MSLB EXERCISE 2: 3-D KINETICS RESULTS WITH RELAP5/PANBOX

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ABSTRACT

This paper presents and discusses results obtained with the nuclear plant safety analysis code system RELAP5/PANBOX for the return to power scenario of Exercise 2 of the OECD/NEA Main Steam Line Break (MSLB) Benchmark. Different coupling and thermalhydraulic model scenarios have been analyzed with respect to important core parameters. Both the external and internal coupling option of RELAP5/PANBOX have been considered, i.e. the COBRA module of PANBOX was used to calculate the core thermalhydraulics in the external coupling option, whereas the core thermalhydraulics of RELAP5 was used in the internal coupling option. For the representation of thermalhydraulic channels both a fine channel geometry based on the 177 FAs as well as a coarse channel geometry based on 19 coarse channels have been investigated for the external coupling option. The 19 coarse channel model was used in the internal coupling option. In all cases, thermalhydraulic solutions where gained using both mixing between the channels and a closed channel model. The comparison of the results shows very good agreement of important core parameters between all considered variants. This demonstrates that RELAP5/PANBOX is well suited to calculate the core behavior for this MSLB transient.

1. INTRODUCTION

As a contribution to the continuing verification and validation process of the nuclear plant safety analysis code system RELAP5/PANBOX[1,2] Siemens KWU took part in the PWR Main Steam Line Break (MSLB) Benchmark[3] issued by OECD/NEA. This paper is directed to present and discuss results gained for Exercise 2 of this benchmark where the more interesting case of the return to power scenario has been investigated in detail. Different coupling and thermalhydraulic model scenarios have been analyzed with respect to important core parameters. Section 2 gives a short description of the architecture and the capabilities of the RELAP5/PANBOX system. The neutronics modelling of the reactor core for the MSLB benchmark is described in Section 3. The different coupling and thermalhydraulic model options chosen in the analyses are given in section 4 in detail. Initial and boundary conditions for the MSLB transient are described in Section 5. Results are discussed in Section 6 and final conclusions are formulated in Section 7.

2. R/P/C CODE DESCRIPTION

RELAP5/PANBOX – in short R/P/C – is a nuclear plant safety analysis code system comprised of the core simulation package PANBOX coupled directly to the best-estimate plant simulation code RELAP5. Coupling of RELAP5 has been achieved via the general RELAP5 interface package EUMOD (External User MODels). R/P/C has the capabilities of RELAP5 with added ability to calculate 3-D neutron kinetics and thermal margins with COBRA, the core thermalhydraulic module of PANBOX. External I&C models may also be linked through EUMOD to R/P/C. Control of RELAP5 variables and PANBOX control assembly position is possible. The input is simple to adapt from the standalone versions of the codes.

Three separate coupling options exist. In the first of these, COBRA boundary conditions are taken from RELAP5, and the cross section update in PANBOX is performed with COBRA thermalhydraulic data (external integration).

In the second option, no COBRA calculation is performed. The core thermalhydraulic data is passed from RELAP5 to PANBOX and this data is mapped and interpolated onto the 3-D core nodalization (internal integration). The resulting node-wise thermalhydraulic data is used to update the PANBOX cross sections, and a neutron kinetics time step is calculated. The PANBOX power distribution is collapsed as in option 1 onto the RELAP5 nodalization, and the information is passed back to RELAP5. Both options 1 and 2 have been used for the Exercise 2 calculations discussed below.

The third option is identical to option 2, except that a COBRA core thermalhydraulic solution is calculated in parallel to the RELAP5/PANBOX calculation. The COBRA calculation has no influence on the coupled RELAP5/PANBOX calculation but is useful for the calculation of conservative thermal margins.

PANBOX contains its own 1-D and point neutron kinetics models. Coefficients of these models are generated automatically and a dimensionally adaptive algorithm exists to switch between models as a transient evolves.

