STUDY OF ACCELERATED UNIT UNLOADING MODE INITIATED BY TURBINE FEED PUMP TRIP WITH TVSA FUEL ASSEMBLIES OPERATION IN WWER-1000

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This paper provides the study results of accelerated unit unloading mode (AUU) initiated at WWER-1000 unit operated at 100 % power and its expediency in the event of single Turbo Feed Pump (TFP) failure. Modeling was performed using an advanced calculation code RELAP/SCDAPSIM/Mod3.4 and relevant model for KhNPP Unit No. 2. As the study shows, SCRAM cannot be prevented in case of failure of 3 main circulation pumps (MCPs) due to steam generators (SG) level drop. Based on the results obtained, it is reasonably justified to allow SCRAM signal instead of AUU activation in case of single TFP failure at power level more than 90 % of N_{nom}. This will provide more sparing temperature modes for fuel assemblies and equipment, as well as prevent additional thermal cycling loads and violation of safe operation limits as SG water levels.

1. Introduction

To attain high technical and economic indices of electricity generation from NPPs aimed, inter alia, at providing competitiveness generating utilities all over the world seek to reduce the fuel component, which can reach 30 % of the total electricity generating cost. This makes the tasks of optimizing the fuel component obvious [1]. One of the main ways to solve the problem of optimizing the fuel component is to increase the level of nuclear fuel burnup. At the same time, fuel manufacturers tend both to increase an initial enrichment of nuclear fuel and an amount of fuel in the core. However, alteration of water-uranium ratio may also change the coefficient of reactivity, i.e. higher fuel burnup enlarges a reactivity coefficient range. These changes can be significant and lead to deviations from the anticipated behavior of a number of transients in the reactor facility (RF), including neutron power change rate due to feedback action. This paper provides the modeling results and analysis of AUU behavior induced by TFP failure in the WWER-1000 unit at N_{nom} = 100 %.

The variation range of core reactivity coefficient during the fuel campaign with water-uranium ratio specific for alternative fuel assemblies, such as TVSA, used at Ukrainian NPPs with WWER-1000 units is much greater compared to previous generation fuel. Such trend will continue in future when using next generation FAs, in particular fuel rods with increased fuel column height or pellets without central opening. An increase of reactivity coefficients in absolute value has significant impact on AUU anticipated behavior.

In recent years, several NPPs with WWER-1000 units operating in AUU mode have recorded significantly higher neutron power excursion than ever observed since the start of WWER-1000 operation. Despite the fact that such modes proceed in the subcritical state the AUU operation is accompanied by reaching safe operation limits for neutron power growth rate leading to drop of SG levels below SCRAM setpoints.

2. Problem description

Since 2007 a number of NPPs with WWER-1000 units have recorded activation of AUU mode reaching SCRAM setpoints by period of reactor that has not been previously observed. On the one side, this could be due to introduction of the state-of-the-art neutron flux monitoring system (NFMS), which, unlike previous generations, allows to keep records of more rapid and dynamic processes with specific times of an order of seconds.

On the other side, the parameters of neutron power changes:

- level of neutron power drop upon AUU activation;
- rate of neutron power raise due to the feedback action;
- level of neutron power excursion,

differ significantly from the neutron power behavior (Fig. 1) recorded under PHARE SRR 1/95 [2] that addressed AUU mode in the WWER-1000 as an international test problem investigated in various scientific centers using different software programs (Table 1).

Table 1. **List of institutes and verified programs under PHARE SRR 1/95**

Institute	Code
FZR, Germany	DYN3D/ATHLET
VTT, Finland	HEXTRAN/SMABRE
KI, Russia	BIPR8KN/ATHLET
STCNRC, Ukraine	DYN3D/ATHLET
INRNE, Bulgaria	DYN3D/ATHLET

The reasons for such differences include, inter alia, as follows:

- decrease of mean drop time for control rod (CR) of control and protection system (CPS) from ~ 3 s in the late 90's to ~ 1.5 s now;
- increase in absolute value of reactivity coefficients by coolant and fuel temperature.

