Selected Examples for Safety Analysis in VVER-440 Type Reactors Simulated by the Coupled ATHLET/KIKO3D Code System

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ABSTRACT

Recently several projects have been initiated in Hungary aiming at the introduction of new fuel type, increased maximum allowed power and economic fuel cycle. The planned upgraded power and parallel application of new fuel type require the renewal of the relevant chapter of the Final Safety Analysis Report (FSAR). One of the main tools used for analyzing transient scenarios initiating by reactivity and power distribution anomalies was the ATHLET/KIKO3D coupled neutron kinetic / thermal-hydraulic code.

This paper gives an overview of two analyses, which was prepared in the frame of the revision of Paks FSAR, namely the “withdrawal of one control rod” and “initial phase of main steam line break” events.

1 INTRODUCTION

The safety assessment of a nuclear power plant includes the analysis of its response to abnormal events to determine if the plant is capable of adapting such postulated transients. The scenarios of different type of accidents must be analysed in the frame of the licensing process as part of the safety analysis report prescribed by the authority. Recently such type of projects have been initiated in Hungary aiming at the introduction of new fuel types, increased maximum allowed power and more economic fuel cycle. The planned upgraded power and parallel application of new fuel type require the renewal of the relevant chapter of the safety analysis report. Due to the modified reloading schemes and reactor physics characteristics, the reactivity initiated accidents (RIA) and the anticipated transients without scram (ATWS) requiring coupled 3D neutronics and thermal hydraulics analyses are of special importance.

The most characteristic behaviour of the investigated scenarios in the reactor is the potential development of the asymmetric power distribution of the core. Overcooling in one loop, boron slug entering in the core, control rod ejection and/or the malfunction of the scram control rods are the most common sources of the reactivity insertion. The hazard of the event depends on the initial steady state conditions and the supposed single malfunction. Special attention has to be devoted to the three ionization chambers (IC) encircling the core.

The simulation of such complex transients with tight link of the whole-plant thermal hydraulics and the core neutronics are very challenging for best-estimate safety analysis tools, even though their simulation capability had been considerably extended and improved in the last decade. The main tool in the analysis is the coupled ATHLET-KIKO3D-TRABCO code.
system used for such RIA and ATWS analyses at KFKI-AEKI for a long time. Its qualification and validation had been performed by means of experimental data gained in nuclear power plants or code-to-code comparisons in the frame of numerous international benchmarks.

This paper shortly describes the software used for analyzing transients, then two, not frequently investigated cases are presented. First, a mild overcooling transient is simulated then the scenario of an inadvertent control rod withdrawal is shown.

2 CODE SYSTEM USED IN THE SIMULATION OF TRANSIENTS

In order to predict the spatial power distribution in a VVER-440 type nuclear power plant, model with detailed space-time resolution in the core neutronics (KIKO3D) is directly coupled with the plant thermal hydraulic code (ATHLET 1.2 cycle-A). The code complex comprises hot channel calculations (TRABCODA analysis code) and a model for simulation of the time dependent signal of the outcore detectors. The last two models could not run simultaneously with the coupled system code. The data transfer of the coupled calculation is illustrated in Figure 1, while the codes are detailed below.

![Figure 1: The coupled 3D kinetics, thermal hydraulic and hot channel calculation](image)

2.1 KIKO3D Neutron Kinetic Model

KIKO3D is a three-dimensional reactor dynamics program for coupled neutron kinetics and thermo-hydraulics calculation of VVER type pressurized water reactor cores [1-2]. The code was developed in the MTA KFKI Atomic Energy Research Institute. It is a nodal code where the nodes are the hexagonal or rectangular fuel assemblies subdivided into axial layers. Typical numbers of assemblies and axial layers for a VVER-440 core are 349 and 10, respectively. Including the new Russian profiled fuel, parameterized response matrices for 20 different VVER fuel can be found in its nuclear data library. The neutron kinetics model of KIKO3D solves the two-group diffusion equations for homogenized fuel assemblies. Special, generalized response matrices of the time dependent problem are introduced.

