Typical steam generator tube rupture (SGTR) effect on thermo-hydraulic parameters of VVER-1000 primary loop

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Abstract

In operation of nuclear power plant, it is very important to evaluate different accident scenarios in actual plant conditions. One of the main accidents is (SGTR) steam generator tube rupture in the field of nuclear safety. In this research variation of thermo-hydraulics parameters in primary loop under SGTR accident in VVER-1000 nuclear power plant is analyzed by Relap5/Mod 3.2 thermo-hydraulics code.

In simulation of this accident, it was supposed that after establishment of steady-state condition in the system, instantaneous break with equivalent diameter of 100mm in the area of lower row heat exchanging tubes is created [8]. The accident scenario is assumed as the most conservative version of damage, with loss off-site power and two diesel generators of in-site power in loops 1and 2 of the primary cycles. The results show that after initiating of accident, the pressure above the core decrease from 16 MPa to 8 MPa during 130s and finally, it will stabilize about 7.1 MPa . The coolant temperature at the reactor outlet decrease from 598 k° to 500 k° during transient. Also, primary-to-secondary coolant leakage reduces from 800 Kg/sec to approximate 0.0 kg/sec during first 150 sec of the accident .

As it is seen from RELAP5 predicts correctly the behavior of main plant parameters in comparison with the reported PSAR data.

Keywords: VVER-1000 , Steam Generator, Tube Rupture, thermo-hydraulics code

Nomenclature

SG	Steam generator
RCP	Reactor coolant pump
DG	Diesel generator
TG	Turbine generator
RCC	Reactor collection chamber
MSIV	Main steam isolation valve
PRZ	Pressurizer
RP	Reactor plant
PSD	Pulse safety device
EFWP	Emergency feed water pump
NPP	Nuclear power plant
HPIS	High pressure injection system

1. Introduction

As a result of primary-to-secondary leak there is a long-term loss of coolant beyond the boundaries of the containment. The consequence of this accident is radioactive release into the atmosphere and probable loss of reactor core cooling. Prevention (or significant decrease) from release of radioactive coolant through steam dump devices of affected SG is achieved by introduction of a special automatic algorithm of accident management at the initial stage of the accident which is a certain sequence of actuation of various systems.

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The algorithm is started by a special signal after identification of the accident as primary-to-secondary leak. It makes possible to begin automatic execution of the complex of measures aimed at localizing the releases. The process actions within the framework of automatic algorithm are directed to pressure maintaining in the RP primary system at the level which makes possible to prevent from opening the safety valve of affected SG. Such actions are injection of boron solution into PRZ by pumps of additional boron injection system and accelerated RP cool down through BRU-A of intact SGs.

In this research the reference power plant is Bushehr NPP site. Operational data from this NPP are available for the purpose of assessing how the RELAP5 model compares against the plant data.There were several modeling by Relap5/mod 3.2 done relate to VVER-1000 NPP up to now. Some of them are, Simulation of Loss-Of-Flow Transient in a VVER-1000 Nuclear Power Plant with RELAP5/MOD3.2 [2], Analysis of natural circulation phenomena in VVER-1000 [3], simulation of the effect of a small-break loss of coolant accident (SBLOCA) transient on VVER-1000 reactor pressure vessel parameters [4] .

2. Description of the transient

At the beginning of the accident the thermal power was 3120 MW. The initiating event is instantaneous break with equivalent diameter of 100 mm of SG2 cold collector in the area of lower row of heat exchanging tubes. It results in abrupt primary pressure and pressurizer water level decreasing, initiated by the leak of the primary coolant to secodndary side of affected steam generator through break of steam generator cold collector [1]. As a result of increase in boiler water level in affected SG by 200 mm from the nominal value, RCP set of loop 2 is tripped at 9,75 s. Loss of coolant in the primary system results in pressure decrease in it up to the setpoint of reactor scram at 11,2 s of the transient (pressure decrease at the core outlet to 14,7 MPa at reactor power more than 75 % against the nominal one).

Coolant leak from the primary system into the affected steam generator results in signal generation at 10 s by the fact of increase in gamma - background in the affected SG steam line. Level in affected SG2 increases by 300 mm from the nominal value at 10,5 s and at 11,75 s (1,0 - time of signal generation, 0,25 s - the time of signal passing in electrical circuits) of the transient, a signal “ primary-to-secondary leak” is generated when two process parameters reach the established values.