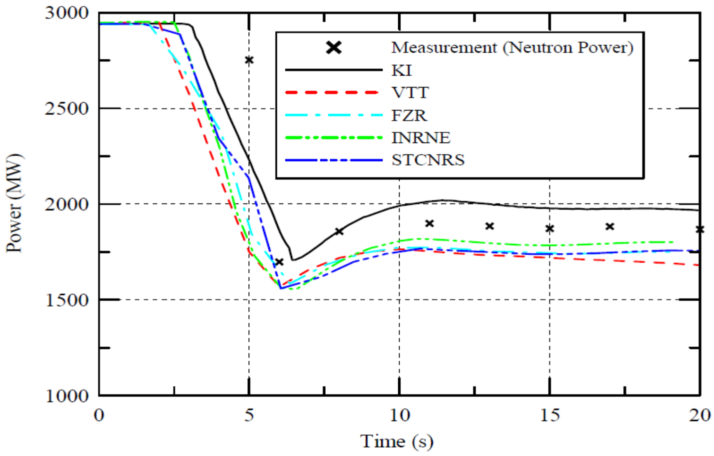


Fig. 1. Comparison of the measured power variation with calculation results obtained using different programs for AUU mode. The results of dynamic tests performed during Balakovo Unit 4 startup were used as the experimental data.

The [2] specifies that there are no reasons to consider AUU mode dangerous in terms of nuclear safety and fuel thermal reliability.

The validity of this statement for AUU mode recorded, for example, at Khmelnitsky Unit No. 2 in 2007 with some parameters having significant difference (Table 2) from the study results [2], should be the subject of further investigation.

Table 2. **Some parameters of reactor in AUU mode**

Parameter	Design of SRR 1/95	KhNPP-2, 22.09.2007
Drop time of AUU control group, s	3.5	1.37
Minimum neutron power, %	52	48
Maximum neutron power during feedback action, %	60	70 as per design 61 – SCRAM actuated
Maximum rate of neutron power raise	62 – 100 MW/s	162 MW/s

After the AUU mode is actuated the operation of WWER-1000 series power unit of V-320 type is accompanied by dropping of CPS control group named an AUU group and triggering of reactor power unloading and limiting device that should render unit to 50 % power level by interacting with CPS control group. An alteration of neutron power in AUU mode depends on efficiency of AUU group and its drop time having significant impact on feedback action intensity. The decrease of the drop time for AUU control group from 3.5s to 1.37s will increase the power ascension rate from 62-100 MW/s [2] to 162 MW/s [3 - 6]. It should also be noted that neutron power ascension rate significantly depends on feedback parameters in terms of reactivity coefficients and their time constants. For example, the reactivity coefficient by coolant temperature changes in absolute values in ~ 2 times during the fuel campaign from the start till the end of boron control.

Therefore, AUU behavior at KhNPP Unit No. 2 c significantly different from that simulated under the Project SRR 1/95. The most significant concern relates to the so called neutron power excursion that can exceed 20 % without account of further power raise due to colder coolant coming as a result of closing primary circulation.

The power excursion mode up to 20 % is provided for in the fuel documentation [7, 8]. The number of such excursions during the design lifetime is 15 for usual fuel assemblies (FA) and 20 for TVSA.

Therefore, significant divergences between the anticipated behavior of AUU mode and the actual behavior observed recently at a number of WWER-1000 units [3 - 6, 9] generate specific interest to simulating this mode.

3. Computer code and the model used in the investigations

In this paper, for modeling AUU mode in case of single TFP failure the thermal-hydraulic code RELAPSCDAPSIM/Mod3.4 with point reactor kinetics model was used [10]. Such approach is applicable due to absence at AUU asymmetric perturbation of the neutron field in the reactor core.