Its thermo-hydraulic model calculates in separate axial hydraulic channels of the core, each of which relates to one fuel assembly. The conservation equations of mass, energy and
momentum are solved for liquid and vapour phases. In order to get an accurate representation of the fuel temperature feedback, a heat transfer calculation with several radial meshes is done for an average representative fuel rod in each node. The release of prompt and delayed nuclear heat in the fuel is modelled. In the present version of the code, the VVER-440 correlations are used in the thermo-hydraulical module.

The signals of IC detectors are evaluated from the space and time dependent flux and current in the core, which are calculated by KIKO3D, and from predetermined transfer functions between the outgoing current at the core periphery and the detector reaction rate. This latter function was determined by a set of MCNP Monte-Carlo calculations.

The timing of the scram signal is influenced to a great extent by the asymmetric character of the transient. Namely, the voting logics of the ERPS (Emergency Reactor Protection System) results chambers. The power level of the core is measured by detector sets consisting of three detectors positioned in rotational symmetry of 120 degree around the core. Two sets of IC’s can be found around the core, their relative position is 15° rotation. The positions of the loops and those for the reactor protection detectors used for the scram actuation are shown in Fig. 2.

The limit value is corresponding to an azimuthally symmetric neutron flux-distribution and the voting logics is; therefore, if one ionization chamber is located just in front of the power peak, the two other ones are reaching the limit values only at a significantly higher reactor power. The scram signal is initiated either if the reactor-period is less than 10 s, or if the signal level reaches the prescribed value. It turned out, that the initiation of scram is really delayed due to the space dependent detector readings.

Several activities were performed in international co-operation to validate the KIKO3D code [3].

2.2 Thermal-Hydraulics Code

The ATHLET code is a thermo-fluid-dynamic system code for a wide range of applications comprising anticipated and abnormal transients, small and intermediate leaks as well as large breaks in LWR. The code structure allows an easy implementation of different physical models, such as thermo-fluid-dynamics, neutron kinetics, General Control Simulation Module. A two-fluid, 6-equation model, with completely separated equations for mass, energy and momentum for both phases, taking into account also the non-condensables is included in the last release versions. In our calculations, the 1.2A version is used. In addition to the thermal hydraulics, it includes models for describing components like coolant pumps, pressurizer and steam generator, as well as for modelling control systems and protection system [4]. It helps us to model all of those systems, which in our case differ from other PWRs. For the Russian type pressurized water reactors, a detailed six loop input model was developed for the ATHLET code in our institute. Including the upper head, downcomer and lower plenum, the pressure vessel is divided into six parallel, separate segments. The six sectors are expected to reflect the asymmetrical behaviour of the loops during the modelled transient and to take into account safety systems (ECC systems, control and protection systems) distributed loop by loop. The details of the plant description were also increased and such a way the primary and secondary circuits are modelled by 12 fluid-dynamic systems.
Coolant mixing in the downcomer is modelled. Junctions are introduced among the nodes of the downcomer and lower plenum to fit the cross flow, which are prescribed from the results of the former CFX calculations. The bundle of the horizontal steam generator is divided into three parts in the primary side and five in secondary side. Three of the nodes contain the U tubes. The coupling approach for spatial neutronics models implemented in ATHLET allows various options, such as:

- Internal coupling, when ATHLET models completely the thermal-fluid–dynamics in the primary circuit including the core region. In that case the neutronics model simulates only the 3D power distribution,
- External coupling, when the 3D neutronic model including the fuel rod model and the fluid-dynamic model of the core region coupled to the ATHLET system code which model only the primary circuit excluding the core region,
- Parallel coupling, when the ATHLET simulates the full plant and gives boundary conditions for the core and the neutron kinetics simulates the full time dependent behaviour of the core based on the boundary conditions developed by the system code.

Two ways of the coupling have been implemented for KIKO3D: internal and parallel coupling. In the second case, the two programs run parallel and the KIKO3D code uses their own thermo-hydraulic model in the core. The coupled system code was successfully validated against benchmarks and direct industrial applications [5-6].

In the calculations, presented in the next chapters the 3D core behaviour is modelled by the KIKO3D code using the internal coupling option. Due to the very asymmetric perturbation, all the 349 assemblies of the core were calculated. The assemblies were grouped into 27 ATHLET super channels as it is shown in Figure 2.