At 12,8 s DG start-up with subsequent stepwise loading takes place with a time delay of 2 s in response to a signal of loss-of-power to the sections of auxiliaries. Closing of the TG stop valves leads to a sudden rise of the secondary coolant pressure and BRU-A opening on the affected SG 2 steamline at 13,5 s of the transient (BRU-A open on SG1, 3, 4 steam lines by the signal from automatic algorithm).

As a result of primary-to-secondary coolant leak, operation of steam dump devices of the secondary side and reactor scram, the primary pressure is decreasing, coolant saturation temperature is simultaneously decreasing too. Within time interval of 35,0 - 40,0 s, difference between saturation temperature at the core outlet and the temperature in hot legs of loops 1,3,4 decreases to 10° C that results in signal generation for start-up of safety systems, closing of localizing valves of the containment and termination of boron solution supply from makeup - blowdown system of the primary system due to closing of isolation valves on the pipelines of this system.At 45 s of the transient, the pressurizer becomes empty. Coolant boiling in the RCC starts that slows down decrease in the primary coolant pressure and boron solution injection by two emergency injection pumps starting from 140 s leads to its stabilization at level of 7,1 MPa. Due to the primary leak, the affected SG level rises that leads to the beginning of steam-water mixture outflowing through BRU-A on the SG2 steam line at 28 s of the transient. BRU-A on the steam line SG2 is closed at 130,0 s due to pressure decrease above the core below 8,0 MPa. Consequently, steam-water mixture outflows through BRU-A till this time.

MSIV closing on the affected SG isolates it from the rest steam system. After isolation of affected SG the pressure begins to grow in it and it becomes practically equal to the primary pressure. After BRU-A closing on emergency SG only the safety valve could open.The emergency feedwater is supplied only in response to a signal of boiler water collapsed level lowering in SG to Hnom minus 900 mm not earlier 1000 s of accident process. Under these conditions, two channels of additional boron injection system, functioning for injection into PRZ, are not enough for pressure maintaining in the RP primary system at the level that makes possible to prevent opening the safety valve of affected SG2.

After 200 s increase in the primary pressure and affected steam generator begins that results in reaching the setpoint for opening control SG PSD and outflowing the steam-water mixture at 720 s.

At 1000 s of the accident process supply of emergency feedwater from EFWP into SG 3,4 begins, by 2200 s of the accident process the level in these steam generators reaches the nominal value. Supply of emergency feedwater from EFWP into SG3, 4 results in cooling of the coolant in the reactor pressure chamber, primary pressure decrease and it prevents from repeated actuation of control SG PSD.Filling of PRZ to the nominal level occurs at 1100 s from the beginning of the transient and, further, the pumps of additional boron injection system are transferred into the condition of maintaining PRZ nominal level (functioning either via recirculation line or for injection into PRZ).

At 1800 s of the accident process with the aim of prevention of loss of circulation the operator opens the valves on the line of emergency gas removal from the reactor to remove steam-gas mixture from the upper places of the equipment and to arrange effective cooldown.

At 2200 s after RCC filling the operator closes the valves on the line of emergency gas removal from the reactor. Opening of valves on the line of emergency gas removal from the reactor top head at 1800 s and operation of additional boron injection system for injection into PRZ results in decrease in the primary pressure to the level of start-up of the pumps of emergency boron injection for leak compensation. Subsequent accelerated cooldown through BRU-A in combination with above mentioned measures results to termination of boiling in the volume of RCC and to thereof filling with coolant at 2180 s.

By 2200 s of the transient the level in emergency SG is stabilized.

In case of primary temperature decrease to 200°, accelerated cooldown stops (at 5000 s of the accident process) according to automatic algorithm. Thus, operation of automatic algorithm and the actions of the operative personnel change over the reactor plant into safe state: boiling in the primary system stops, the primary system and the pressurizer are filled with boron solution, the cooldown of the coolant of the primary and secondary side is performed.

3. Initial conditions

The comparison between the initial conditions of the plant data parameters before initiation of the accident and the RELAP5 calculation at nominal reactor power are shown in Table 1. In this Table , plant data and RELAP5 calculation parameters for steady state have been reached very close to each other. Majority from the RELAP5 initial parameters including total reactor power, primary and secondary pressure Coolant temperature at the reactor inlet & outlet , and other are close to the plant data.

4. Result and discussion

The transient test scenario is modeled using the RELAP5/MOD3.2 computer code and the VVER1000 input model for Bushehr NPP[5]. As the results show, RELAP5 predicts the plant behavior correctly. The chronological sequence of transient events is shown in Table 2.

