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Physics

SENSITIVITY ANALYSIS OF THE FIRST CIRCUIT OF COLD CHANNEL PIPELINE RUPTURE SIZE FOR WWER 440/270 REACTOR

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This calculation describes an accident to analyze and study the emergency instructions in case of fracture in the airtight zone of the first circuit. Sensitivity analysis was performed to determine the extent of the fracture, when a high-pressure feeder pump remains $100N/cm^2$ higher than the saturation pressure in case of emergency pressure of the circuit (the pressure drop allows the main circulating pumps work). The conventional diameter of the fracture is 29 *mm* and the schedule of the development of events is given in the study.

***Keywords*:** WWER reactor, high-pressure feeder pump, relap5, pressure compensator.

Introduction. Safety in nuclear energy is of primary importance, especially when power plant is situated close to megapolices or other inhabited areas. The main risk is associated with radioactive emissions and their environmental impact. In the present reliable and secure operation of the power units is at high level. Strict adherence to the correct operating conditions and instructions worked out over the years, on the basis of analyzing the causes of accidents and emergencies is essential for security, including the Armenian NPP. Methods of assessing the safety help to understand and to find the right solutions in nonstandard situations, anticipate possible developments in case of failure of equipment. One must carefully consider not only the "normal" state of emergency, but also the effects of the simultaneous failure of several independent systems as a result of improbable occurances. The main principle of preventing accidents is to ensure the reliability of the equipment, normal operation of systems and highly qualified staff.

The basis of the methodology of the security includes the estimation of possibility of accidents, as well as the development of the initiative emergency procedures.

In the paper are worked out instructions for the analyzes and study of accident in case of the rupture in the first circuit airtight zone [1–3].

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In Table the settings of initial conditions of the reactor are given.

Settings of initial conditions of the reactor

Parameters	Nominal values	Working area	Error, %
Power, MW			
Active zone, 6 loop	1375.00	0 – 1402.500	27.500
Active zone, 5 loop	1141.25	0 – 1164.075	22.825
Active zone, 4 loop	921.25	0 – 939.675	18.425
Active zone, 3 loop	687.50	0 – 701.250	13.750
Linear power, W/cm	–	325.0	–
Residual warmth (% of ANS 79)	100	120	–
Balancing the starting position, m	1.75 – 2.0	1.50 – 2.25	0.005
Higher volume pressure, MPa	12.35	12.15	0.157
The level at pressurizer, m	5.12	4.82 – 5.42	0.028
The flow, m^3/h			
Reactor	42000	42000 – 43020	–
The reactor's active zone	39060	39060 – 40000	–
Detour emergency protection	2940	2940 – 3020	–
The collection of fuel	111.9	111.9 – 114.6	–
Self managed cassettes	111.9	111.9 – 114.6	–
Temperature, $^\circ C$			
In the reactor entrance, $^\circ C$	4.61	260 – 268	2
In the reactor exit, $^\circ C$	297	260 – 297	2
Δt $^\circ C$	29	0 – 29	0.5
Steam generators:			
Pressure, MPa	4.61	4.41 – 4.80	0.041
Level, m	2.12	2.12 – 2.295	0.028
Warmth bearing weight, t	31.5	31.5	–
Replenishing water temperature,$^\circ C$	223	155 – 225	1.5
Replenishing water flow, t/h	456	493.2 – 478.8	22.8
Pressure in main steam collector, MPa	4.5	4.31 – 4.6	0.041
Boric acid Emergency tank			
Volume, m^3	≥ 800	800 – 1050	16
Temperature, $^\circ C$	≤ 55	55 – 60	0.2
[H_3BO_3], g/kg	≤ 12	12 – 15	1.5

Calculation and Results. Calculations in the present work were done by RELAP5/MOD3 program, which analyzes the behavior of thermal-hydraulic systems operating light water. The RELAP5 code has been developed for analysis of thermal-hydraulic complex interactions that occur as a result of small or large leak of coolant from the system.