2.3 Hot Channel Simulation

Coupling of thermo-hydraulic/neutronic codes should be extended to fuel behaviour codes, mainly for fast transients.. Our code system was extended by the TRABCO code, which is a part of the transient code SMATRA, developed by VTT, Finland. It comprises an axially 1D two-group reactor dynamics code, fuel rod heat transfer, and thermal hydraulics of typical coolant channel. Fuel temperature rise after boiling crisis and clad oxidation are modelled as extreme phenomena. TRABCO core model can be used for hot channel analyses based on the output files made by the KIKO3D code. Detailed validation procedure was done for that code package, too. The validation work for SMABRE and SMATRA against Loviisa measurements is described in ref. [7].

Usually the hot channel analysis is performed by the TRABCO code supposing conservative power distribution in the chosen assembly based on KIKO3D calculation. The time-dependent KIKO3D axial power distribution of the most loaded assembly is multiplied by the $K_x$ radial power peaking factor. This factor is determined from the core design limits of the maximum linear heat rate and the maximum pin power. Assuming that these limits are valid at conservatively modified value of the nominal power, the maximum value of $K_x$ radial power peaking factor was limited to 1.59 due to the 57 kW pin power limit. As the initial value of $K_x$ and $K_e$ are known from the KIKO3D calculation, $K_x$ is responsible for $K_e$ and the engineering safety factor. While the time dependent power distribution is taken from the KIKO3D calculation, the inlet enthalpy flow, etc. come from the ATHLET results.

2.4 Overview of the Activity Concerning the New Fuel Type

The upgraded power and the use of a new fuel type usually lead to modified reloading schemes and reactor physics properties of the core. Consequently, in the first step of the analyses the reactor physics core design calculations had to be performed. The reactivity
coefficients etc. and their enveloping values, the so-called “frame parameters” were obtained from these calculations. Classical RIA and ATWS analyses for main steam line break, control rod ejection and boron dilution scenarios were performed by the coupled ATHLET-KIKO3D code. In the most serious cases, transient fuel behaviour calculations were performed by TRABCO code for evaluation of the fuel design criteria, too [8-9].

In all cases, the improved safety enhancement measures for VVER-440 elaborated after the “Advanced General and New Evaluation” of Safety project in 1997 are considered.

3 STEAM LINE BREAK TRANSIENT

The double-ended main steam line break inside the containment at a VVER-440 type reactor was investigated. Not only the fuel, but also several new safety measures make the scenario important, such as:

• the first level protection system (single failure must not be accepted) is extended by signals, such as: overpressure (\(\Delta p \geq 0.11\) MPa) in the containment initiates a reactor shut down and 0.5 MPa pressure difference between steam header and generator cause turbine trip,

• a new function is built into the protection and control system, namely the protection against cold overpressure. It is not allowed that the pressure in the primary cycle to be higher than that one comes from the border curve of the brittle fracture value calculated for 40 years at the coldest temperature available in the core.

The initiating event is an asymmetric break of the main steam line at the end of the equilibrium fuel cycle with profiled Russian fuel working under upgraded power condition (1485 MW). The steam blowing out from the break causes overpressure in the containment, which initiates a reactor shut down with one control rod stuck in its upper position, conservatively. In the scrammed reactor, the main circulation pumps (MCP) are switched off too and later on natural circulation is developed. The pressure difference increases between the main steam header and the damaged steam generator that is why its main steam isolation valve closes at 4.8 s, when the pressure difference is 0.5 MPa. The normal feedwater is not stopped (single failure) in the damaged SG. The continuous fast pressure decrease in the secondary circuit results that the valves of the other turbine are closed, too. See Table 1.

Due to the closing valves cold (159 °C) feedwater enters into the SGs and the steam produced in the intact SGs increase the pressure in the secondary circuit up to 4.9 MPa, which is the opening value for the turbine bypass valves (BRU-K: blowdown to the condensers). It is a conservative assumption for DBA analyses as during normal conditions the steam header relief valves (BRU-A) work and they open to the atmospheric steam dump and cause higher pressure in the SG than the BRU-Ks. The lower pressure results lower coolant temperature in the primary side (Fig. 3 and 4.). At \(T = 502.3\) s the level in the Pressurizer becomes lower than 2.4 m and this together with the low primary pressure initiate the high pressure safety injections (HPIS) within 1 s, which means that cold highly borated water enter to the core and it shuts down the reactor.

