

Influence of Reactivity Feedback Modelling in RELAP5 NEK LONF ATWS Calculation

Davor Grgić, Vesna Benčik, Siniša Šadek, Paulina Dučkić

University of Zagreb, Faculty of Electrical Engineering and Computing

Unska 3, 10000 Zagreb, Croatia

davor.grgic@fer.hr, vesna.bencik@fer.hr, sinisa.sadek@fer.hr, paulina.duckic@fer.hr

ABSTRACT

Standard NPP Krško input deck for RELAP5/mod 3.3 code was used in calculation of LONF ATWS accident with and without AMSAC actuation. The influence of fuel temperature reactivity, moderator temperature reactivity and boron concentration to peak primary pressure and core thermal hydraulic conditions was analyzed for BOL, MOL and EOL core. The main intention is to evaluate acceptability of current simplified boron feedback model in case of low coolant density in the core or partial core uncover.

Keywords: RELAP5, NEK LONF ATWS, reactivity feedback, boron reactivity, core uncover

1 INTRODUCTION

Anticipated Transients Without Scram (ATWS) are operational occurrences followed by the failure of automatic reactor trip signal as well as of control rod insertion. An ATWS is one of the "worst" accidents and the alternative means shall be provided in order to ensure turbine trip and an adequate heat sink on the secondary side. The United States Nuclear Regulatory Commission (USNRC) in its Codes and Federal Regulations Section 10 CFR 50.62, "Requirements for the reduction of risk from Anticipated Transients without Scram (ATWS) events for light water cooled nuclear power plants", requires the installation of actuation systems that are diverse to the reactor trip system, to automatically initiate the following actions: trip the turbine, start the auxiliary feedwater pumps, close the Steam Generator (SG) blowdown isolation valves, close the SG sample line isolation valves, limit the steam dump (SD) to 50 % of the full capacity and activate the turbine trip controller mode for steam dump control. This alternative means is provided in the Krško NPP by the ATWS Mitigation System Circuitry (AMSAC), which provides an alternative means for tripping the turbine and initiating the Auxiliary Feedwater (AFW) flow based on the conditions of loss of secondary heat sink.

Among all ATWS accidents, Loss of Normal Feedwater (LONF) accident and Loss of load accident have the greatest potential for damage to the reactor coolant system boundary, considering pressure peak in the primary coolant system. Loss of normal feedwater could result either from a malfunction in the feedwater condensate system, simultaneous trip of both main feedwater pumps or simultaneous closure of all feedwater control valves. When feedwater flow to the steam generators is terminated, the secondary system can no longer remove all of the heat generated in the core and the primary system heats up. Normally, reactor trip on low steam generator level or on any signal resulting from the primary side heat-up would reduce the imbalance in heat source/heat sink relationship. For LONF ATWS scenario steam generator inventory is rapidly reduced due to the fact that boil off steam is extracted to the turbine and the feedwater is lost. When the steam generator level falls to the point where the steam generator tube bundle is exposed, primary-to-secondary heat transfer is reduced and reactor coolant temperature and pressure begin to increase at

greater rate. A large imbalance in the heat production on the primary side and removed heat results in a significant primary pressure increase due to insurge of expanding reactor coolant into pressurizer. Eventually, pressurizer becomes solid and liquid is discharged through the pressurizer relief and safety valves. AMSAC signal is generated on low SG level and actuates turbine trip and auxiliary feedwater. SG relief and safety valves open after turbine trip but SG inventory depletion is terminated due to auxiliary feedwater flow. Eventually, the heat transfer capability of steam generators is recovered and they are able to efficiently remove heat produced in the core via SG relief valves. Primary temperatures stabilize at a new higher value and the insurge into pressurizer ceases. Consequently, pressurizer level and pressure begin to decrease and loss of inventory through the pressurizer relief and safety valves is terminated.

Due to overheating nature of LONF event, nuclear power will be reduced during the accident due to negative moderator density reactivity feedback despite of the failure of reactor trip system. The influence of negative fuel temperature reactivity feedback is in positive direction but it is rather small when compared with moderator reactivity feedback. The influence of reactivity feedback due to boron may become significant due to the fact that large overheating may lead to significant changes of density of the liquid taking into account that boron solution is transported in liquid phase only.

A design basis analysis criterion for overpressure protection is to verify that Reactor Coolant System (RCS) peak pressure reached during the transient will be less than 18.95 MPa (110% of RCS design pressure 17.2 MPa).

We have analyzed the Loss of Normal Feedwater ATWS for NPP Krško, cycle 29 using RELAP5/MOD 3.3 code. The calculations for Beginning Of Life (BOL), Middle Of Life (MOL) and End Of Life (EOL) were performed. Two accident scenarios were performed; 1. Base case with AMSAC system available and 2. Case based on base case but with auxiliary feedwater not available.

2 CALCULATION MODEL FOR NPP KRŠKO

The standard NPP Krško nodalization developed at FER was used in analyses, ref. [1] and [2]. The nodalization has been updated along with plant modernization and modifications; e.g., SG replacement and power uprate in 2000, RTDBE (Resistance Temperature Detector Bypass Elimination) in 2013. and UFC (Up Flow Conversion) in 2015. The UFC modification was performed in order to minimize the baffle jetting across the baffle-barrel bypass and the core. The modification consisted in altering the reactor vessel internals in such way that the coolant downflow path in the baffle-barrel region was converted to an upflow path. The RELAP5 nodalization consists of 506 thermal-hydraulic volumes, 543 junctions, 383 heat structures with 2127 mesh points, 732 control variables and 197 variable and 221 logical trips. RELAP5 model contains detailed nodalization of NPP Krško Safety Injection (SI) system, Main feedwater and Auxiliary feedwater (AFW) system as well as models of protection and control systems including the detailed models of automatic control rod, pressurizer pressure and level control, steam generator level control and steam dump control with realistic steam dump valves. The scheme of NPP Krško nodalization for RELAP5/MOD 3.3 is shown in Figure 1.