The code models the coupled behavior of the reactor coolant system and the core for loss-of-coolant accidents and operational transients such as anticipated transient without scram, loss of offsite power, loss of feedwater and loss of flow. A generic modeling approach is used that permits simulating a variety of thermal hydraulic systems. Control system and secondary system components are included to permit modeling of plant controls, turbines, condensers and secondary feedwater systems. The program is able to study the issue in greater details. So, it has been successfully used to analyze both large and small leaks LOCAS, as well as the operation of various experimental and industrial reactors (PWR, in transition). The software is also uses (to a lesser extent) boiling water reactor (BWR) system for analysis. The model component is a friction between phases: different flow regimes in vertical and horizontal groups. The program has a kinetic point model to simulate the behavior of reactivity. The program also has the opportunity to simulate the presence of a liquid and substance nearby. So, the material can model the transference of the energy to the substance and vice versa. The control system allows us to model the equipment elements, to balance and block devices (for example, turbines, pumps, capacitors) and common-mode version of the process (for example, a heat transfer volume to another). RELAP5 / MOD3 code has no apparent opportunity for (a) the equation group created by RELAP5 / MOD3, thus, examined the behavior of the system in light water and unbalanced conditions, which is not homogeneous in terms of the actual pressure in any of the critical atmospheric conditions; (b) the kinetic point model allows analysis of the various transition processes in the future without an overload block (ATWS), as well as the retrogressive link of thermal hydraulic and neutron effect; (c) the modeled capabilities of the control system allows a wide range of simulated model of the balance of any nuclear power plant components [4–7].

The total duration of calculation is 18000 *s*. The emergency cooling system of ANPP consists of two channels. Each channel consists of 3 high-pressure pumps (main circulating pump (MCP) $1 \div 6$), supplying water to an emergency cooling zone. Transition process begins in 0 seconds cause of the 29 *mm* gap. The first circuit pressure drop occurs (Fig. 1) and the level in pressurizer decreases (Fig. 2).

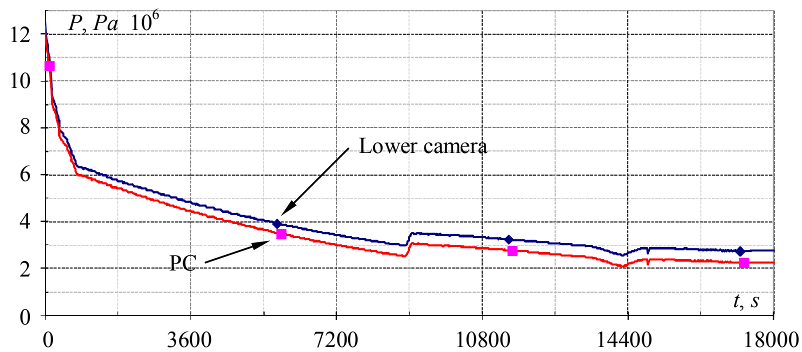

Fig. 1. Pressure in the first circuite.

At the 15.868 *s* FP-1 turned on (Fig. 3) (water level in pressurizer dropped 300 *mm* from nominal value), at the 24.699 *s* turns on FP-2 (water level in pressurizer dropped 500mm from nominal value), at the 45.86s turn on in FP-3,4 (pressurizer scales down to 1 meter from nominal value). At the 121.574s appears the "Reactor Stopped" beating signal based on the water level falling to 2.56m and lowering the pressure, in this situation the emergency power pump (EPP) is switched on and FPs are disconnected, this is reported by a signal. The heaters of pressurizer are switched off at the 126.227s, because of level reduction 2.7m in pressurizer. 10s after reactor stopped beating signal the valves of locks are closed turbogenerator (TG)-1,2.

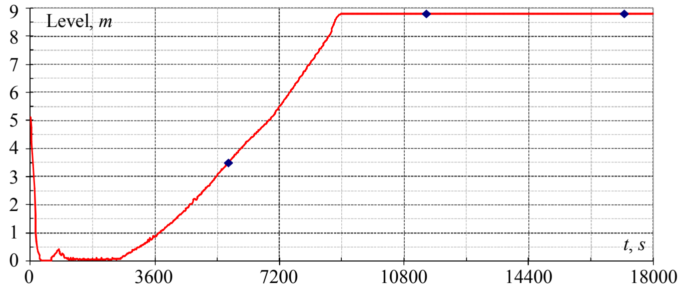

Fig. 2. Water level in PC.