3. NEUTRONICS MODELLING

The neutronics core model for the MSLB benchmark is based on a radial nodalization of 1 node per fuel assembly. In axial direction 28 layers (26 in core and 2 upper and lower reflector layers) have been modelled, where the axial mesh sizes have been taken from the specification with exception of the 2 layers with height 29.76 cm which have been subdivided into 4 layers of height 14.88 cm. The decay heat model is taken over from a table provided. The results demonstrated below have been gained using the semi-analytical Nodal Expansion Method with a zero incoming current boundary condition at the outer reflector boundaries. The time discretization is based on the implicit Euler method combined with the exponential transformation technique.

4. THERMALHYDRAULICS MODELLING

Solutions for the MSLB benchmark, Exercise 2, have been generated with RELAP5/PANBOX using different thermalhydraulic model scenarios A1, A2 – C1, C2 of the coupled system described in the following:

A1 The external integration of RELAP5/PANBOX is selected, i. e. the core thermalhydraulics solution is calculated by the PANBOX thermalhydraulics module COBRA. The channel geometry is based on 1 channel/FA. Mixing between the FAs is considered using the crossflow model of COBRA.

A2 Same as A1, but the mixing between the FAs is suppressed, i. e. the closed channel model of COBRA is used.

B1 Same as A1, but the channel geometry is coarsened and based on the 18 coarse TRAC-PF1/NEM channel geometry with exception of the FAs lying on the x-axis: these FAs have been lumped together into one additional channel, i.e. 19 coarse channels have been defined at all, see Figure 1.

B2 Same as B1, but the mixing between the FAs is suppressed, i. e. the closed channel model of COBRA is used.

C1 The internal integration of RELAP5/PANBOX is selected, i. e. the core thermalhydraulics solution is calculated by RELAP5. Like in the cases B1 and B2, the channel geometry is based on the 18 coarse TRAC-PF1/NEM channels plus one additional channel for the FAs lying on the x-axis, see Figure 1. Mixing between the channels is realized using RELAP5 connections between all neighboring channels for all axial layers.

C2 Same as C1, but no mixing between the channels is considered.

The external integration scenarios A and B use 24 layers in COBRA, whereas in the internal integration scenarios C 11 layers are modeled in RELAP5.

5. INITIAL AND BOUNDARY CONDITIONS

Initially, the reactor is operated at 100% power (= 2272 MW). The reactor trip is initiated at 7.01 sec. The thermalhydraulic boundary conditions for the core (inlet mass flow, inlet liquid temperature, outlet pressure) have been provided by the benchmark team for the 18 coarse TRAC-PF1/NEM channels. These data have been appropriately rearranged for the 177 channels of variants A and the 19 coarse channels of variants B and C. In the cases of 19 coarse channels, the conditions for the channel gathering the FAs on the x-axis have been approximated by the average of the corresponding TRAC-PF1/NEM channels.

6. RESULTS

Results have been obtained for the six variants A1 to C2 described above. Relevant steady-state parameters are shown in Table I.

Table I. Comparison Between the Variants for Relevant Steady-State Parameters

Parameter	A1	A2	B1	B2	C1	C2
Eigenvalue k_{eff} (-)	1.004924	1.005087	1.006490	1.006680	1.005622	1.005753
Axial Offset (-)	-.0147	-.0075	.0185	.0269	.0063	.0237
Core Averaged Moderator Temperature (°C)	305.57	305.40	305.31	305.14	306.63	306.56
Core Averaged Moderator Density (kg/m^3)	713.96	714.39	714.75	715.17	711.96	712.17
Core Averaged Doppler Temperature (°C)	551.23	550.98	548.72	548.52	553.11	552.96

Comparing the eigenvalues for the initial hot full power (HFP) state, a good agreement between the variants can be found. For the variants B and C (using a coarse channel geometry) the eigenvalues are slightly higher than for cases A (based on a fine mesh grid). The maximum difference between the lowest and the highest value is about 170 pcm. Coolant mixing in the core does not significantly influence the steady-state eigenvalues - tendentious, the eigenvalues increase for cases with the closed channel modeling.