In the calculation with original RELAP5/MOD 3.3 unrealistic results for the reactivity feedback due to change in boron concentration (BORON reactivity) was obtained. In the subsequent chapter the changed model for the calculation of BORON reactivity is presented.

2.1 Boron Concentration Calculation in RELAP5/MOD 3.3

RELAP5 has separate equation to calculate transport of boron within primary loop. In input boron is usually entered as boron concentration C_B (ppm) and internally the calculation is performed in terms of boron density, $RHOB$ (kg/m^3) = $C_B \cdot (\alpha_f \cdot \rho_f)$.

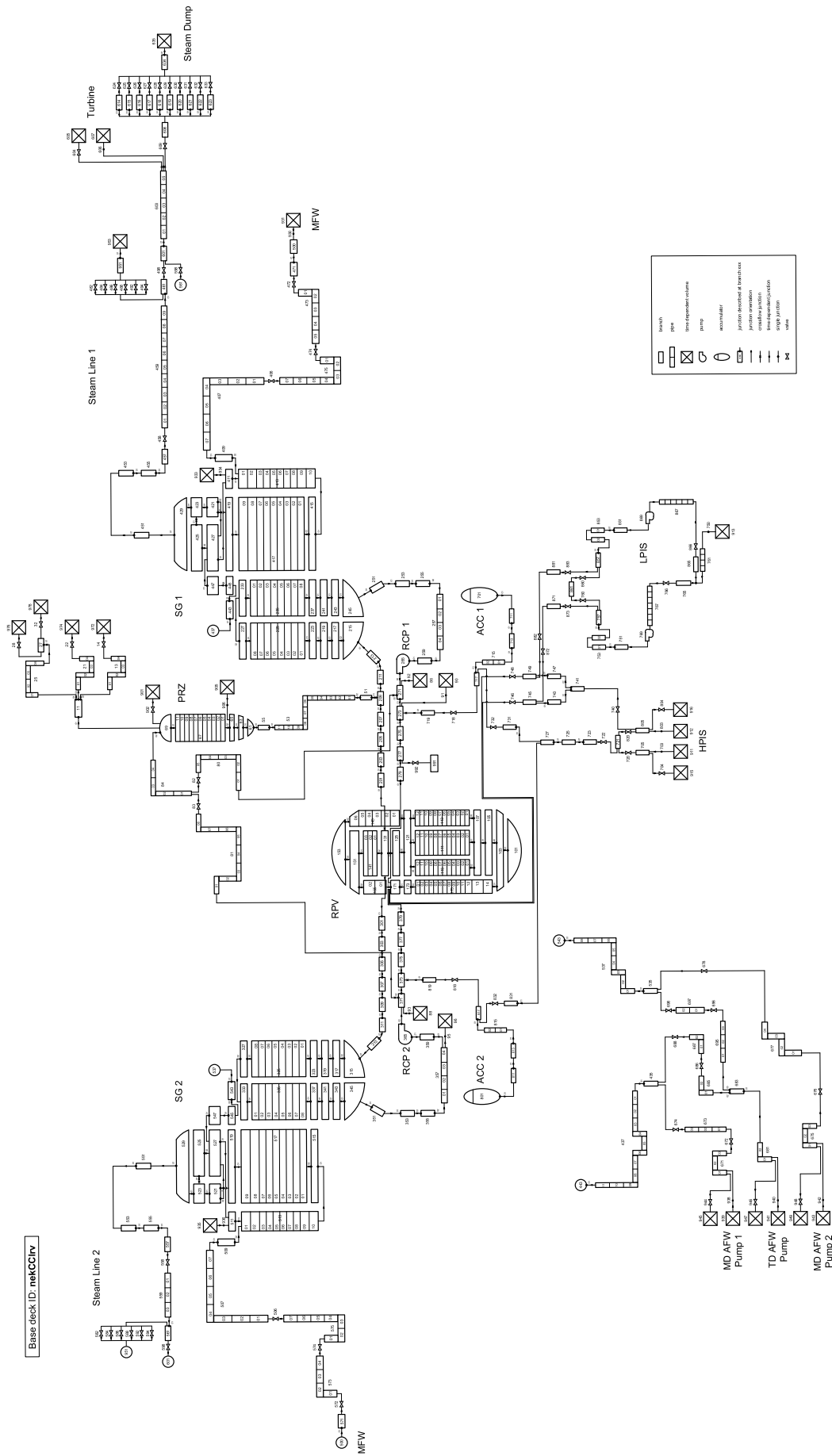


Figure 1: RELAP5/MOD 3.3 nodalization scheme for NPP Krško

RELAP5 has two ways to take into account reactivity feedbacks, the simple one – SEPRABL, using concept of reactivity coefficients and more complicated one – TABLE that uses multi-linear interpolation dependent on selected thermal hydraulic variables. The simple capability was added to RELAP5 code to model boron reactivity feedback using differential boron worth (\$/ppm) calculated using reactor physics codes or taken from nuclear design report of the plant. The change of boron reactivity in point kinetics reactor equations is product of boron reactivity worth and change of current boron concentration compared to initial one calculated during steady state. Boron concentration is calculated for each core active node from corresponding boron density using relationship $C_B = RHOB/(\alpha_f \cdot \rho_f)$. It is obvious that the calculated boron concentration can be large if volume of liquid in the node is small. That situation can happen during core dryout when liquid disappears and boron stays in the node of the core. The analyses that follow give one example when we can get unrealistic response due to this implementation of boron feedback.

