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Analysis of Rod Withdrawal at Power (RWAP) Accident using ATHLET Mod 2.2 Cycle A and RELAP5/mod 3.3 Codes

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ABSTRACT

The system code ATHLET (Analysis of THermal-hydraulics of Leaks and Transients) is being developed by the Gesellschaft für Anlagen-und Reaktorsicherheit (GRS) mbH in Garching, Germany. In the paper, an overview of activities performed at Faculty of Electrical Engineering and Computing (FER), University of Zagreb, in application of system code ATHLET in transient analyses for NPP Krško (NEK) is presented. Newly, the NEK input deck for the released ATHLET version (Mod 2.2 Cycle A) has been developed. For that purpose, the NEK data base that has been developed and maintained at FER for the last two decades primarily for development of standard input deck for RELAP5 code was used. The ATHLET model has been validated by analyzing the Rod Withdrawal At Power (RWAP) accident at nominal power. The results for steady state calculation as well as RWAP transient were assessed against the analysis performed by RELAP5/mod 3.3 code. In both ATHLET and RELAP5 calculation, the RWAP accident was simulated by constant reactivity insertion rate equal to 2.4 pcm/sec. For ATHLET analysis, two fluid dynamic options were tested for the primary side: a) base case analysis with 5 conservation equations and mixture level model and b) two-fluid model with separate conservation equations for liquid and vapour phases for all the volumes except for the pressurizer where 5 equations+mixture level model was retained. The Steam Generators (SGs) were built using basic ATHLET elements together with the dedicated separator model. For RELAP5/mod 3.3 analysis, a standard option with thermal and mechanical non-equilibrium (6 equations) was used. The results of the steady state calculation for the ATHLET model have shown a very good agreement with RELAP5 calculation. In the transient analysis very small differences for the main physical parameters between ATHLET and RELAP5 as well as between the two ATHLET models were obtained.

1 INTRODUCTION

The system code ATHLET, ref. [4] is being developed by the GRS for the analysis of anticipated and abnormal plant transients as well as of the whole spectrum of Loss of Coolant Accidents (LOCAs) in light water reactors. The code consists of several basic modules, e.g., Thermo-fluiddynamics (TFD), Heat Transfer and Heat Conduction (HECU), Neutron Kinetics (NEUKIN) and General Control Simulation Module (GCSM). The GCSM is a block oriented simulation language for modelling of balance of plant (BOP) systems and actions, e.g., the plant control systems, trip logic, as well as the interaction between thermal-hydraulics and BOP.

In 1996, the NPP Krško input deck for ATHLET Mod 1.1 Cycle C has been developed at FER. The input deck was tested by analyzing the realistic plant event 'Main Steam Isolation Valve Closure' and the results were assessed against the measured data, ref. [5]. The input deck was established before plant modernization that took place in 2000 and included the power uprate and SG replacement. The released ATHLET version (Mod 2.2 Cycle A) is now being available at FER.

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Accordingly, the NEK input deck for ATHLET Mod 2.2 Cycle A has been developed. A completely new input deck has been created taking into account the large number of changes due to power uprate and SG replacement as well as taking advantage of developmental work on NEK data base performed at FER. The new NEK input deck for ATHLET code has been tested by analyzing the RWAP accident and the results were assessed against the analysis performed by RELAP5/mod 3.3 code.

The RWAP accident is caused by a continuous uncontrolled Rod Cluster Control Assembly (RCCA) bank withdrawal while reactor is at power. The RWAP accident can be either Departure from Nucleate Boiling (DNB) ratio or overpower limiting accident depending on initial power and reactivity insertion rate. The automatic rod control system is assumed unavailable, thus the only negative reactivity is due to Doppler and moderator feedback. Due to positive total reactivity and nuclear power increase, the core heat flux as well as transferred heat in the steam generators (SGs) rise, too. Since the steam flow to the turbine and the extracted power from the SGs remain constant, the unbalance between produced and extracted heat is established. Consequently, the secondary side pressure and primary side temperature continue to increase until either reactor trip terminates the power production or SG relief or safety valves open thus increasing the heat removal. Unless terminated by manual or automatic action, the power mismatch between primary and secondary side and the resultant coolant temperature rise could eventually result in DNB ratio and/or fuel centreline melt. In order to avoid core damage, the reactor protection system is designed to automatically terminate the transient before the DNB ratio falls below the limit value, or the fuel rod power density limit (kW/m) is reached. Depending on the initial power level and the reactivity insertion rate, the reactor protection system will actuate any of the following reactor trips: a) Power range high neutron flux, b) Power range high positive neutron flux rate, c) Overtemperature ΔT (OTΔT), d) Overpower ΔT (OPΔT), e) High pressurizer pressure, f) High pressurizer water level and g) Manual reactor trip.

The steady state was calculated for 1000 seconds for both ATHLET and RELAP5 analysis. Thereupon, the RWAP accident was simulated for 200 seconds by constant reactivity insertion rate equal to 2.4 pcm/sec.

2 CALCULATIONAL MODEL FOR NPP KRŠKO

For RELAP5/mod 3.3 calculation, the NPP Krško nodalization developed at FER, [1], [2] and [3] has been used. Here, only a brief description of the RELAP5 nodalization is provided since it has already been described in a number of papers; e.g., recently in ref. [6]. The standard RELAP5/mod 3.3 model comprises the Resistance Temperature Detector (RTD) bypass lines thus enabling the accurate representation of Narrow Range (NR) temperature measurement at the plant. However, since the NPP Krško is considering the replacing of the RTD bypass manifold system with fast-response thermowell-mounted (TW) RTDs, the latter (TW RTD) configuration was assumed in this analysis. The RTD bypass lines were removed from the model while the TW RTD temperature measurement was modeled by imposing the measurement delay and filter on calculated loop temperatures. After removing the RTD bypass lines, the RELAP5 model has 469 thermal-hydraulic nodes, 497 junctions, 378 heat structures (with 2107 mesh points), and 691 control variables, with 199 variable and 208 logical trips, respectively.