The axial offset is shifting from negative values for the fine channel modeling to positive values for the coarse channel calculations. It is slightly higher for the closed channel models, although the differences to the variants with open channels are low.

The comparison of core averaged thermalhydraulic conditions shows, that differences between the variants are comparatively small. The core averaged moderator temperatures for variant C (using RELAP5 thermalhydraulic conditions) are about 1 K higher than for variants A and B (using COBRA thermalhydraulic conditions). The number of axial nodes (variants A and B use 24 nodes while variant C uses 11 nodes) affects the generation of the core averaged parameter. For the core averaged Doppler temperature a difference of about 2 K can be found between the fine and the coarse channel modeling of the external coupling (variants A and B). The differences between the external and internal coupling method for the coarse channel calculations (variants B and C) is about 4 K. Again, coolant mixing in the core hardly influences the steady-state results.

During the transient coolant densities are increasing. Due to feedback effects (combined with an artificial low value for the tripped rod worth) a return to power occurs after reactor scram and a second power maximum at about 57.6 s is reached. The transient course of the event is shown in Figures 3 – 6. For all parameters, a good agreement between the six variants can be found. Some details of the total core power are presented in Table II for two different time points. T= 7 s is time point just before scram where the highest reactor powers during the transient are reached. The second time point (at t = 57.6 s) gives the total core power at the local power maximum during the return to power period.

Table II. Comparison Between the Variants for the Total Core Power at 2 Time Points

Core Power (MW)	A1	A2	B1	B2	C1	C2
at t = 7 s	3292.44	3322.69	3343.82	3344.67	3293.67	3303.56
at t = 57.6 s	856.43	841.32	818.09	802.61	859.19	858.63

The comparison of the total core power at the first maximum (before scram) shows, that reactor power is somewhat higher for the closed channel calculations. The highest values are found for variants B. To the contrary, at the second time point variants B show the lowest values and reactor powers are somewhat higher for the open channel calculations. The maximum difference between the variants is about 50 MW for both time points.

A typical 2-D power distribution at the time of maximum power during the return to power period is shown in Figure 2 for case A1. The highest values can be found in the region surrounding the stuck rod at position N12.

The core averaged thermalhydraulic conditions do not appreciable differ among each other. Obviously, the coarse channel models in variants B and C based on 19 channels are adequate to simulate the core thermalhydraulics behavior for this transient. Coolant mixing in the core is less important in these cases.

7. CONCLUSIONS

Several coupling and thermalhydraulic model scenarios have been analyzed for the return to power case of Exercise 2 of the MSLB benchmark. The comparison of the results shows very good agreement of important core parameters between all considered variants. This demonstrates that both the COBRA thermalhydraulics and the RELAP5 thermalhydraulics are appropriate to determine the relevant core parameters during a steam line break. Furthermore, a core representation using 19 coarse channels has proven to be adequate in comparison to a fine channel representation based on the 177 FAs. Thus RELAP5/PANBOX is well suited to calculate the core behavior for this MSLB transient.