3 DISCUSSION OF RESULTS

The LONF ATWS analysis has been performed for Beginning of Life (BOL), 1711 ppm initial boron concentration as well as MOL (boron concentration 1147 ppm) and EOL (boron concentration 51 ppm) using RELAP5/MOD 3.3 code. In the analysis it was assumed that AMSAC signal is actuated on SG NR level <8% with 15 seconds delay. AMSAC signal further actuates turbine trip and initiates auxiliary feedwater. Both motor driven pumps were assumed available having 5 seconds delay. The turbine driven auxiliary feedwater pump was not credited. Safety Injection (SI) signal was assumed available, but safety injection pumps were not accounted for. No operator actions were assumed except maintaining SG level at no load conditions (60-70% narrow range) and reactor Coolant Pump (RCP) trip on Loss of subcooling together with SI signal (60 seconds delay). Chemical and Volume Control System (CVCS) was assumed not available. Pressurizer spray and pressurizer heaters were credited as well as steam dump operating in plant trip mode (50% of total steam dump capacity).

Two transient scenarios cases were analyzed for different core conditions (BOL, MOL and EOL).

1. Base case with turbine trip on AMSAC signal (delay = 0 s), delay of MD AFW pumps = 5 s, TEND=5000 s.
2. Base case with AFW not available. The calculation is stopped 100 seconds after core cladding temperature exceeded 1000 K.

Following the isolation of feedwater, SG inventory is rapidly reduced. Due to absence of reactor trip on one side and aggravation of heat transfer capability of steam generators RCS heat-up results, see Figure 2 and Figure 3. Nuclear power is being reduced due to negative moderator reactivity feedback, Figure 4 and Figure 5. Due to coolant specific volume increase insurge into pressurizer and pressure increase result, see Figure 6. In the analysis it was assumed that pressurizer pressure control is active and therefore the maximum pressure is well bellow the design basis criteria (18.95 MPa). The AMSAC signal is actuated on SG level less than 8% with 15 seconds delay. It actuates turbine trip and auxiliary feedwater. Following turbine trip, steam dump valves open in turbine trip mode. However, due to low-2 steam line pressure main steam line isolation is actuated. Safety injection signal is also generated on low-2 steam line pressure. In the analysis it was assumed that operator stops the RCP pumps following the loss of subcooling together with SI signal. Along with auxiliary feedwater injection the steam generator heat transfer capacity improves and the heat transfer from primary to secondary side is established. Primary temperatures stabilize and the insurge into pressurizer ceases.

In the case with AFW not available the heat sink is completely lost after first 200 seconds of the transients since the SG inventory depletes before turbine trip (69 seconds after transient begin)

and the SG inventory is not recovered. In that case an intense evaporation in the core takes place and finally core dry-out occurs, Figure 7.

As already mentioned before, unrealistic results for reactivity feedback due to change of boron concentration was obtained in original RELAP5/MOD 3.3 code, see Figure 4. The effect is slightly visible in the base case where the BORON reactivity does not decrease below -0.4\$ while in the case with AFW not available BORON reactivity reached up to -200000 \$ (in Figure 4 the results up to -100 \$ are shown). After correction, the realistic BORON reactivity was obtained with slightly positive reactivity at the beginning and finally realistic reactivity values during the rest of the transient were obtained (up to -57 \$). It is important to note that the correction of BORON reactivity calculation leads to more adverse results (maximum pressurizer pressure, time to core dry-out) for all the cases but for MOL and EOL in a much lesser extent than for BOL case. In Table 1 the results for time sequence of events together with most critical parameters for all the cases are summarized. After start of AFW each cell in the table has two entries, upper is for base case and lower is for case without AFW. It is clear that the case without AFW results with more adverse conditions and last two rows apply only for the case without AFW. For any case more conservative values are obtained for BOL core conditions, what is usual for core overheating accidents. Maximum pressurizer pressure and time of core dryout are given for original boron feedback calculation and for corrected one. The corrected model gives little bit higher pressures and earlier core dryout, meaning it is acting in nonconservative direction.

Table 1 Time sequence of the main events (LONF ATWS)

Event	BOL	MOL	EOL
Loss of normal feedwater	0.0 s	0.0 s	0.0 s
AMSAC signal (SG level<8%, 15 s delay)	68.7 s	68.8 s	68.9 s
Turbine trip (on AMSAC signal)	68.7 s	68.8 s	68.9 s
MD AFW pumps started (5 seconds delay)	73.7 s	73.8 s	73.9 s
SI signal (on low-2 steam line pressure)	90.7 s (base case) 91.0 s (w/o AFW)	90.1 s (base case) 90.4 s (w/o AFW)	89.4 s (base case) 89.8 s (w/o AFW)
Steam line isolation (on low-2 steam line pressure)	90.7 s (base case) 91.0 s (w/o AFW)	90.1 s (base case) 90.4 s (w/o AFW)	89.4 s (base case) 89.8 s (w/o AFW)
RCP trip (on loss of subcooling), 60 s delay	150.7 s (base case) 151.0 s (w/o AFW)	166.1 s (base case) 163.6 s (w/o AFW)	228.5 s (base case) 193.3 s (w/o AFW)
Maximum pressurizer pressure	17.5 MPa (base case) 17.5 MPa (w/o AFW)	17.25 MPa (base case) 17.3 MPa (w/o AFW)	16.26 MPa (base case) 17.05 MPa (w/o AFW)
Maximum pressurizer pressure (BORON corrected)	17.5 MPa (base case) 17.6 MPa (w/o AFW)	17.24 MPa (base case) 17.3 MPa (w/o AFW)	16.26 MPa (base case) 17.07 MPa (w/o AFW)
Core dry-out	765 s	1008 s	1221 s
Core dry-out (BORON corrected)	711 s	1000 s	1221 s