The transient calculations are compared with the plant data in Figs. 1-6. The calculation was performed up to 5200.0 s of the transient time. Before running of the investigated transient event , the RELAP5 model was run in real plant equilibrium conditions up to 200.0 s to establish steady state conditions at nominal reactor power.

Table 1
Steady state conditions at nominal reactor power

Parameter	Dimension	BNPP value	RELAP5 value
Thermal reactor power	MW	3120	3120
Pressure at the reactor outlet	MPa	15.7 $\pm$0.3	16.0
Pressurizer level	M	7.80	7.80
Coolant temperature at the reactor inlet	K	564^{+2}_{-5}	566
Average coolant temperature at the reactor outlet	K	594 $\pm$5	600
Secondary pressure	MPa	6.27 $\pm$0.1	6.3
Main feed water temperature	K	493 $\pm$5	493
Emergency feed water temperature	K	313	313

The initiating event of this analysis steam generator tube rupture in SG#2. Since the primary system pressure is initially much greater than the steam generator pressure, reactor coolant flows from the primary into the secondary side of the affected steam generator (Fig 2) . from This figure the leakage is considerable at the beginning of accident because of high pressure difference exist between primary & secondary side in break position But after pressure decrease in primary circuit and also pressure increase in secondary side of affected SG in order to closing of MSIV in it. finally, lead to stopping leakage at the end of transient. In response of this loss of reactor coolant, pressurizer level and RCS pressure decrease. Fast depressurization of the primary circuit and rapid increasing of the water level in the afected SG#2 characterize the initial phase of transient. Fig. 3, presents the measured primary pressure during the plant transient event and the calculated one.

The figure indicates good agreement between the plant data and the calculated pressure behavior. Due to fast primary pressure decrease, after the signal "P <14.7 MPa" there is actuation of the Reactor Scram and consequent reactor power decreasing.
this event comes at 11.2 s. Coolant discharge leads to the further rapid decreasing of primary pressure (see Fig. 3) and coolant boiling in the RCC with HPIS injection from 140 s leads to its stabilization at level of 7.1 MPa at the end of transient. In accordance with the designed mode [6] HPIS start to inject borated water with temperature 313 k to cold legs of the third, and fourth loops when the primary pressure decrease to 7.8 MPa. Figs. 4. and 5 , are the coolant temperature in the core inlet and outlet . As seen from the figures there is good agreement between the plant measurements and the calculation results. The core inlet temperature differences between the measured and the calculated results after 2000 s come due to work of BRU-A valves of the corresponding SG.