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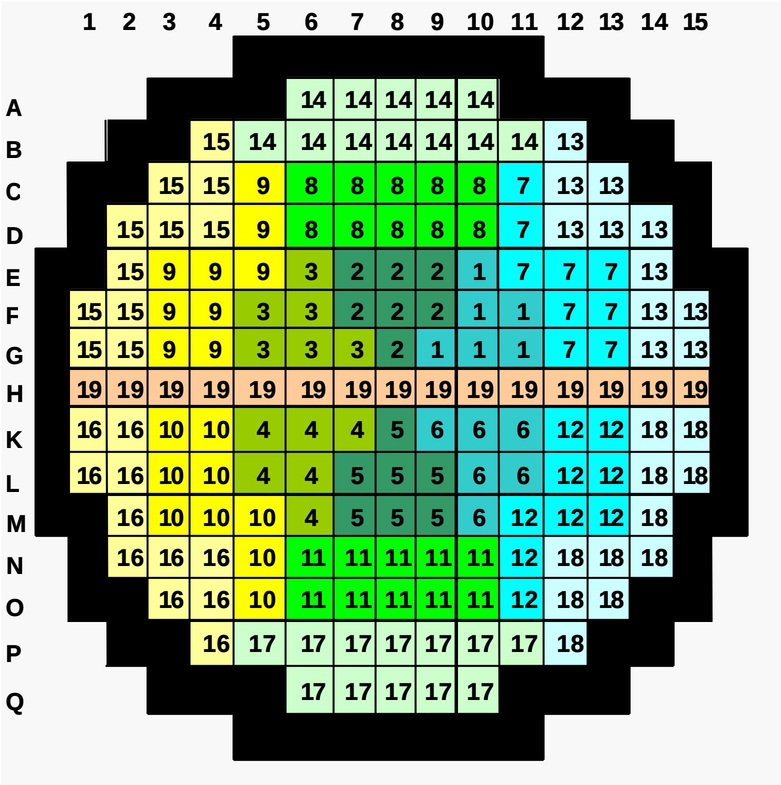