4 BORON REACTIVITY MODEL

In Figure 8 liquid density and liquid fraction in 12 active core nodes are given during LONF ATWS base case. In Figure 9a corresponding changes in nodes boron density are given. Even though the boron density is decreasing boron concentration is increasing due to faster decrease of product of liquid fraction and node liquid density (definition of boron concentration). In reality boron is less effective for absorption with increase of void fraction. In order to make temporary correction of the nonconservative boron reactivity behavior, the boron concentration is additionally weighted by multiplication with node liquid fraction, $C_B = \alpha_f RHOB/(\alpha_f \cdot \rho_f)$. That is not formally correct, but it is closer to the way how boron worth is calculated in reactor physics codes.

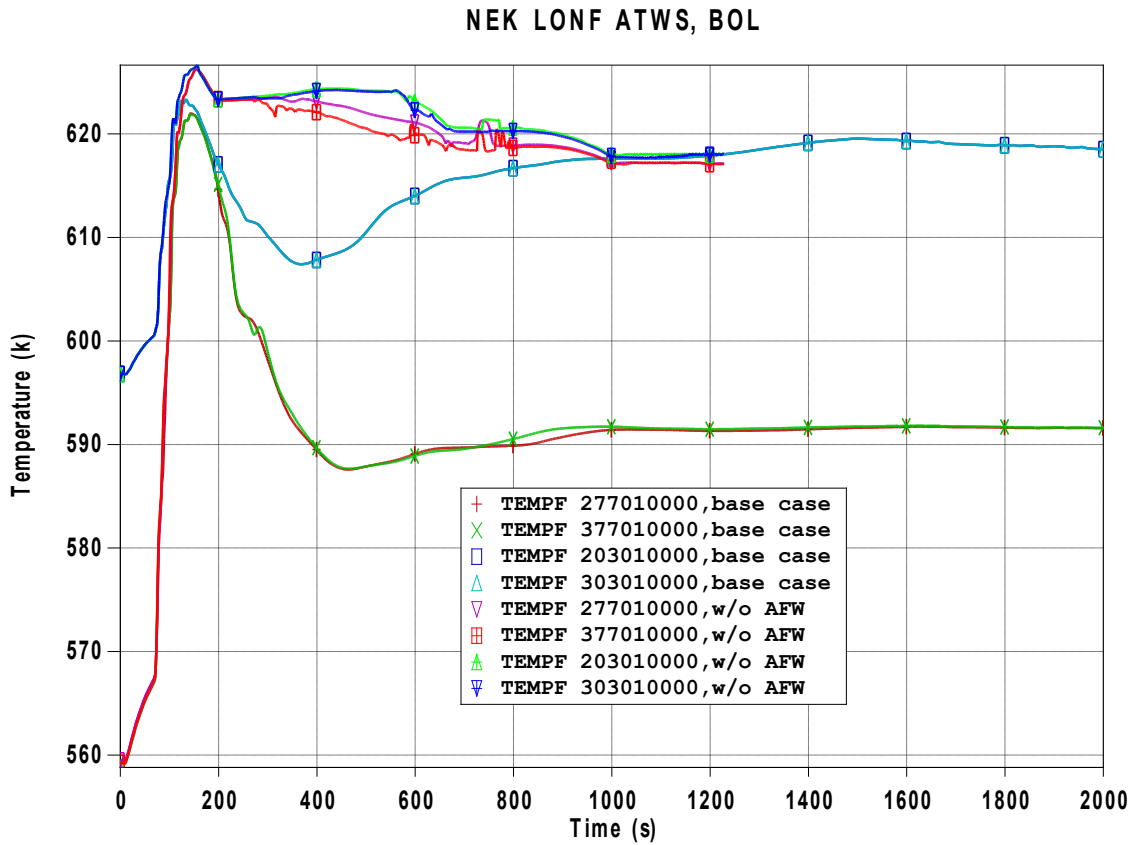