As a basis for ATHLET model, the NEK data base that has been used for development of the RELAP5 model was used. In Figure 1, a part of ATHLET nodalization for NPP Krško is presented. Due to complexity, only components of the first loop are shown; i.e., the reactor pressure vessel and the first coolant loop (hot and cold leg, reactor coolant pump and steam generator primary and secondary side) and the pressurizer. The NPP Krško model for ATHLET code has 108 Thermo-Fluiddynamic Objects (TFOs) containing 497 control volumes (CVs), 340 junctions, 355 heat slabs (with 1407 mesh points) and 398 GCSM signals. Contrary to RELAP5 model, the ATHLET nodalization does not contain the detailed models for a number of systems so far; e.g., the safety injection system and steam generator feedwater (main and auxiliary) system. For the rest of the plant, the same approach with rather comprehensive description of the plant components was applied in ATHLET model, too. The reactor pressure vessel (RPV) is represented with 18 TFOs. In particular, the lower plenum is divided into four TFOs; the reactor core (type PIPE simulating 1 coolant channel consists of 12 CVs) represents the active core (from bottom to the top of active fuel) and, parallel to the core with the same length; the empty RCCA guide tubes inside core are represented with GUIDET; the region between baffle and barrel is represented with BUFFBPSS; TFOs UPCORE1 and UPCORE2 represent the core region above top of active fuel; the RCCA guide thimbles (connecting core outlet and the upper head) is represented with volume RCCBPSS; TFOs UPPL1 and UPPL2 form the upper plenum and TFOs UPHEAD1 and UPHEAD2 represent upper head; the bypass flow path from upper head is introduced by connecting the UPPHEAD2 with upper downcomer (UPDCM) that is connected to the RPV inlet (TFO RVIN); the cold legs 1 and 2 ends are connected to RVIN that is further led to lower downcomer first part (LWDCM1) which is connected to the rest of the downcomer (PIPE LWDCM2 having 14 CVs) that is finally connected to the lower plenum (TFO LWPL2); the hot leg's (HOTLEG1, HOTLEG2) inlets are connected to the upper plenum (UPPL1); the bypass flow path between the lower downcomer (LWDCM2) and TFO BUFFBPSS is also considered. The pressurizer is connected to the hot leg 1 via surge line whereas the both cold legs are connected to the top of the pressurizer via pressurizer spray lines. The pressurizer has been built using the standard ATHLET elements (TFOs, Heat Conduction Objects -HCOs and GCSM signals). Realistic models for pressurizer relief and safety valves with loop seal have been developed.

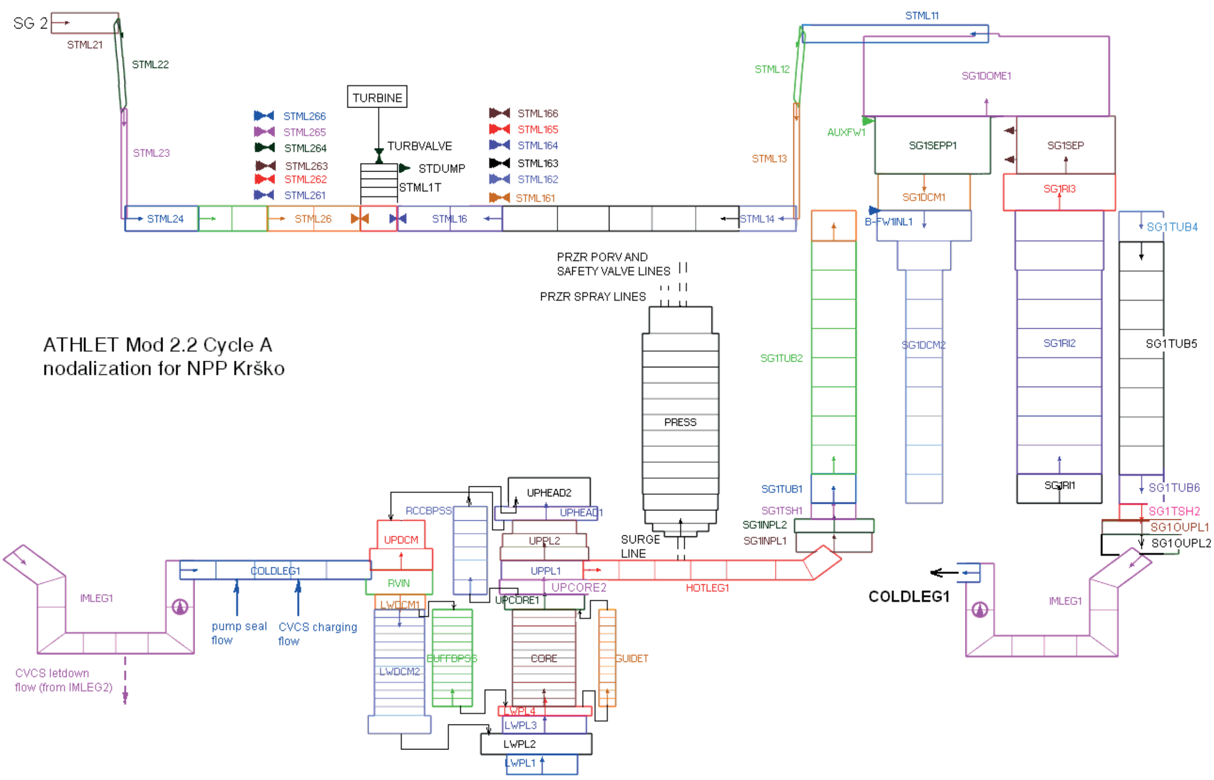
For modeling of the steam generator, the flexible U-tube SG model built out of standard ATHLET elements together with the dedicated separator model were used. Contrary to RELAP5 model, the detailed models for steam generator main and auxiliary feedwater with realistic control valves are not included in the ATHLET model. Instead, the Single Junction Pipes (SJP) with FILL components are used (B-FWINL1 and AUXFW1 in Figure 1). The steam lines are modeled with their physical lengths (the lengths are different for loop 1 and 2). Also, each steam line is equipped with 1 SG PORV and 5 safety valves that discharge into containment (modeled as time dependent volume). The steam lines are connected via respective main steam isolation valves to the steam header line (STML1T). The steam header line is further connected via the control valve to the turbine that is modeled as time dependent volume thus representing the boundary condition on the secondary side. The simple steam dump model was applied with SJP flow calculated in the steam dump GCSM control block. The ATHLET Mod 2.2 Cycle A offers the possibility of choosing between two different models for the simulation of fluid-dynamics: 1) 5-equation model, with separate conservation equations for liquid and vapour mass and energy, and a mixture momentum equation, and including a mixture level tracking capability, and 2) two-fluid model – 6 conservation equations, with separate conservation equations for liquid and vapour mass, energy, and momentum (without mixture level tracking capability). In the analysis presented in this paper two fluid dynamic options have been tested for the TFOs on the primary side: a) base case analysis with 5 conservation equations+mixture level option (IARTO=1 in input) and b) two-fluid model (IARTO=2) with separate conservation equations for liquid and vapor mass, energy and momentum (without mixture level tracking capability) for all the volumes except for the pressurizer where 5 equations+mixture level option was chosen. On the secondary side, only 5 equations + mixture level model was applied. This is the only available option for dedicated separator model (SG1SEP), which on the other side requires the mixture level model that is not allowed in six equation model for the associated volumes to the separator (downcomer and steam dome).