In order to study the behavior of processes with reducing neutron reactor power (working in a subcritical state), the model of WWER-1000 for the RELAP5/Mod3.2 code was applied. The calculations at the same time were performed with using the RELAPSCDAPSIM/Mod3.4 code, which is a fusion of later RELAP5/Mod3.x versions and SCDAP models. The RELAP5 models allow to calculate the general thermal hydraulics, the interaction of control systems, reactor kinetics and transport of non-condensed gases. The SCDAP models simulate the core behavior during a severe accident.

RELAPSCDAPSIM/Mod3.4 code allows to perform a realistic analysis of the design, beyond the design and severe accidents with severe core damage before the destruction of the reactor vessel, at the same time modeled in detail the processes in the reactor facility. The neutron kinetics is presented in the point approximation, it does not take into account changes in the spatial profile of energy release during local reactivity changes.

For the calculations the three loops WWER-1000 model of the KhNPP Unit No. 2 (loop 3 is double and corresponds to the 2-nd and 3-rd real loops) was used. It was previously developed and validated in framework of the safety analysis work [11]. The model includes the basic equipment of the normal operation systems and safety systems of the first and second circuits. Also in the model takes into account changes in thermal-hydraulic characteristics of the first reactor circuit caused by the operation TVSA in the core, moreover clarified the characteristics of the gas gap fuel element.

Particular attention in model was paid to the correct description of the reactor kinetics input data, since they have the greatest influence to the results of the modes simulation associated with the reactivity change. At the same time the neutron-physical characteristics for the beginning- and end-of-core-lifetime of seventh fuel loading on KhNPP Unit No. 2 [12] was used.

The reactivity at transient is defining as the sum of the reactivity inserted by the displacement of the CR CPS, and released reactivity due to feedback action. During operation of the seventh fuel loading on KhNPP-2 in case of AUU activation drop into the core of the 4th group of CR takes place.

To determine the reactivity, released due to feedback actions, the code takes into account the following reactivity coefficients: reactivity coefficient by average temperature of the fuel, $\frac{\partial \rho}{\partial T_U}$; reactivity coefficient by average coolant temperature without changing the density, $\frac{\partial \rho}{\partial T_{H2O}}$; reactivity coefficient by coolant density, $\frac{\partial \rho}{\partial T_{\gamma}}$.

To account for the radial distribution of energy release, the reactor core is conventionally divided into two unequal volumes, at one of which the heat exchange of coolant with a "hot" fuel assemblies takes place and at other heat exchange with the "average" fuel assemblies. In addition, at the hot and average FA select "hot" and "average" fuel elements, this is also taken into account when modeling the heat exchange. In this case the coefficient of nonuniformity of energy release by fuel assemblies of the core k_q and coefficient of nonuniformity of energy release by fuel elements of the core k_r is used.

To account for the distribution of energy release by height, the reactor core in the model, namely the volume in which heat exchange of coolant with hot and average FA takes place, divided into 10 equal control volumes. For each such volume calculated weights coefficients, according to which the defined contribution to the total reactivity some reactivity released in a separate volume, when is changing the thermal-hydraulic parameters of the coolant in it.

4. Calculation results

Using the model, which takes into account features discussed in the previous section, a study AUU mode initiated by single FWP failure was performed. For this mode the calculation results of the neutron reactor power are shown in Fig. 2, which shows how, after a rapid decrease in neutron power to ~ 50 % Nnom due to fall of AUU group (time of fall ~ 1.3 s), under the feedback action is a growth of neutron power to a level more than 67 % Nnom that differ significantly from the simulation results obtained in the framework of the project [2].