Figure 2: LONF ATWS, BOL: RCS Temperature

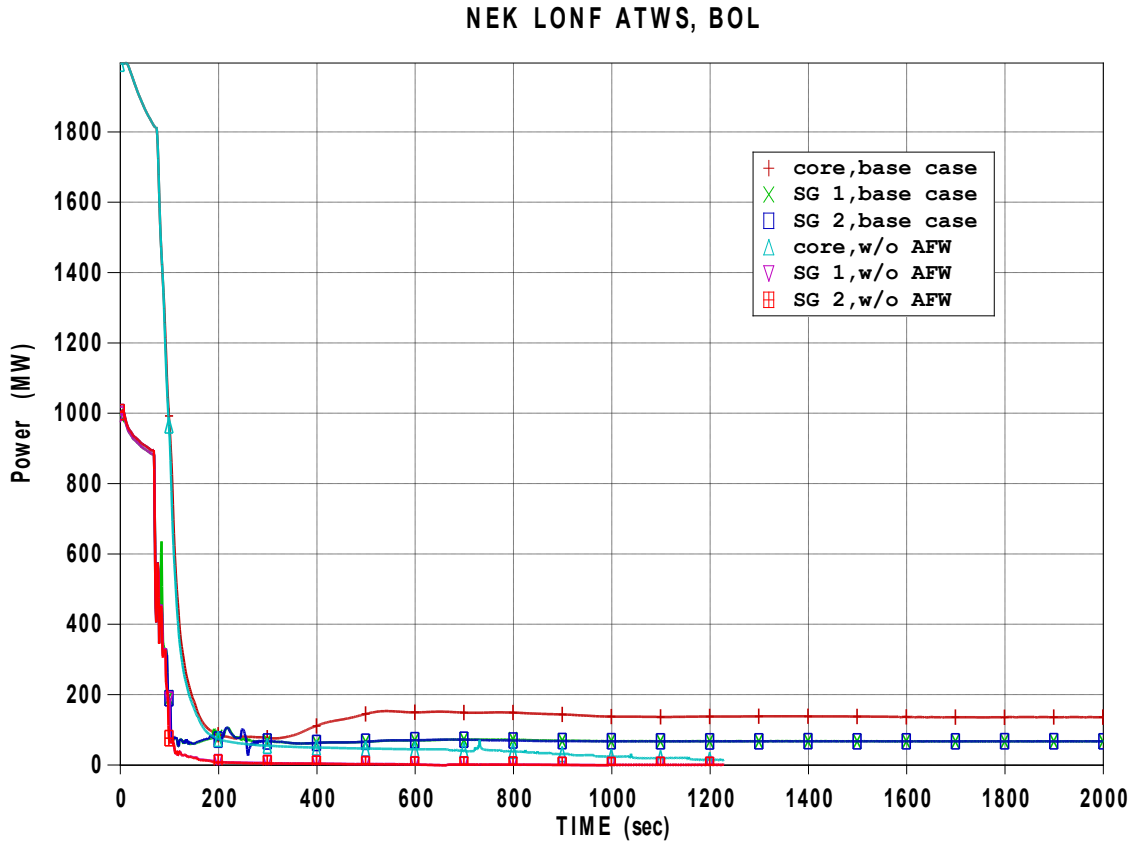


Figure 3: LONF ATWS, BOL: Transferred heat in the core and steam generator

NEK LONF ATWS, BOL

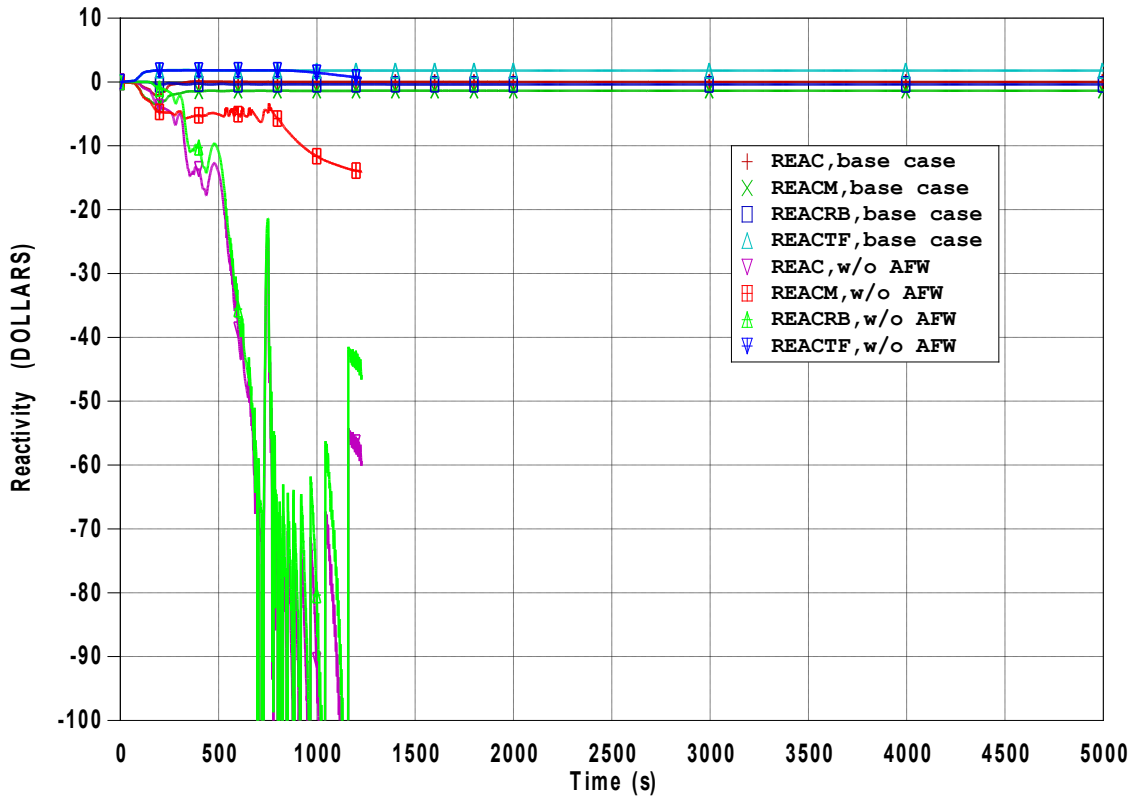


Figure 4: LONF ATWS, BOL: Reactivity

NEK LONF ATWS, BOL, BORON corrected

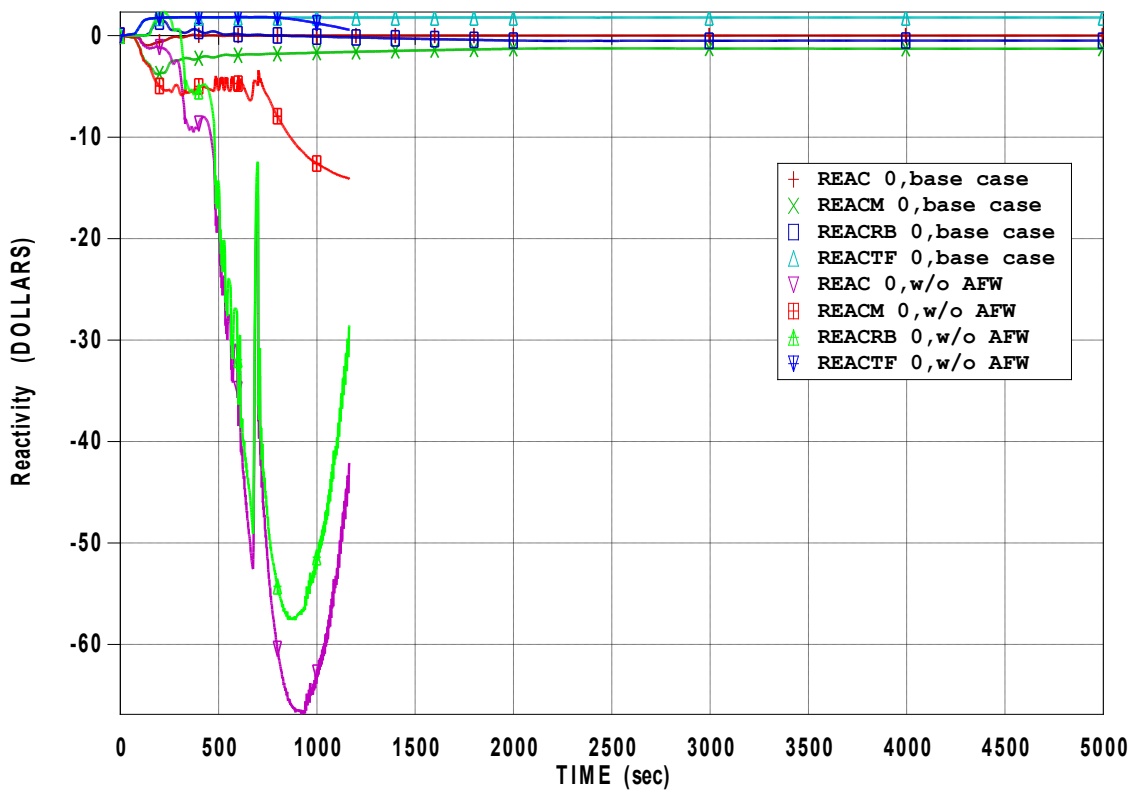


Figure 5: LONF ATWS, BOL, BORON corrected: Reactivity

NEK LONF ATWS

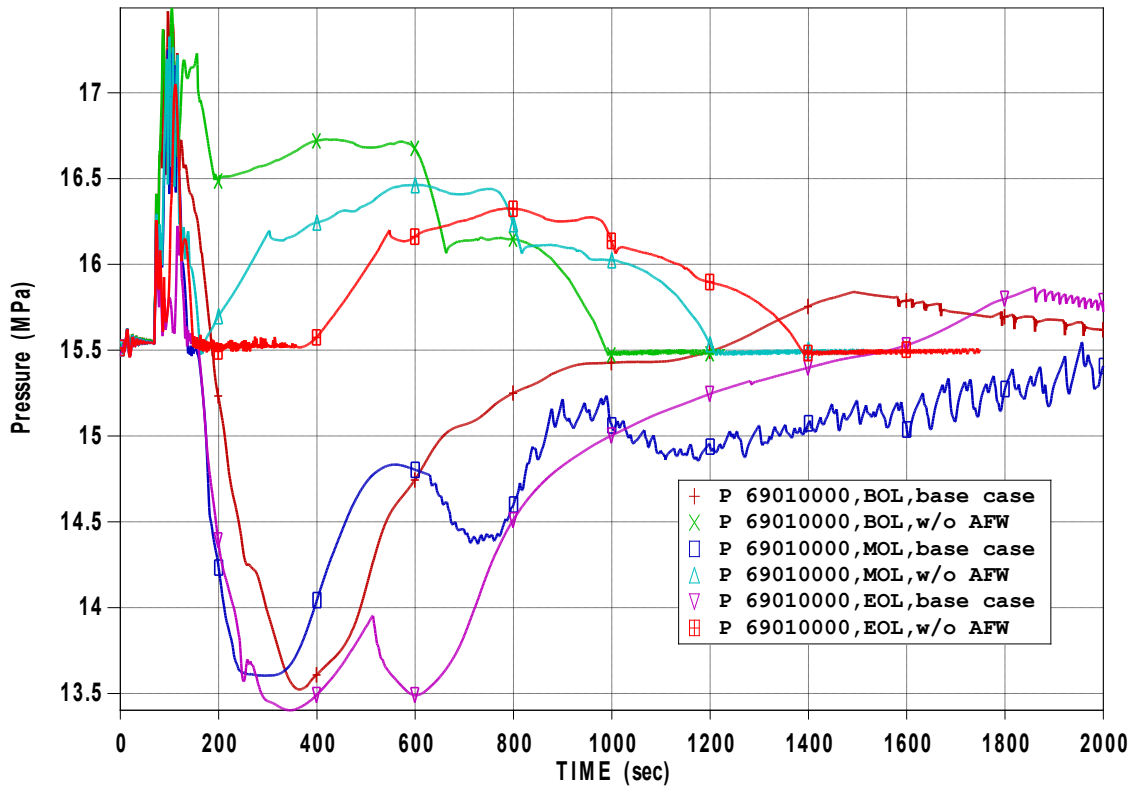