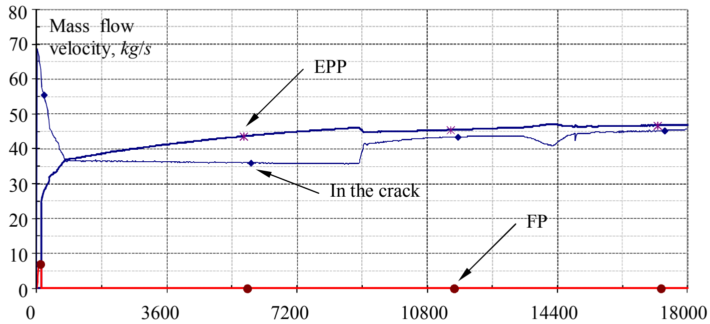

Fig. 3. Mass flow velocity through EPP, FP and crack.

After closing the blocking valve of TG the pressure in main steam collector is growing up till starting of the quick reduction device of the steam in atmosphere (QRDS-A)-1 at the 140s (Fig. 4).

Starting at 600s the operator is regularly opening QRDS-A and lowering the temperature of the first circuit at a rate 30°C/h and using the shower in pressurizer lowers the pressure in the first circuit. Starting at 900s flow from emergency feeding pump exceeds the flow through the crack and at 2600s the level in the pressurizer starts rising, at the 9060s pressurizer is completely filled. After pressurizer is filled the pressure in the first circuit increases, simultaneously increases the flow to crack. But the flow of the EPP continues not much to add the flow into the crack, which compensates the reduction in the volume of water first circuit, because of the density increase. Flow into the active zone increases although to existence of the crack (Fig. 5), because decreasing of the temperature of heat carrier and density increase of it. After MCP disconnected the flow to the active zone falls practically to zero.

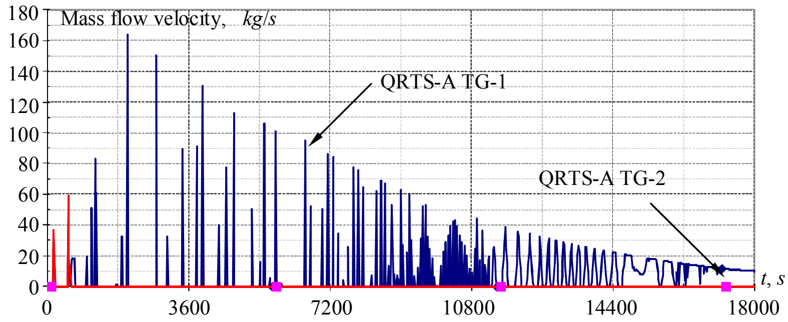

Fig. 4. Mass flow through QRTS-A.

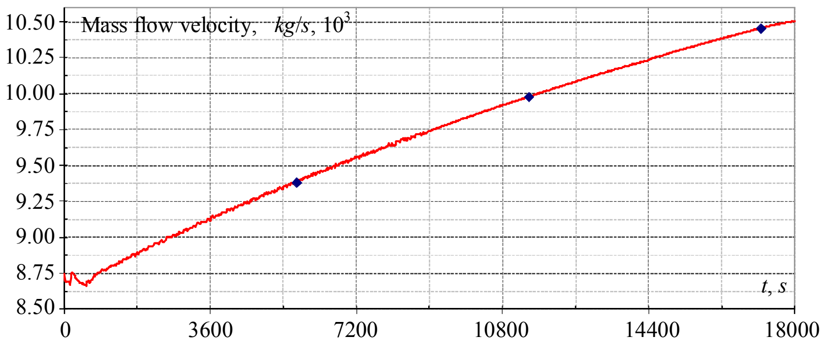

Fig. 5. Mass flow through the active zone.

Conclusion. Calculations show that the action of the operator during cooling and pressure reduction in the primary circuit, enables to restore coolant stock of the first circuit. But the emergency pump flow almost $10 kg/s$ exceeds the flow in the gap, as a result PC is filled water.

When there is no steam bag and the rigid circuit occurs, crack isolation can cause a sharp increase of pressure, can lead to violations of pressure standards in the body of the reactor in the cold state. But the flow through the crack at the pressure remains higher than all possible feeding pumps combined, which makes it impossible to disconnect EPP and connect feeder pumps. When the level in PC is getting to nominal, it is necessary to lower the flow of the EPP through recirculation valve. In this RELAP model the flow of the EPP is only for entire recirculation pomp. It is necessary to conduct additional studies in order to get flow characteristics of EPP combination with the valve open of recirculation EPP.

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