For nuclear heat generation the point kinetics model was selected. The Cycle 24 Beginning Of Life (BOL) data prepared for RELAP5 point kinetics input were adapted for ATHLET model. Both Doppler and moderator density feedback were taken into account. The detailed models of reactor protection as well as plant control systems have been modeled using GCSM module. They include: 1) Reactor protection system, 2) Turbine trip logic, 3) Pressurizer pressure and level control system, 4) Steam generator level control system and 5) Steam dump system.

The initial conditions for RWAP analysis were obtained after 1000 sec steady state calculation for both ATHLET and RELAP5 code. The results of the steady state calculation at nominal power are summarized in Table 1. The 1000 sec steady state calculation took 218 and 117 sec CPU time for RELAP5 and ATHLET code, respectively. Thereby, the maximum time step chosen for steady state calculation was 0.1 sec for RELAP5 and 0.5 sec for ATHLET, respectively.

Table 1 Initial conditions for RWAP analysis (results after 1000 seconds of steady state calculation)

Parameter	Unit	ATHLET	RELAP5
Reactor power	MW	1994	1994
SG power	MW	1001.23/1001.89	996.56/1002.5
Pressurizer pressure	MPa	15.5125	15.5126
SG steam dome pressure	MPa	6.43/6.43	6.42/6.4
RCS average temperature	K	578.16/578.15	578.15/578.056
RCS volumetric flow	m^3/s	6.26/6.26	6.26/6.26
SG steam mass flow rate	kg/s	545.94/543.25	541.28/544.41
SG feedwater mass flow rate	kg/s	545.94/543.25	541.28/544.42
Feedwater enthalpy	kJ/kg	942	943
SG circulation ratio	-	3.73/3.74	3.74/3.72
Pressurizer level	%	55.6	55.7
SG level	%	69.3/69.3	69.3/69.3



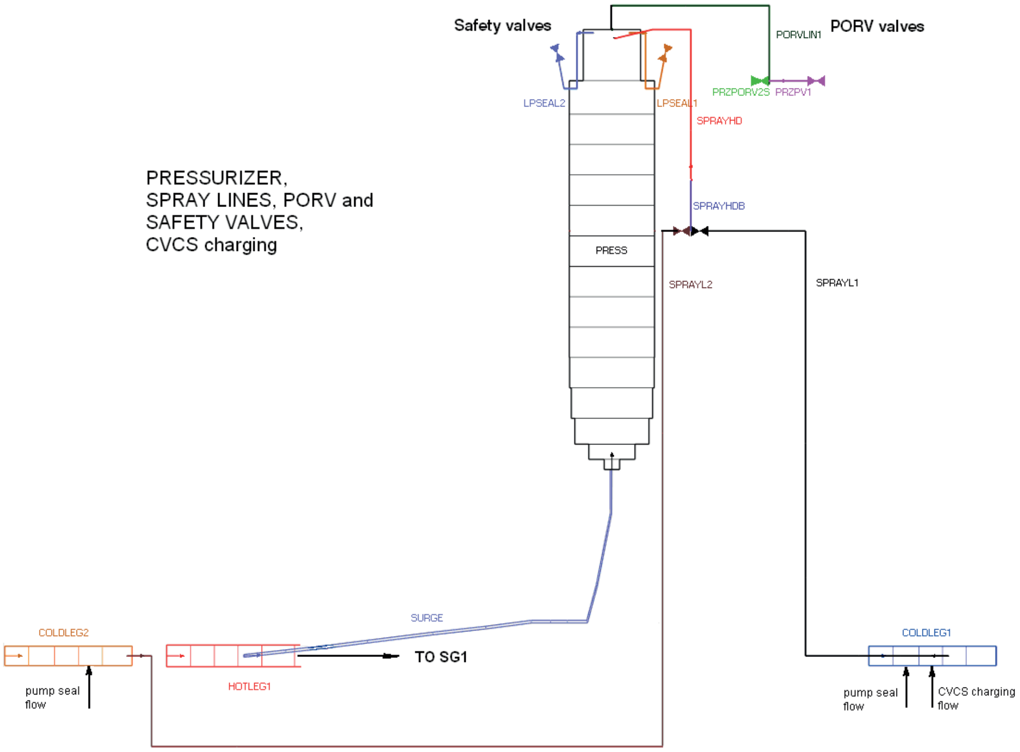


Figure 1: ATHLET Mod 2.2 Cycle A nodalization of NPP Krško

2.1 Reactor Trip Functions during RWAP

In order to avoid core damage because of RWAP, reactor protection system will actuate any of the following reactor trips: a) Power range high neutron flux, b) Power range high positive neutron flux rate, c) Overtemperature ΔT (OTΔT), d) Overpower ΔT (OPΔT), e) High pressurizer pressure, f) High pressurizer water level and g) Manual reactor trip. At fast reactivity insertion rates, a steep nuclear power rise will result in actuation of high neutron flux or high neutron flux rate trip (trips a) and b)) before the RCS temperatures rise significantly to actuate trips c) through f). The nuclear power increase rate is being reduced during the transient due to negative reactivity feedback (Doppler and moderator density). For slow reactivity insertion rates, the increase of core heat power and moderator temperature may significantly reduce the increase rate of nuclear power. Nuclear power may even start to decrease while the coolant temperatures are still increasing. Such scenario may prolong the transient and result in a significant increase of coolant temperature thus compromising the margin to DNBR.

In order to protect the reactor from excessive core heat power increase and low DNBR, the OPΔT and OTΔT are generated. Thereby, for larger reactivity insertion rate, the OPΔT trip is expected to be actuated before the OTΔT trip. In this paper, the relatively slow reactivity insertion rate (2.4 pcm/sec) was analyzed and the OTΔT trip is actuated first.