Figure 6: LONF ATWS: Pressurizer pressure

NEK LONF ATWS, w/o AFW

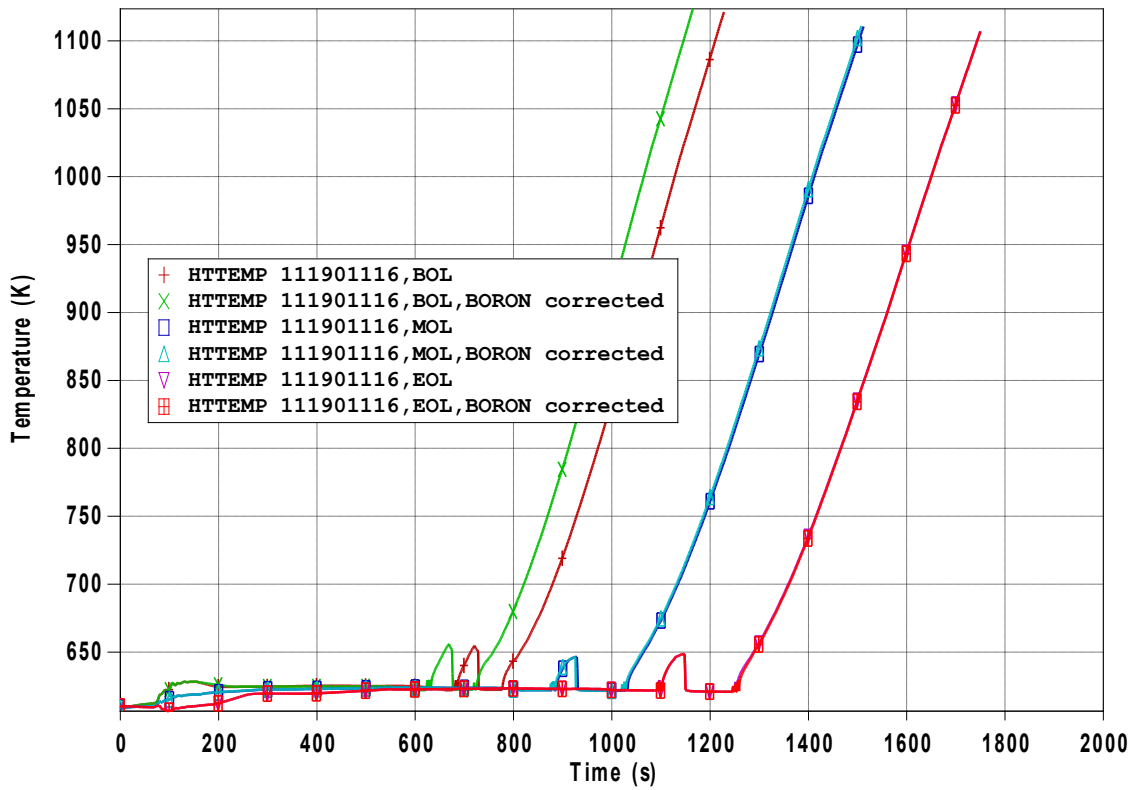
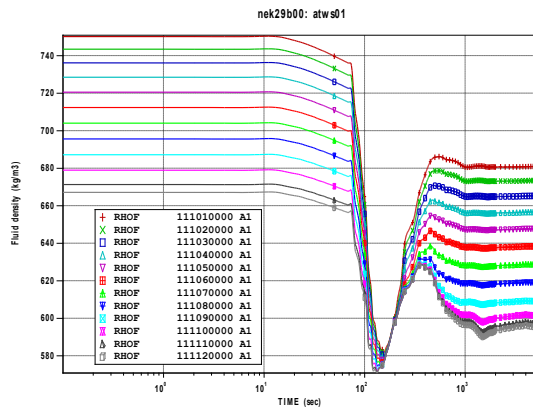
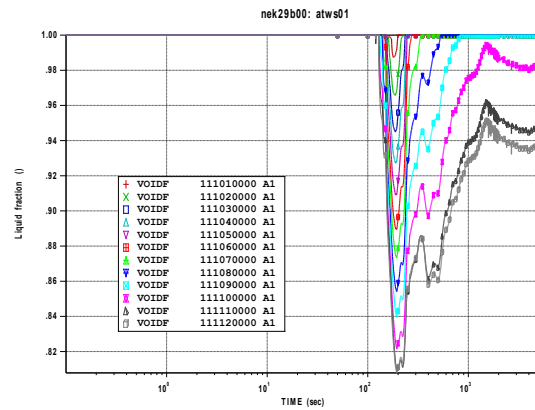


Figure 7: LONF ATWS, w/o AFW: Fuel cladding temperature (N=11)

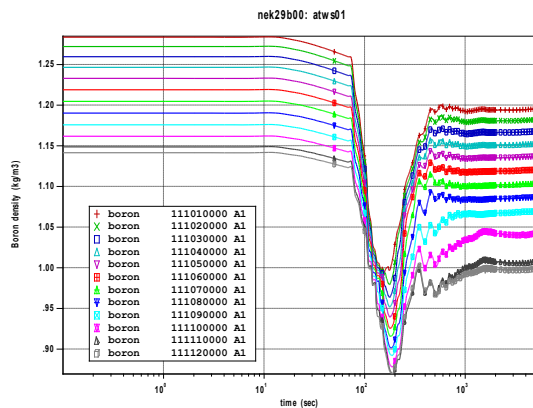


a

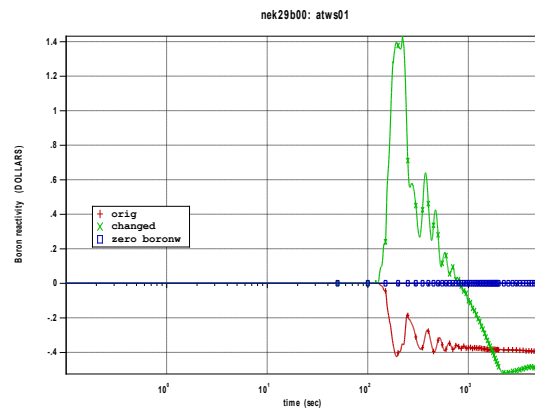


b

Figure 8 LONF ATWS base case, liquid density (a) and liquid fraction in core nodes (b)