Figure 1. Core Channel Mapping

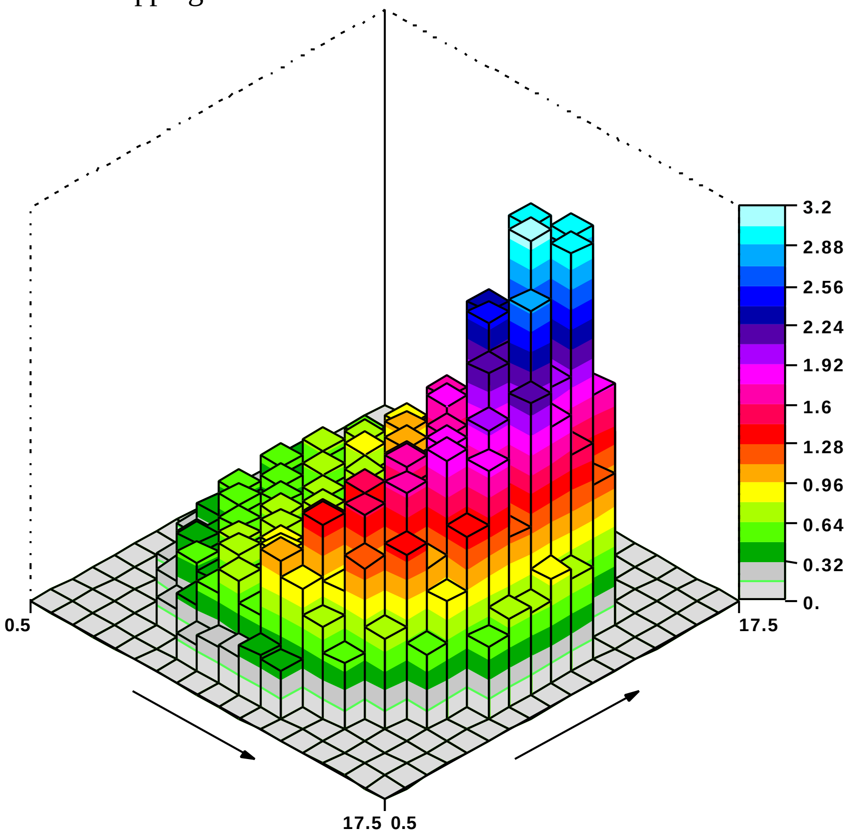

Figure 2. Normalized Radial Power Distribution

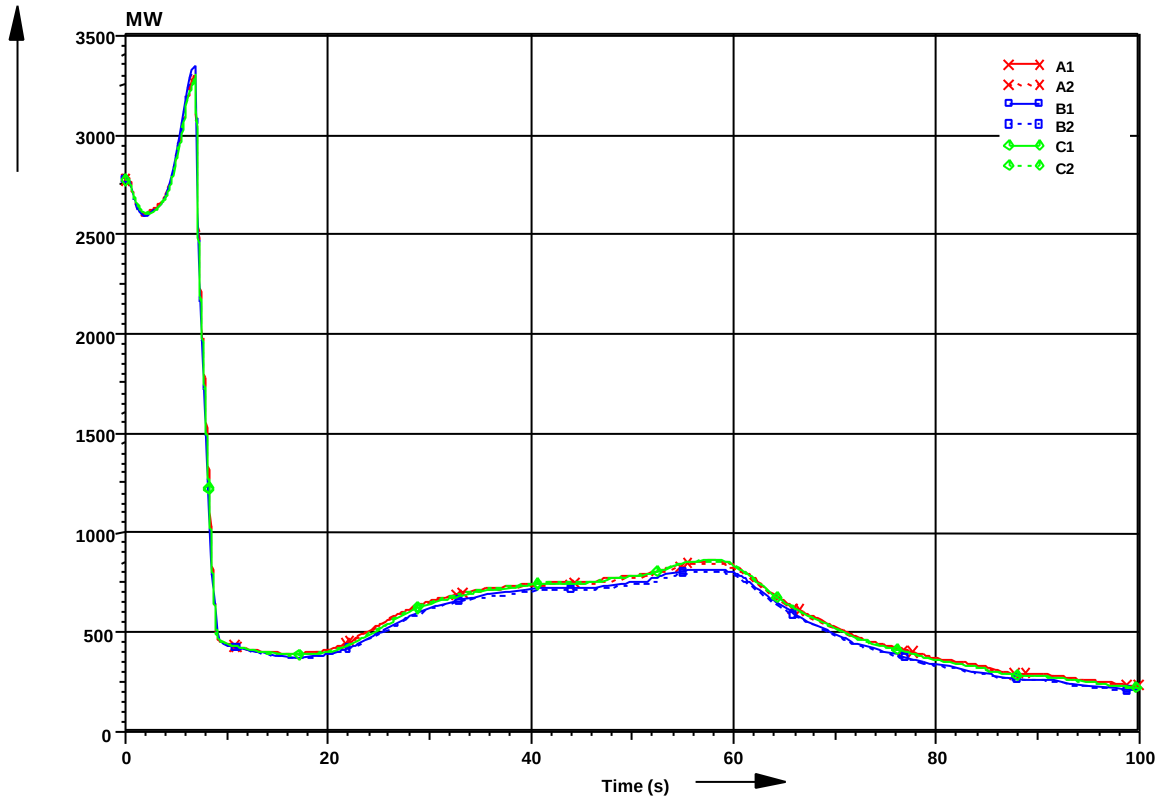