At the plant, the setpoint for OTΔT trip is continuously calculated by solving the following equation:

$$OT\Delta T_{setp} = \Delta T_0\left[K_1 - K_2\left(\frac{1+\tau_1 s}{1+\tau_2 s}\right)\left(T_{avg}\frac{1}{1+\tau_7 s} - T^0_{avg}\right) + K_3\left(P - P_0\right) - f\left(\Delta\Phi\right)\right] \quad (1)$$

Where:

$OT\Delta T_{setp}$ - Overtemperature ΔT setpoint

ΔT_0 - Indicated ΔT at nominal thermal power

T_{avg}, T_{avg}^0 - Measured and indicated loop average temperature at nominal thermal power

P, P_0 - Measured and nominal pressurizer pressure

K_1 - Setpoint bias

K_2, K_3 - Constants based on the effect of temperature (K_2) and pressure (K_3) on the DNB limits

τ_1, τ_2 - Time constants (s) of dynamic signal compensators (lead-lag)

τ_7 - Time constant (s) of dynamic signal compensator (lag)

s - Laplace transform variable (s^{-1})

$f(\Delta\Phi)$ - A function of the neutron flux difference between the upper and the lower long ion chambers (DNBR is lower for non-uniform axial power distribution and for power shifted in the upper part of the core).

The calculated $OT\Delta T_{setp}$ is compared with two sets of loop temperature difference measurements (ΔT) per loop. The OTΔT reactor trip function will trip the reactor on coincidence of two out of four signals satisfying the condition below:

$$OT\Delta T_{setp} \leq \Delta T\,,$$

where ΔT is the compensated lead-lag ($\frac{1+\tau_4 s}{1+\tau_5 s}$) signal of the measured difference between the indicated hot and cold leg temperatures (as determined by RTD thermowell instrumentation). The model is implemented in NPP Krško nodalization for both RELAP5 and ATHLET codes.

3 TRANSIENT RESULTS

The transient was initiated after 1000 sec steady state calculation by inserting the positive reactivity at a constant rate (2.4 pcm/sec). The main events for ATHLET and RELAP5 calculation are summarized in Table 2, and the results for main physical parameters are presented in Figure 2 through Figure 16. The insertion of positive reactivity, Figure 2, leads to an increase of nuclear power, Figure 3, that results in the core heat flux increase, that lags slightly behind the nuclear power, Figure 4. The fuel as well as coolant temperature increase result in addition of negative reactivity due to negative Doppler and moderator density reactivity coefficient. However, since the automatic rod control system is not credited, the total reactivity is positive and nuclear power increases further until reactor scram actuation. The power transferred from primary to secondary side in the steam generators, Figure 5, increases, too, due to primary temperature increase, Figure 6, whereas on the secondary side, the pressure rises, Figure 9, due to increased heat. Since the steam flow to the turbine, Figure 10, is limited to its nominal value, the extracted heat to the turbine remains constant. Due to imbalance between the added and extracted heat, the secondary pressure rises further, which increases the imbalance between core heat power and SG power thus yielding to primary side heat-up. The RCS temperature increase causes coolant expansion and pressurizer level increase, Figure 14, that results in pressurizer pressure increase, Figure 8. The pressurizer pressure control actuates the pressurizer spray valves opening, Figure 15, while the pressurizer relief valves did not open during power increase, Figure 16. The RCS pressure is maintained at the constant value slightly above the setpoint value (15.51 MPa). Thus, the RCS temperature increase is the major cause of OTΔT setpoint decrease (the second term in Eq. (1)), Figure 7. The initial fast increase of the nuclear power and core heat power subsides due to negative reactivity feedback and the compensated measured temperature difference ΔT, Figure 7, that is an indicator of the core heat power follows that trend. On the other side, the RCS temperature continues to rise even faster due to increased difference between produced and removed heat. With RCS pressure at constant value, the RCS temperature rise has the decisive influence on decreasing the setpoint for OTΔT trip. It has also the major influence on actuation of OTDT trip since the compensated measured ΔT rises at a slow rate. It is important to note that the reactor protection functions OPΔT and OTΔT depend on RCS temperature measurement that has a delay related to the actual temperature as it is shown in Figure 6. In all three analyzed cases (ATHLET-base case-5 equations, ATHLET-6 equations, RELAP5), the OTDT trip signal in the first loop trips the reactor. The difference in a time of the generation of reactor trip among the analyzed cases is very little (The largest difference (1.1 sec) was obtained for ATHLET-6 equations and RELAP5). The maximum nuclear power in all analyzed cases (maximum value=2109.8 MW (105.8 % of nominal) obtained for RELAP calculation) was well below reactor trip setpoint (109 %). Negative reactivity insertion following the reactor scram shuts down the reactor and the heat produced in the core decreases. On the secondary side, turbine trip is actuated on reactor trip, Figure 10. The steam dump valves open, Figure 11, with RCS average temperature setpoint at no-load value and with 50 % of the maximum capacity for plant trip mode operation. Due to the fact that at a time of reactor trip the temperatures and accumulated heat in the primary circuit were higher than nominal, the cooldown to no-load conditions was extended when compared with the trip at nominal power. In the time period between 100 and 120 sec, the cooldown process was even stopped as it can be observed on SG power, Figure 5 as well as on RCS temperature, Figure 6 and SG pressure, Figure 9. Despite the steam dump operation the secondary pressure rises significantly above the final no-load value. However, the SG pressure did not exceed the setpoint for SG relief and safety valves. Finally, due to prolonged steam dump operation with high steam flow the RCS cooldown continued and RCS temperatures as well as SG pressure decreased with the trend of achieving their target no-load values.

Following the turbine trip and sudden pressure increase, the SG level shrinkage occurs due to rapid condensation, Figure 13. The main feedwater, Figure 12, is isolated in all three calculations on signal: low RCS average temperature & reactor trip, and the auxiliary feedwater is actuated 5 seconds thereafter. Upon turbine trip, the coolant additional heat-up with expansion into the pressurizer and subsequent pressurizer pressure increase result. The pressurizer relief valve no. 1 opens for a very short time, Figure 16. Immediately thereafter, the reduction of coolant temperature and of specific volume causes the outsurge from the pressurizer and pressurizer pressure drop, Figure 14 and Figure 8. Consequently, the pressurizer spray valves close, Figure 15.

In general, a very good agreement for the relevant physical parameters was obtained between the analyzed cases. There is a certain agreement between the analyzed cases for the period before reactor trip. After reactor trip, the first differences appear as summarized below.