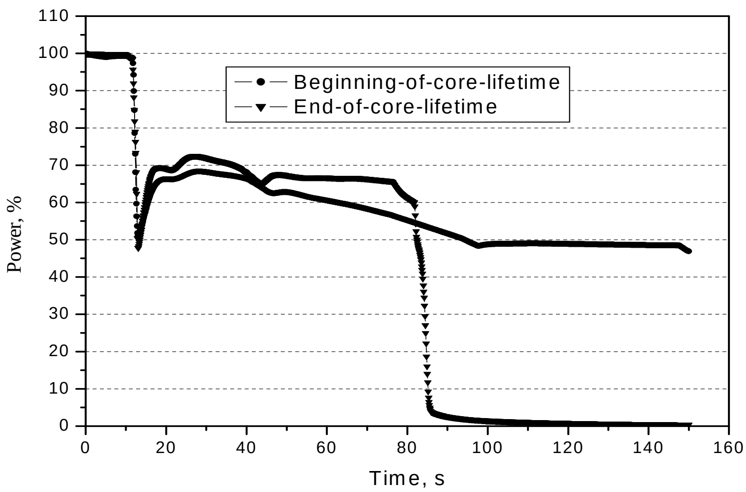
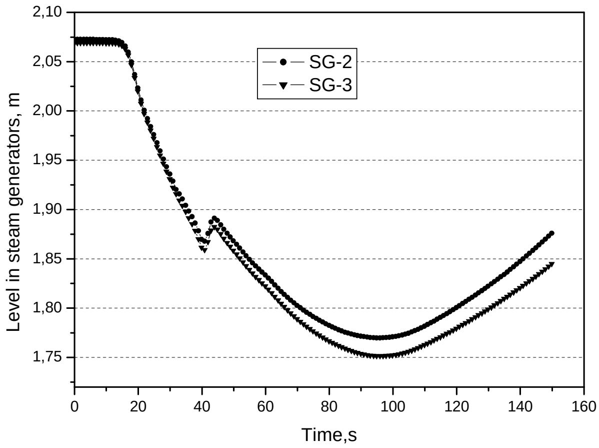


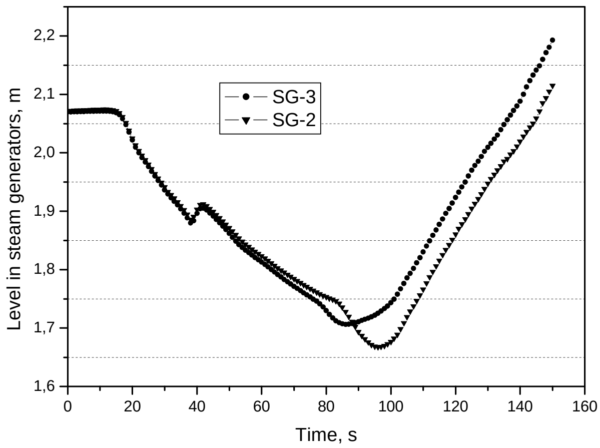
Fig. 2. Changes of reactor neutron power in case of single TFP failure for the beginning- and end-of-core-lifetime.

Simultaneously with AUU activation after TFP failure begins to drop rapidly the water level in the steam generators (Figs. 3 and 4) due to the decrease of SG feeding. When the level becomes below 500 mm from the nominal 2.25 m, as shown in Fig. 4, according to the operating limitations [13] in the emergency mode shuts down the corresponding MCP. In case of failure of 3 MCPs the scram activation takes place (by a condition «Failure of the 3 MCPs from the 4 which is working at power > 5 %»). Such AUU mode behavior for the end-of-core-lifetime (see Fig. 2), due to large integrated power released at transient as a result of the feedback action by reactivity, despite the work of unloading power system (UPS), and the rate of water evaporation exceeds the rate of SG feeding.