a

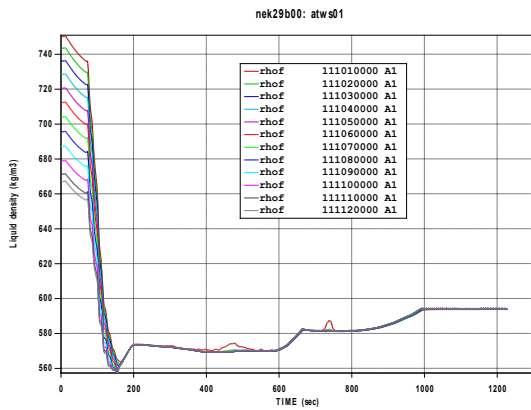


b

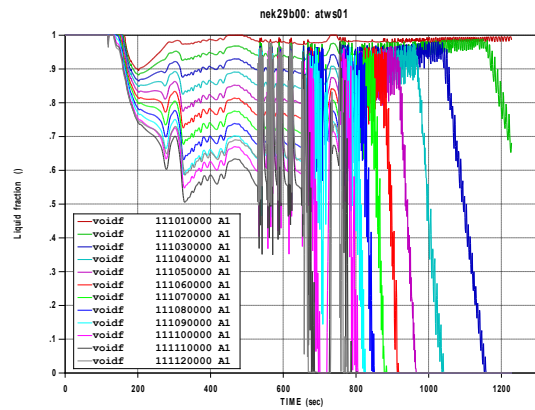
Figure 9 LONF ATWS base case, core nodes boron density (a), boron feedback reactivity (b)

In Figure 9b the change of boron reactivity is shown (label orig), together with change without boron in the fluid and with the corrected case (label changed). Instead of introducing negative reactivity during core heatup, boron reactivity is inserting positive reactivity after correction. It should be mentioned that the influence is partially hidden by inserted negative reactivity due to under moderation of the core.

The situation is even more pronounced during LONF ATWS case without AFW. In Figure 10 the liquid density and liquid fraction in 12 active core nodes are shown. In Figure 11a corresponding changes in nodes boron density are given. Even though the liquid is disappearing the boron density is increasing and boron concentration is increasing even faster due to decrease of product of liquid fraction and node liquid density. In reality boron is almost ineffective and it is in the form of plate out on fuel surface. In Figure 11b very large negative change of boron reactivity is shown (label orig), together with change without boron in the fluid and with the corrected boron reactivity case (label changed). Instead of introducing very large negative reactivity during core dryout, the boron reactivity is negative, but without extreme values. The models with corrected boron feedback are giving little bit higher peak primary pressure and earlier dryouts, for cases without AFW, as already said.

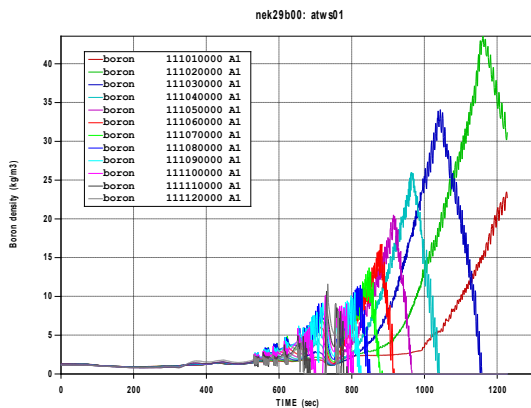


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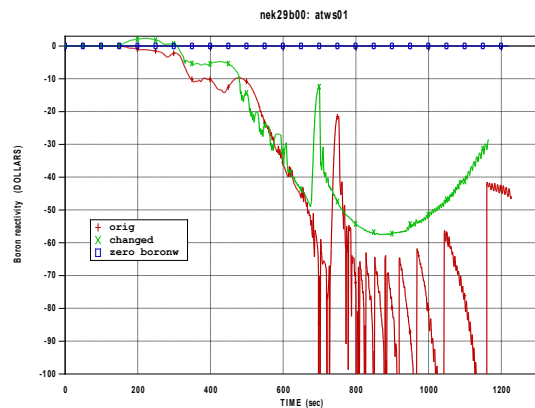


b

Figure 10 LONF ATWS w/o AFW, liquid density (a) and liquid fraction in core nodes (b)



a



b

Figure 11 LONF ATWS w/o AFW, core boron densities (a), boron feedback reactivity (b)

5 CONCLUSION

It is clear that the case without AFW results with more adverse RSC conditions and potential for core dryout. For any case more conservative values are obtained for fresh core, what is usual for core overheating accidents. When dryout starts the boron which is not transported with evaporated liquid stays in the core nodes and that together with very small amount of remaining fluids results in large boron concentrations. That, in simple model using boron reactivity worth introduces very large negative reactivity. In reality boron depositions within core are not so effective in absorption as boron solution in the fluid. The effect is more important in ATWS LONF case without AFW due to less coolant left in the core. In order to temporary correct that situation boron concentration is additionally weighted by multiplication with node liquid fraction. With that correction for situation without dryout the reactivity model is almost the same as the original one and during dryout situations increase in negative reactivity due to boron feedback is limited. The corrected model gives little bit higher pressures and earlier core dryout, meaning that the original model is not conservative enough.

REFERENCES

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- [2] NEK RELAP5/MOD3.3 Steady State Qualification Report for Cycle 29, NEK ESD TR 13/16, Revision 0, Krško 2017.