Figure 3. Total Core Power

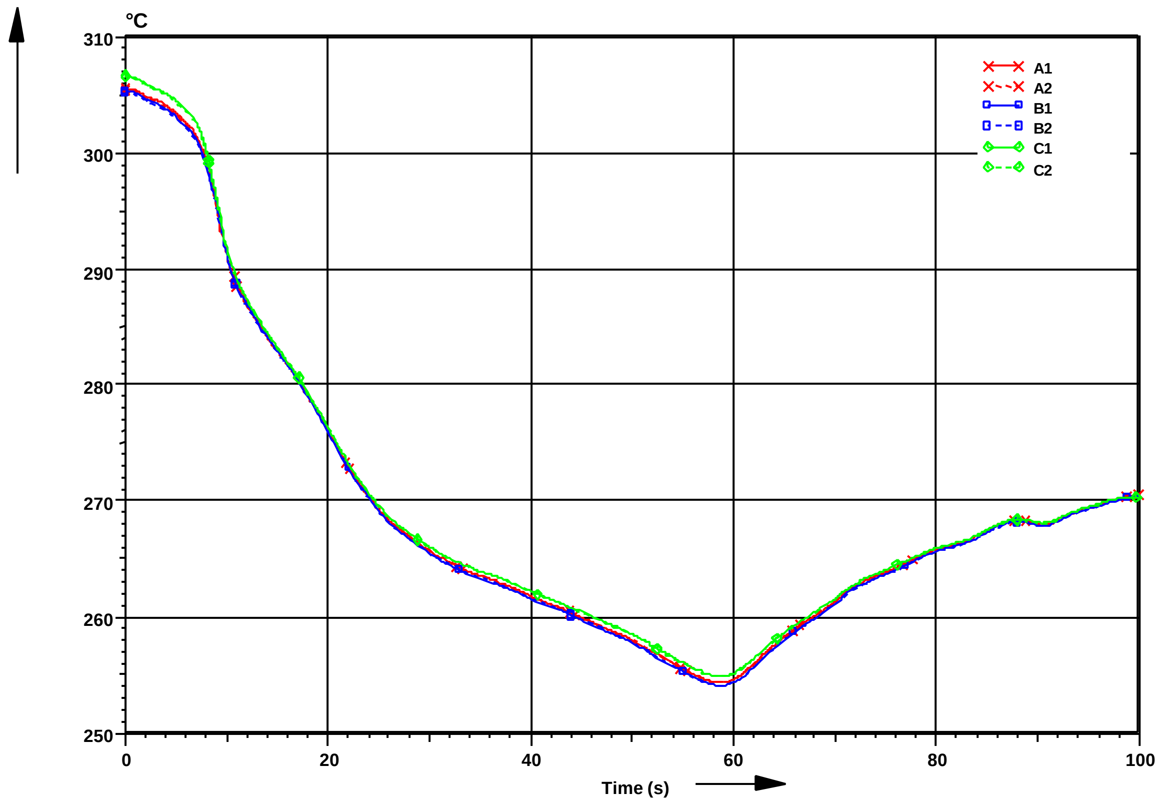

Figure 4. Core Averaged Moderator Temperature