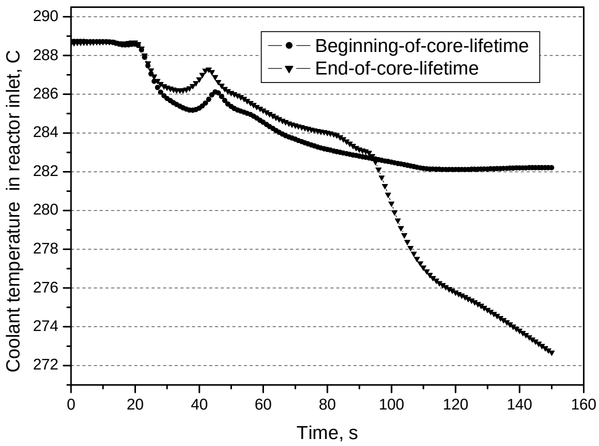
At single TFP failure from two working and AUU activation, also signal to close the turbine control valves in order to reduce the turbine unit power to 55 % Nnom. In same time turbine electro-hydraulic control system (EHCS), operates in the PC-1 mode of maintaining the pressure in the main steam collector (MSC) - [14]. Steam relief device such as BRU-A and BRU-K are closed.

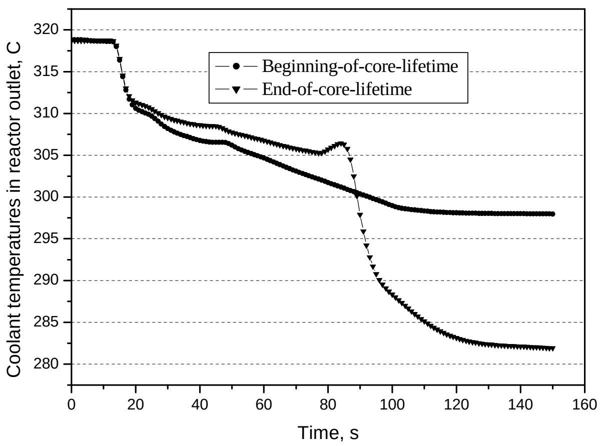
The coolant temperature changes at the inlet and outlet of the reactor is shown in Figs. 5 and 6.





Figs. 3 and 4. Levels in SG-2 and SG-3 in case of single TFP failure for the beginning- and end-of-core-lifetime.





Figs. 5 and 6. The coolant temperature at the inlet and outlet of the reactor for the beginning- and end-of-core-lifetime.

The coolant temperature growth at the outlet of the reactor after 75s for the end-of-core-lifetime due to a sharp decrease of water flow through the reactor. This associated with third MCP failure due to a reduction in the water level of the corresponding generator and achievement of setpoint 1.75 m. Such temperature growth continues until the scram activation, caused by third MCP failure. In Fig. 2, this corresponds to region of a power reduction after 75 seconds to the time of scram activation.

5. Conclusions

According to WWER-1000 design in case of single TFP failure at power more than 75 % of nominal, in order to transfer power unit quickly to a lower power level the AUU mode is made. In AUU mode the AUU group of control rods drops into the core, at the same time the UPS begin work and power is corrected to level of 50 % of Nnom. Simultaneously with the reactor unloading the turbine unloading takes place, that is, steam power and electric power (as a consequence) of the power unit is reduced. At the same time provides stabilization of the pressure in the main steam collector according to a specific mode [14]. According to the design when AUU triggered the power unit must move to a new steady state at half nominal power.

However the results of analysis which was carried out in this work for the seventh fuel loading of KhNPP Unit No. 2, shows that during AUU activation due to TFP failure the neutron reactor power increases to 73 % Nnom after rapid drop. In this case, the efficiency of the UPS is not enough for rapidly reduce the heat power to ~ 50 %, and TFP works cannot to maintain boiler water level at SG. Further, during evaporation the water level in the steam generators is reduced to emergency setpoints which corresponding MCP failure. After the three MCP failures the SCRAM activation takes place. According to the received results of calculations, such transient takes place at end-of-core-lifetime.

Therefore, since SCRAM cannot be prevented after AUU activation, in this case, the AUU mode is not expedient because it leads to additional thermal cycling loads of fuel and equipment, and also violated the safe operation limits as the SG levels.

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