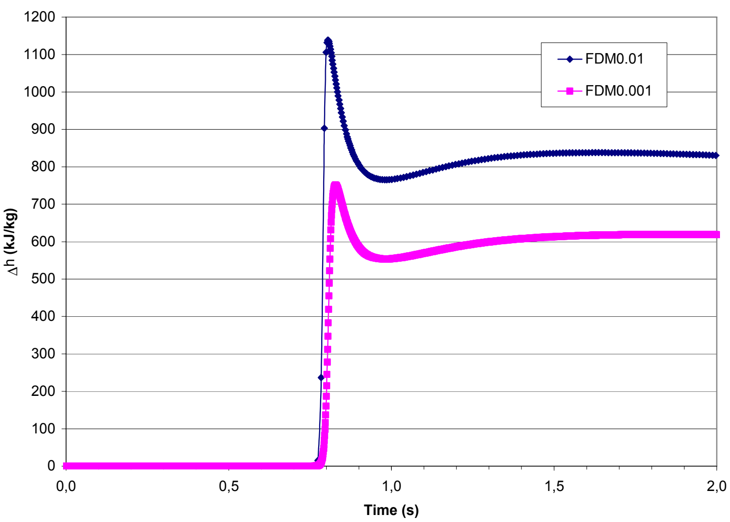
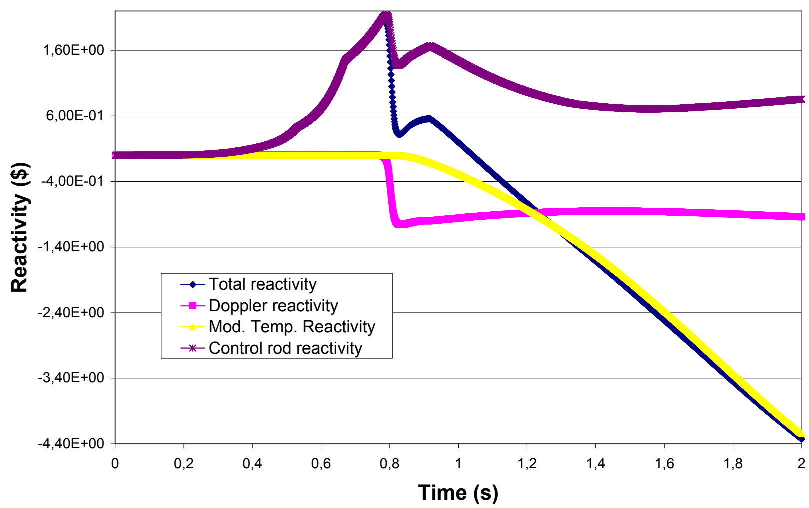


Figure 11: Reactivity with TRACE/PARCS FDM 0.001.



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