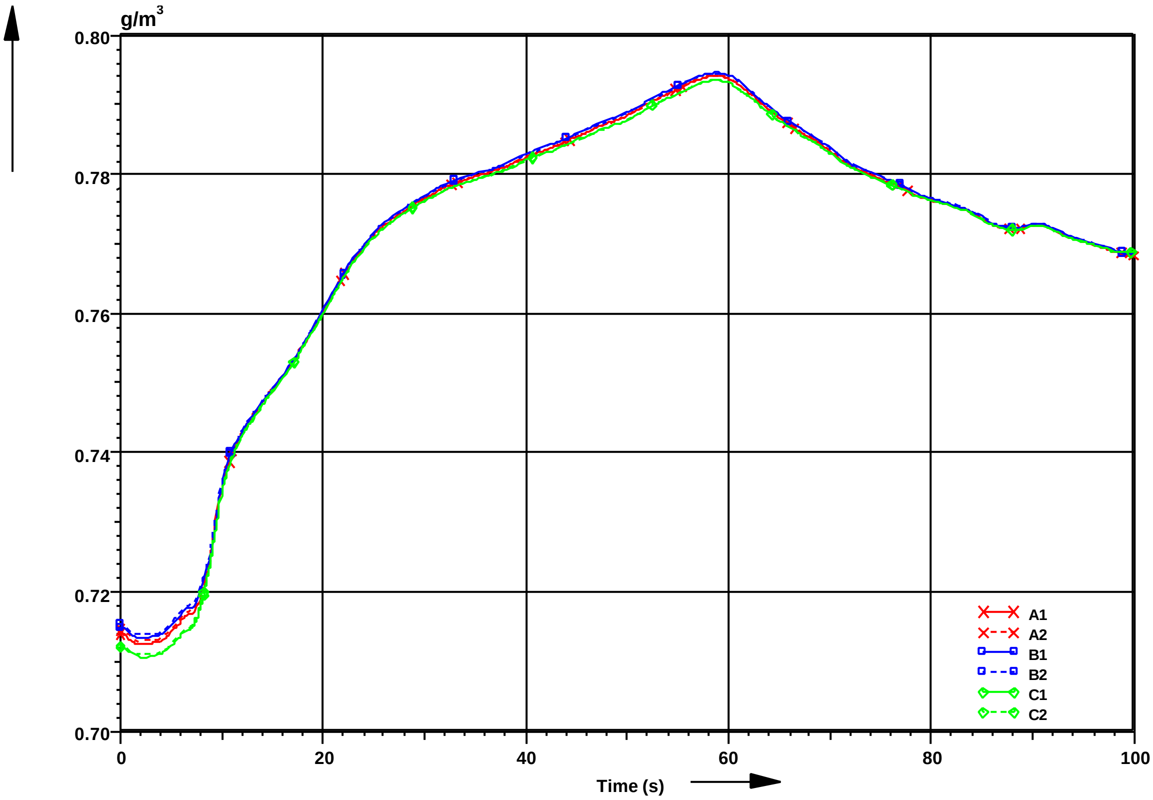

Figure 5. Core Averaged Moderator Density

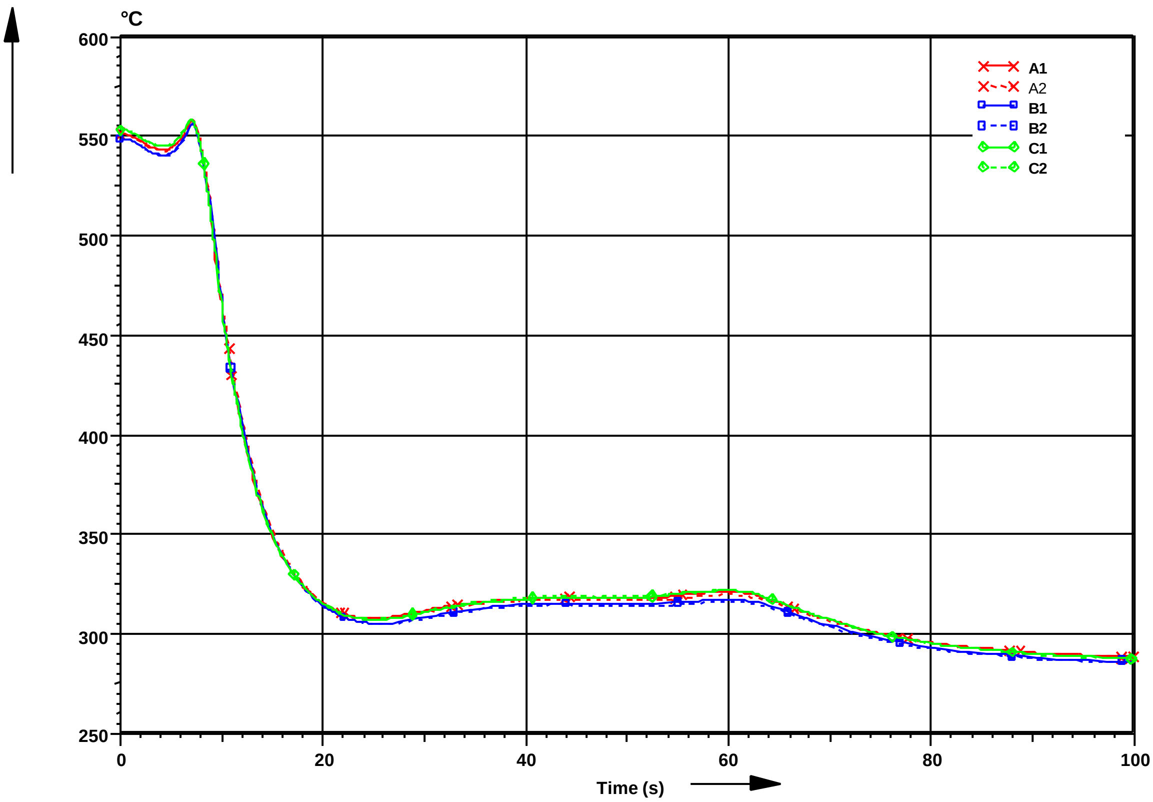

Figure 6. Core Averaged Doppler Temperature