- First, the instant pressurizer pressure increase after reactor and turbine trip is different between the ATHLET (both 5 and 6 equations) and RELAP5 calculation. It is obvious that the resulting insurge into the pressurizer and the pressurizer pressure increase were larger in RELAP5 than in ATHLET calculation. Consequently, in the RELAP5 calculation, the pressurizer pressure control system responded with the higher pressurizer relief valves flow. However, the differences are very small and do not have influence on the global trends.
- A part of the observed differences in the steam generator behaviour can be attributed to a) less detailed model of main feedwater system in ATHLET model so far (modelled only with FILL component) when compared with a detailed RELAP5 model (with the model of piping and realistic main feedwater valve) and b) Time of the reactor trip. Later trip results in larger amount of accumulated heat and greater maximum secondary pressure. However, the time of the reactor trip is only slightly different between the analyzed cases and therefore it cannot explain the major differences between ATHLET and RELAP5 SG behaviour.
- The interaction of control systems (in the analyzed transient, the SG level control and steam dump, in particular) with the inherent system behaviour can impede the identification of the actual causes of the observed differences between the two codes (ATHLET vs. RELAP5).

• There are two differences between the two models for ATHLET calculation (5 vs. 6 equations). First, there is a small difference in the time of reactor trip which may be caused by very small differences in the variables affecting the OTΔT setpoint value. The second discrepancy between the two ATHLET models appears at approx. 120 seconds after transient begin when during the outsurge, the small difference in pressurizer level as well as pressurizer pressure was established.

Table 2 Time table of events – RWAP, 2.4 pcm/sec

Event	ATHLET (base case - 5 equations)	ATHLET (6 equations)	RELAP5
Reactor trip setpoint reached	67.5 sec (OTDT trip in loop 1)	67.0 sec (OTDT trip in loop 1)	68.1 sec (OTDT trip in loop 1)
Rods begin to drop	68.5 sec (1 sec delay)	68.0 sec (1 sec delay)	69.1 sec (1 sec delay)
Turbine trip	68.5 sec (on reactor trip)	68.0 sec (on reactor trip)	69.1 sec (on reactor trip)
Main feedwater isolation	96.8 sec (on signal: low Tavg & reactor trip), 7 sec closure time	96.0 sec (on signal: low Tavg & reactor trip), 7 sec closure time	98.1 sec (on signal: low Tavg & reactor trip), 7 sec closure time
Auxiliary feedwater actuation	101.8 (on MFW isolation, 5 sec delay)	101.0 (on MFW isolation, 5 sec delay)	103.1 (on MFW isolation, 5 sec delay)
Maximum nuclear power	2107.5 MW (68.5 sec)	2107.2 MW (68.0 sec)	2109.8 MW (69.1 sec)
Maximum core heat power	2102.3 MW (68.7 sec)	2102.0 MW (68.2 sec)	2104.6 MW (69.2 sec)
Maximum RCS average temperature	581.9 K (69.7 sec)	581.8 K (69.1 sec)	581.9 K (70.2 sec)

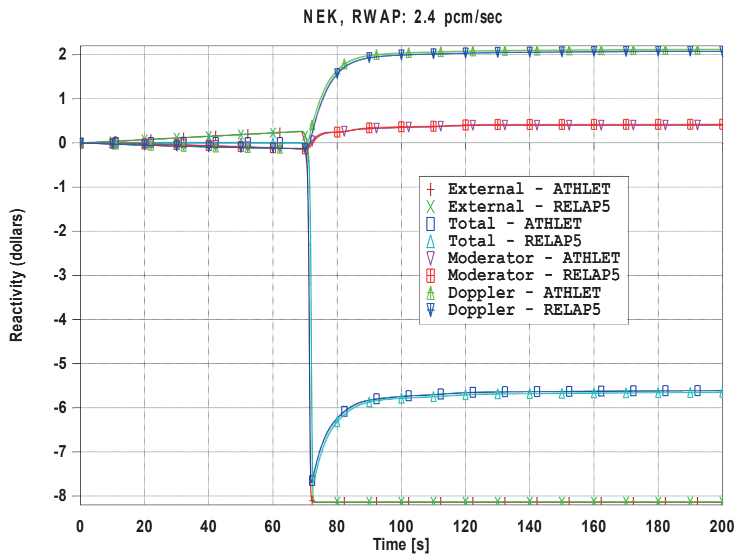


Figure 2: Reactivity (ATHLET – 5 equations)