In the primary side of the damaged SG, the temperature of the coolant continuously decreases, due to the open valve of the feedwater and the break. The larger pressure loss causes higher heat removal and larger flow in the primary side of the damaged loop. The mixing among the damaged and its neighbouring sectors is higher, than in a symmetric case during normal operation.
Fig. 3: Pressure in the primary circuit and the Pressurizer

Fig. 4: Coolant temperature at the inlet of the core segments

Fig. 5: Reactivity

Fig. 6: Power of the super channels of the core segments belonging to the damaged loop (zoom)

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Table 1 Chronology of the events during the MSLB simulation

<table>
<thead>
<tr>
<th>Time (s)</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>Break open in the 2. steam line loop</td>
</tr>
<tr>
<td>0.55</td>
<td>overpressure in the containment ≥ 0.11 MPa initiates a reactor shut down (ERP-1 and ECCS signal), MCPs stop</td>
</tr>
<tr>
<td>0.65</td>
<td>P &lt; 4.1 MPa, turbine connecting to the damaged loop trips, its valve closes</td>
</tr>
<tr>
<td>4.8</td>
<td>Δp (MSH-SG) &gt; 0.5 MPa, MSIV containing the 2. steam loop closes</td>
</tr>
<tr>
<td>7.4</td>
<td>Pressure in the primary system &lt; 11.8 MPa (could have been ECCS signal)</td>
</tr>
<tr>
<td>8.2</td>
<td>The other turbine trips too as the pressure &lt; 3.9 MPa</td>
</tr>
<tr>
<td>16.0</td>
<td>The steam generator blowdown control opens</td>
</tr>
<tr>
<td>20.8</td>
<td>Pressure in the primary circuit &lt;11.3 MPa (could have been ECCS signal)</td>
</tr>
<tr>
<td>502.3</td>
<td>Level in the pressurizer &lt; 2.4 m</td>
</tr>
<tr>
<td>503.3</td>
<td>High Pressure Safety Injection starts to work, inlet temperature of the core sector connecting to the damaged SG is 208.3 °C</td>
</tr>
<tr>
<td>504.5</td>
<td>First opening of the Pressurizer relief valve (YP-22)</td>
</tr>
<tr>
<td>900.0</td>
<td>First action of the operator, switching off the HPIS</td>
</tr>
<tr>
<td>1101.4</td>
<td>Pressurizer relief valve closes for this transient (YP-23)</td>
</tr>
</tbody>
</table>

Even the inlet-temperature of the coolant in the cold leg of the damaged SG is only 122.6 °C when it is entering the correspondent sector of the core its value 208.3 °C. This temperature (Fig. 4) is lower than the recriticality value (210 °C) and the dynamic reactivity asymptotically converges to zero (Fig. 5), however no return to power can be observed, only a very small peak in power (Fig. 6) appears. It is a consequence of the inhomogeneous feedback distributions.

The new protection system in connection of the brittle fracture does not influence the first part of the transient. As the inlet temperature of the core coming colder pressurizer relief valves open from time to time and the pressure in the primary system decreases. It causes boiling in the upper plenum and some disturbance occurs in the natural circulation at about 750s (Fig.3). It is supposed that no special operator intervention happens in the first 15 minutes that is why the HPIS is switched off only at 900 s. The plant is safely stabilized after 1100 s.

4 INADVERTENT CONTROL ROD WITHDRAWAL

Another typical initiating event belonging to RIAs is the uncontrolled withdrawal of a control assembly at full power (1485 MW) with 2 cm/s velocity value from the most unfavourable position permitted by the operational rules (125 cm). During normal operation the regulating control assembly group consisting of 7 assemblies the only one, which is movable by the Unit Power Controller (UPC) in a VVER-440 type NPP. Some electrical failure could cause, that nor the Emergency Reactor Protection (ERP) neither UPC can influence the motion of that group, but one of them inadvertently starts to move upwards gradually. As the transient, slow enough the asymmetric power increase causes asymmetric coolant distribution. The influence of modelling the IC signals for the scram was investigated and the hot channel analysis in the effected sector was carried out.
The best estimate nuclear cross section data were used however the reactivity of the single control assembly ($\Delta \rho = 0.21\%$) and the temperature coefficient ($\partial \rho / \partial T_{isoT} = 0.0$ pcm/K) were tuned to the enveloping parameters. As the feedbacks are important, the beginning of equilibrium cycle is chosen for the transient.