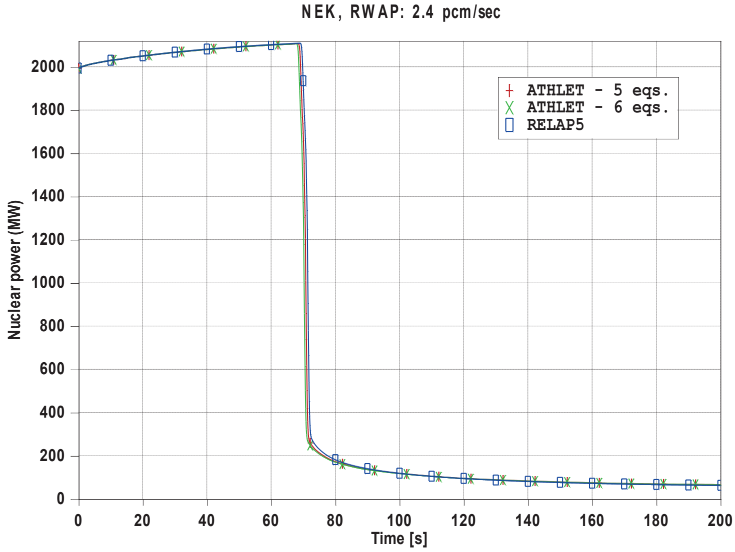


Figure 3: Nuclear power

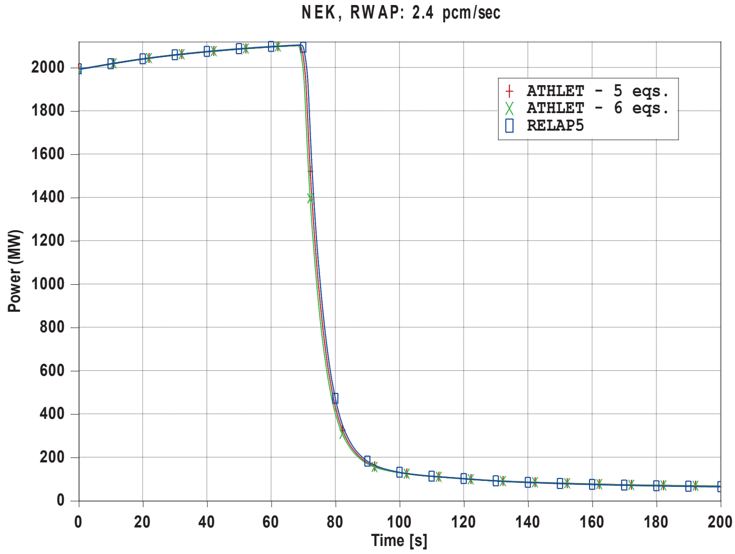


Figure 4: Core heat power

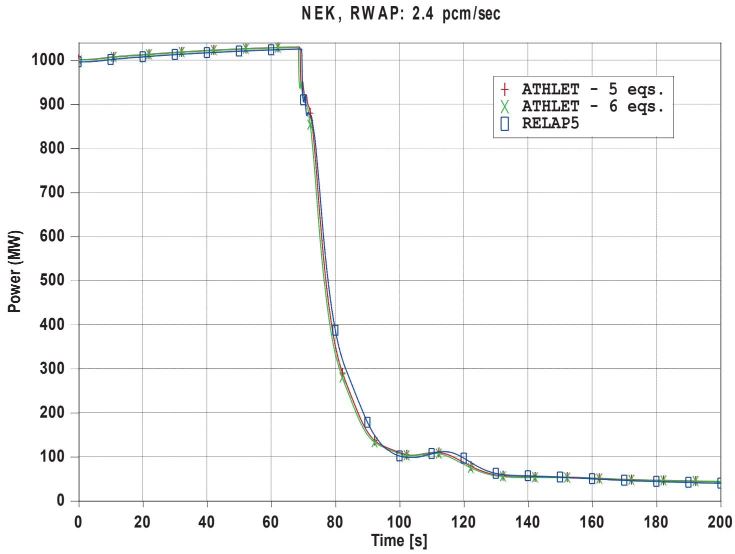


Figure 5: SG 1 power

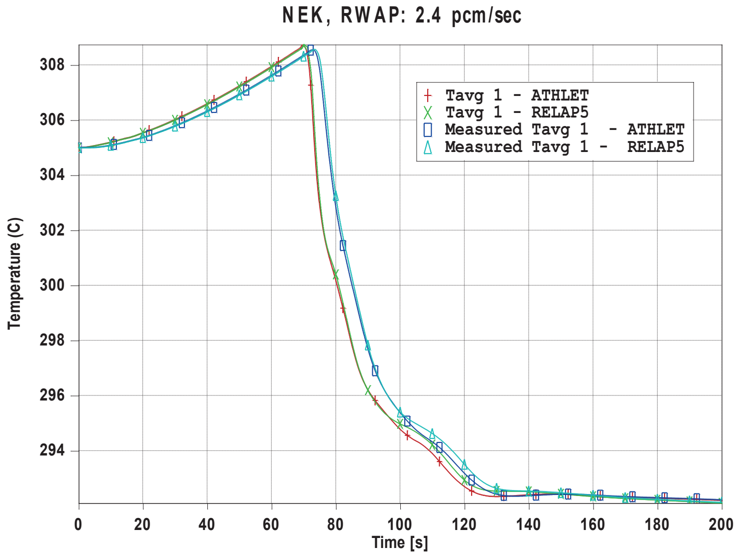


Figure 6: RCS average temperature – loop 1 (ATHLET – 5 equations)

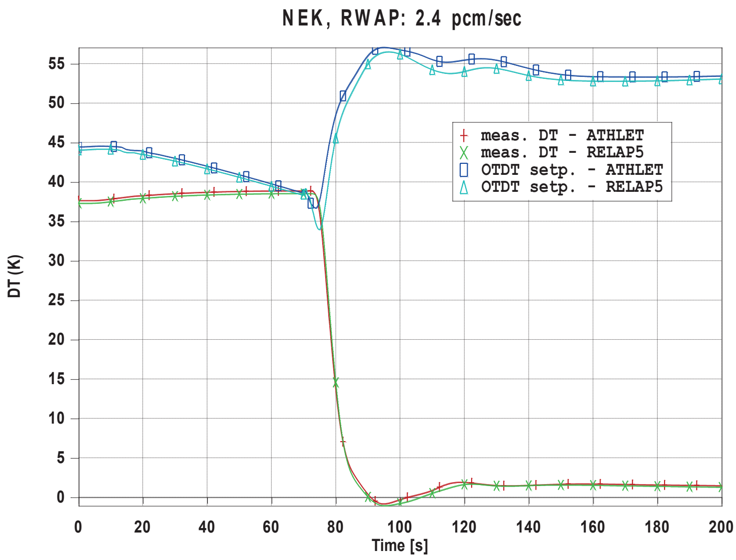


Figure 7: Measured DT and OTΔT setpoint (loop 1, ATHLET – 5 equations)

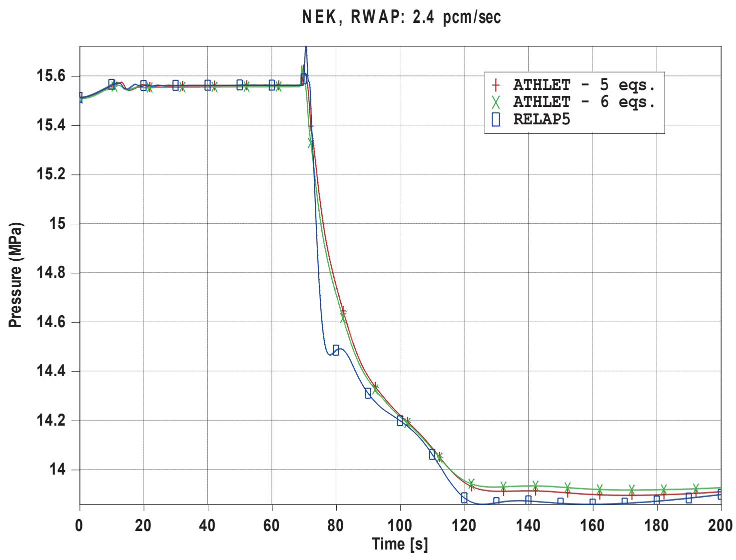


Figure 8: Pressurizer pressure

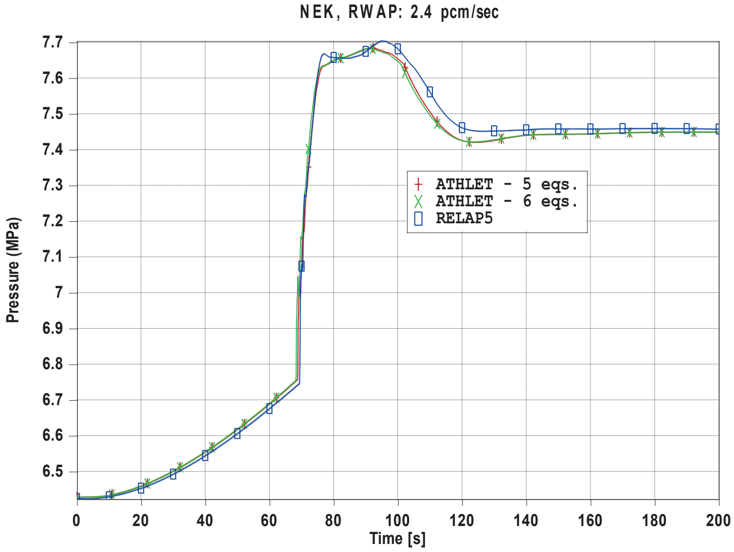


Figure 9: SG 1 steam dome pressure

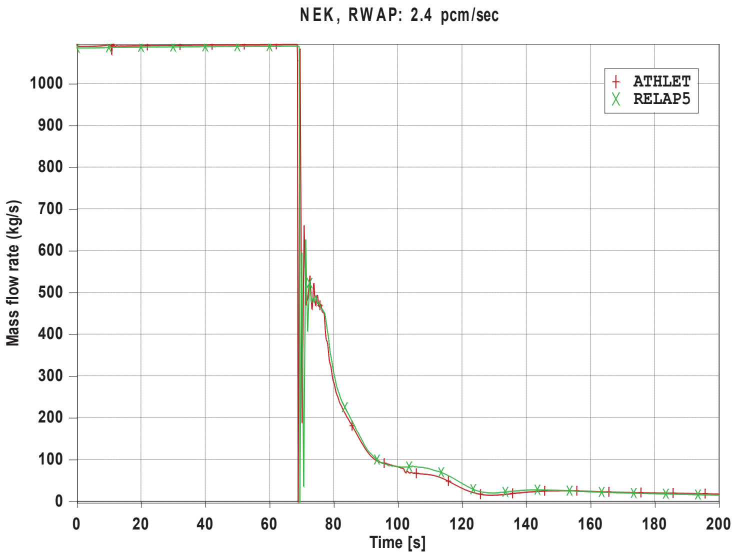


Figure 10: Total steam mass flow rate (ATHLET - 5 equations)

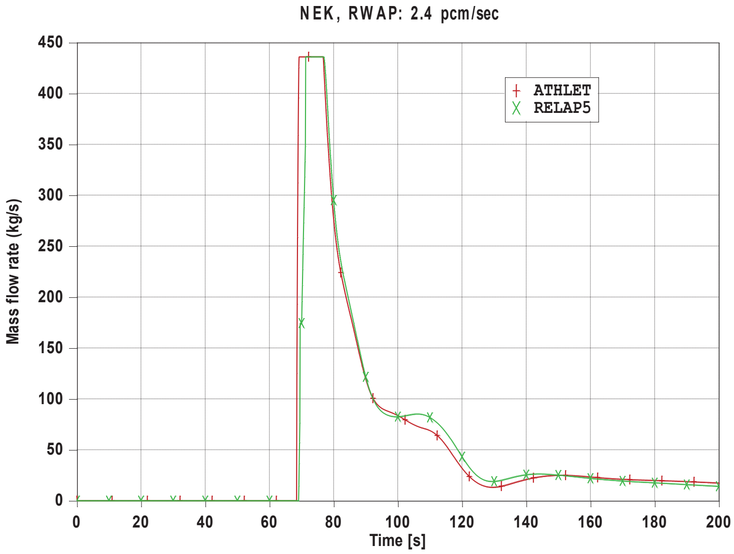


Figure 11: Steam dump mass flow rate (ATHLET - 5 equations)

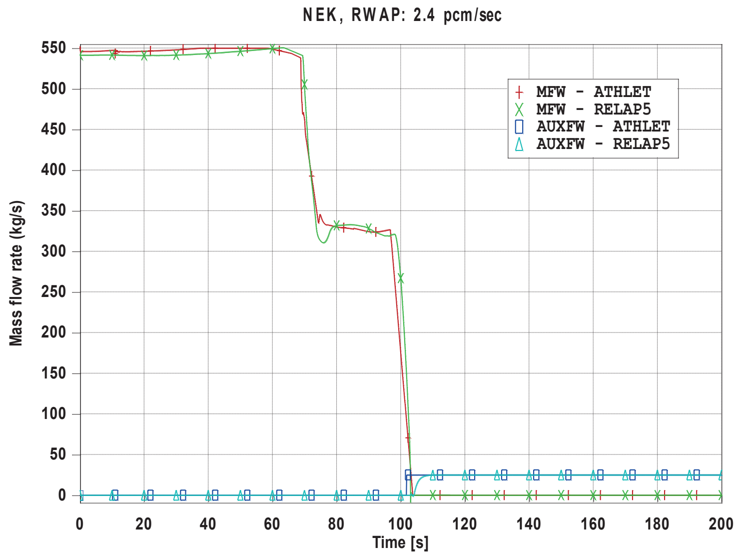


Figure 12: SG 1 main feedwater and auxiliary feedwater mass flow rate (ATHLET – 5 equations)

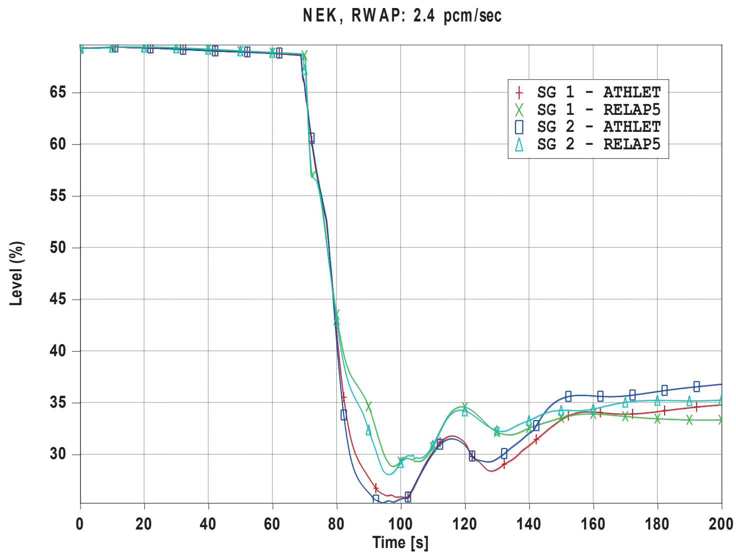


Figure 13: SG level (ATHLET – 5 equations)