The calculation has been performed with the KIKO3D/ATHLET coupled code. The sequence of the calculation procedure and the correspondent core parameters are summarized in Table 2. The withdrawal of the asymmetric control rod takes 62.5 s, while the reactor power increases (see Fig. 7) so do the temperature and pressure of the primary system, too. There is no mixing in the upper plenum that is why the special scram signal activated from the high outlet temperature (310 °C) in the loop is not accepted in this simulation. Due to the reactivity insertion, a new state is developed at the higher permitted power level at about 100 s. Because of negative feedbacks, the reactor turns back to a state close its original parameters very slowly.

Table 2 Chronology of the events during the uncontrolled withdrawal of a control assembly

<table>
<thead>
<tr>
<th>Time (s)</th>
<th>Event</th>
</tr>
</thead>
<tbody>
<tr>
<td>0.0</td>
<td>94% of nominal power/ Position of the control group $H_c = 125$ cm/ Core inlet temperature $T_{in} = 269$ °C. An asymmetric control assembly starts to go up</td>
</tr>
<tr>
<td>53.6</td>
<td>Outlet temperature of the 1. loop &gt; 310 °C, scram is not activated</td>
</tr>
<tr>
<td>60.4</td>
<td>The pressurizer spray control is switched on in the 1. loop</td>
</tr>
<tr>
<td>62.5</td>
<td>The moving control assembly reaches its upper position $H_c = 250$ cm, the power maximum in the core is 108.8% of nominal value</td>
</tr>
<tr>
<td>78.1</td>
<td>Pressure in the primary system &lt;11.8 MPa (could have been ECCS signal, but failed)</td>
</tr>
<tr>
<td>78.1</td>
<td>PRZ Spray control is switching on/off.</td>
</tr>
<tr>
<td>167.8</td>
<td>PRZ Spray control has been switched off, last</td>
</tr>
</tbody>
</table>

The reactivity insertion strongly increases the power level in the affected sector (see Fig. 7), therefore hot channel analyses for the most loaded assembly has to be done. The heat transfer strongly depends not only on the power, but also the temperature and pressure of the coolant. That is why the pressure in the primary circuit (see Fig. 8) was set conservatively low value and its increase during the transient was kept as low as possible by activating the pressurizer spray control. To keep the transient more conservative, in another simulation a non-conventional power-manoeuvring mode was used. In that case, the main steam collector pressure kept constant by changing the throttle of the steam line [the automatic control system] (instead of power). In such a case, the temperature in the cold leg is increasing and the power in the affected sector is higher due to the feedback, than in the first case. It was turned out, that the second scenario is the more dangerous. The simulated IC signals of the detectors positioned properly show the asymmetry of the power generation, too.

The currents of the two groups of the IC’s (their positions can be seen in Fig. 2) are presented in Figure 10. The asymmetric character can be realized. From the signals it can be concluded that taking into account the asymmetry in the “two from three” logics, the scram actuation would not occur.

The most loaded assembly of the core was chosen for hot channel analyses. The calculation was made by the TRABCO code, which used the time dependent flow, the inlet enthalpy, and outlet pressure of the assembly from the ATHLET calculation and the axial power from KIKO3D. Boiling crisis did not occur, the DNB ratio (Gidropress correlation) was higher than 2.0 (see Fig. 9).
Fig. 7: Power of the six core segments

Fig. 8: Pressure in the core segments and the pressurizer

Fig. 9: DNBR in the hottest subchannel

Fig. 10: Currents of the 2 sets of IC’s during control rod withdrawal
5 CONCLUSIONS

This study was intended to investigate the behaviour of the reactor when it is undergoing “the initial phase of main steam line break” and “the withdrawal of one control rod” events using the ATHLET/KIKO3D/TRABCO code. At present this type of perturbations at a VVER-440 type reactor were reanalysed as a part of the plant modernisation project, including of new type of fuel and a power upgrading. The acceptance criteria were fulfilled in all cases.

Earlier the control rod manoeuvring was analysed by a steady state code, as the reactivity change is very slow. This methodology gives similar result, however this simulation is more realistic.

REFERENCES


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