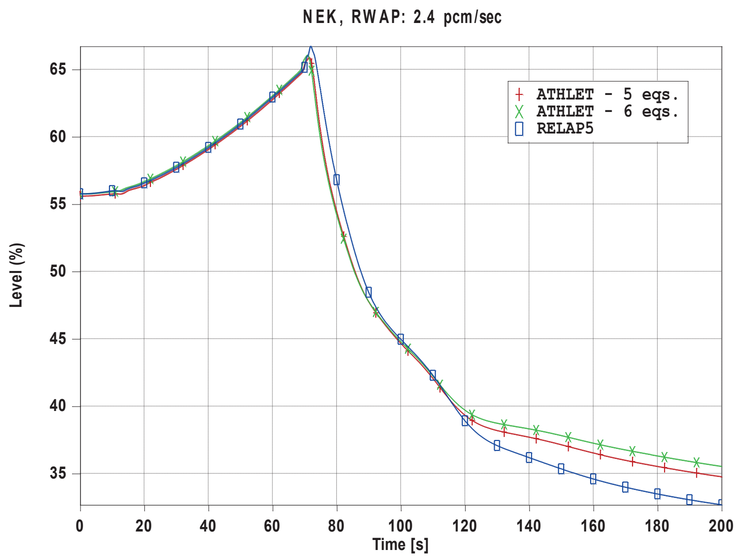


Figure 14: Pressurizer level

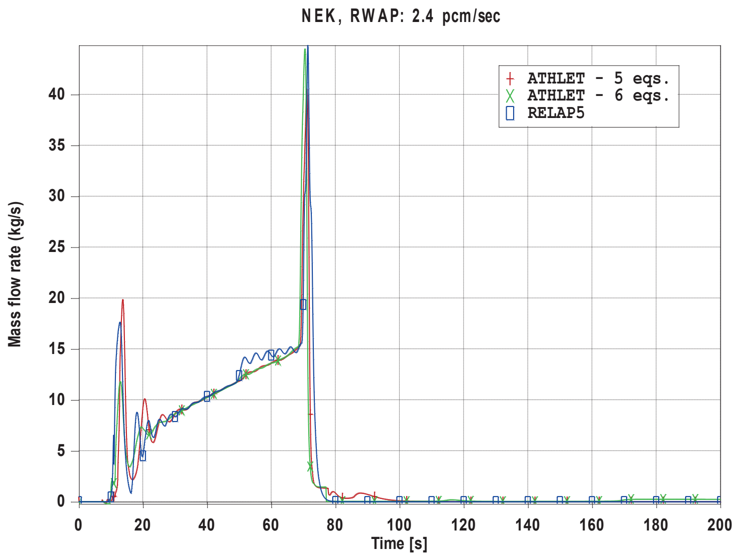


Figure 15: Pressurizer spray mass flow rate

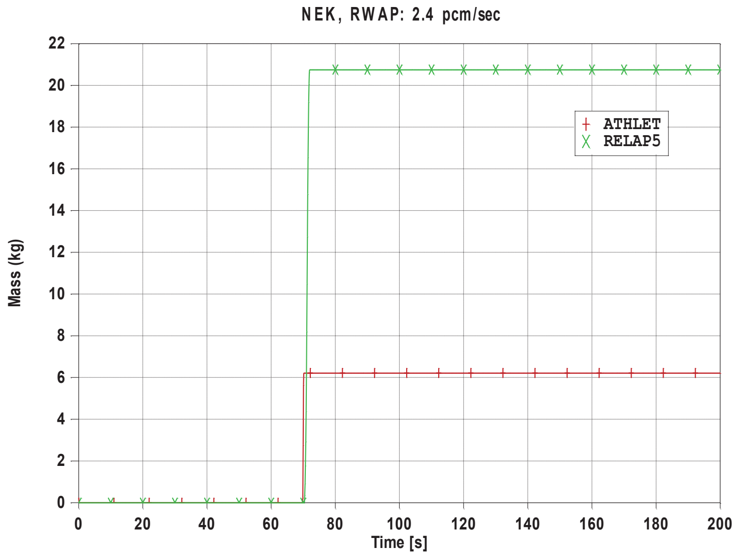


Figure 16: Integral of pressurizer relief and safety valves mass flow rate (ATHLET – 5 equations)

4 CONCLUSION

The ATHLET Mod 2.2 Cycle A model for NPP Krško has been developed. For that purpose, the NEK data base that has been used at FER for development and maintenance of the RELAP5 model for NPP Krško was applied. The ATHLET model was tested by analyzing the RWAP accident initiated after 1000 seconds steady state calculation. The results of both steady state calculation and RWAP transient analysis were assessed against the analysis performed by RELAP5/mod 3.3 code.

For ATHLET analysis, two fluid dynamic options were tested for the primary side: a) base case analysis with 5 conservation equations and mixture level model and b) two-fluid model with separate conservation equations for liquid and vapour phases (6 equations) and the 5 equations+mixture level model for pressurizer. For both cases, 5 equations model was selected for the secondary side. For RELAP5/mod 3.3 analysis, the standard 6 equation model was used. Following conclusions can be drawn from the presented analyses:

- The steady state calculation has been performed for 1000 seconds for both ATHLET and RELAP5. A very good agreement for relevant physical parameters between the two codes was obtained.
- In the transient analysis all three analyses have shown the same trends and the differences for the main physical parameters are very small. First noticeable differences between the analyzed cases appear after reactor and turbine trip. The differences between RELAP5 and ATHLET models (both 5 and 6 equations) are larger than the differences between the two ATHLET models (5 vs. 6 equations).
- The major differences between RELAP5 and ATHLET calculation are related to steam generator behaviour after reactor and turbine trip. One part of the observed differences may be attributed to the differences in the main feedwater model (simplified model in ATHLET compared with the detailed RELAP5 model). In addition, the interaction of control systems (e.g., SG level control and steam dump) with inherent system behaviour can amplify the already existing differences between the two models and impede the identification of the actual causes of the observed discrepancies.
- The observed differences for the time of the reactor trip and slightly different pressurizer behaviour between the two ATHLET models (5 vs. 6 equations) can be attributed to small differences in pressurizer surge line as well as spray line flow.

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