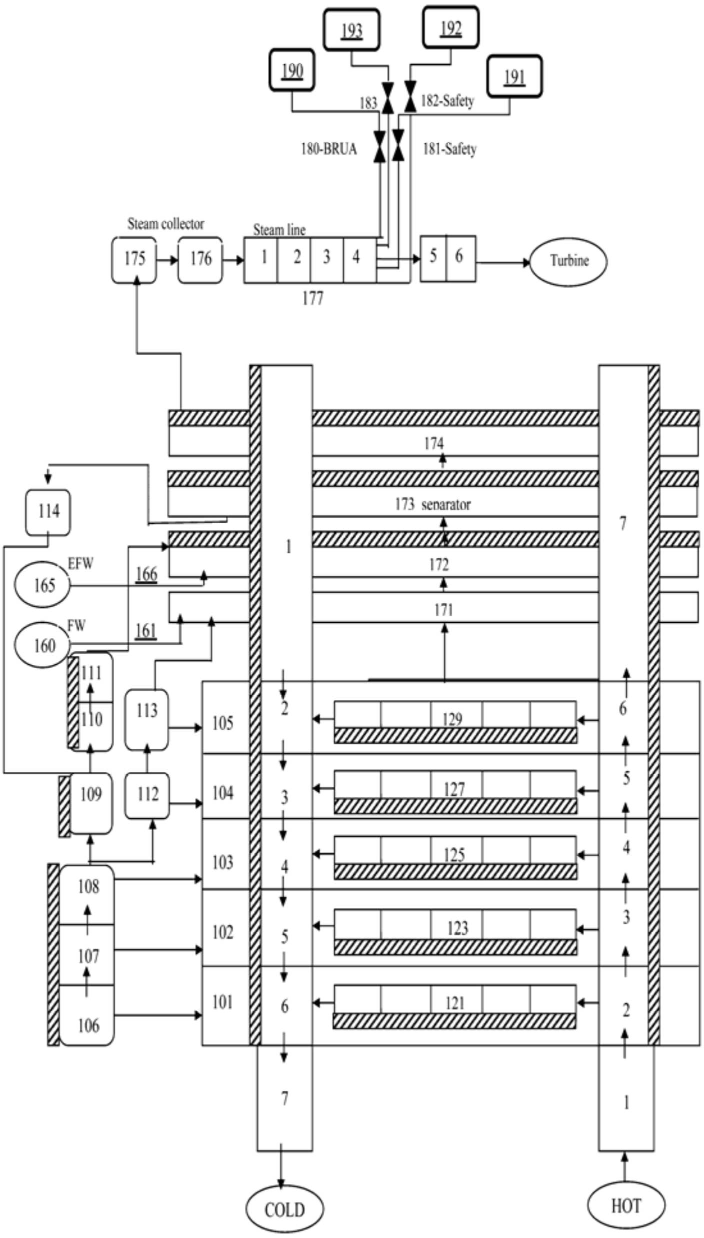

Fig. 1. Nodalization scheme of steam generator .

Table 2
Chronological sequence of transient events

No.	Description of event	PSAR value, sec	RELAP5 calculated results, sec
1.	SG-2 cold collector breaks (100.0 mm equivalent diameter)	0.0	0.0 + 200.0
2.	RCP set trip of affected loop	9.75	9.750 + 200.0
3.	Generation of radiation signal	10.0	No data
4.	loss-of-power supply to the power unit auxiliaries	10.80	10.80 + 200.0
5.	Reactor SCRAM	11.20	11.20 + 200.0
6.	Turbine stop valve closure	11.40	11.40 + 200.0
7.	diesel-generators start-up	12.80	12.80 + 200.0
8.	SG-2 BRU-A opening	13.50	15.0 + 200.0
9.	Additional boron injection start to PRZ	41.75	45.0 + 200.0
10.	End of RCP set coastdown of operable loops	87.0	87.0 + 200.0
11.	Closing of BRU-A of affected SG-2	130.0	130.0 + 200.0
12.	HPIS-(loop 3,4) start injection	140.0	150.0 + 200.0
13.	EFW (SG 3,4) start injection	1000.0	1050.0 + 200.0
14.	Filling of operable SGs 3,4 to the nominal water level	2200.0	2250.0 + 200.0
15.	End of calculation	5000.0	5000.0 + 200.0

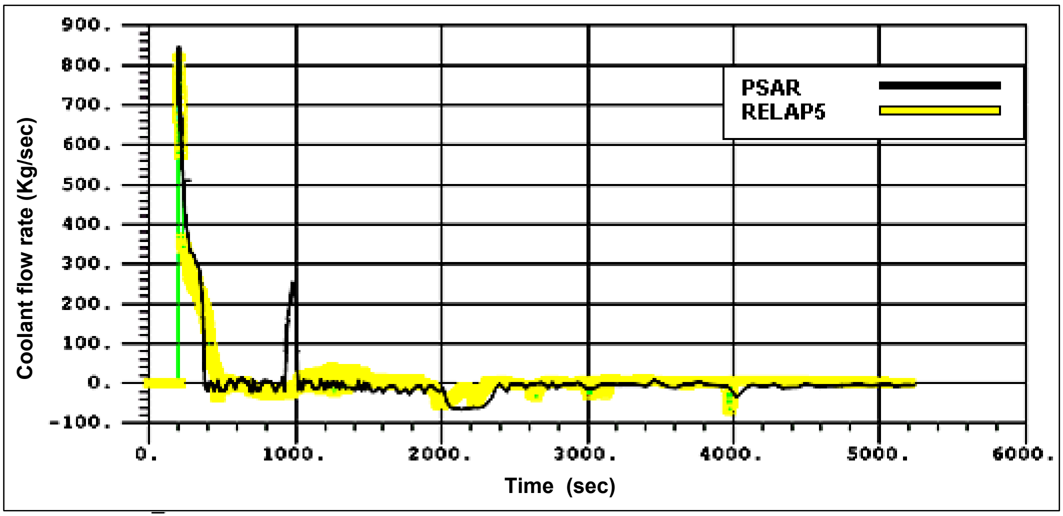

Fig. 2. Primary to secondary leak

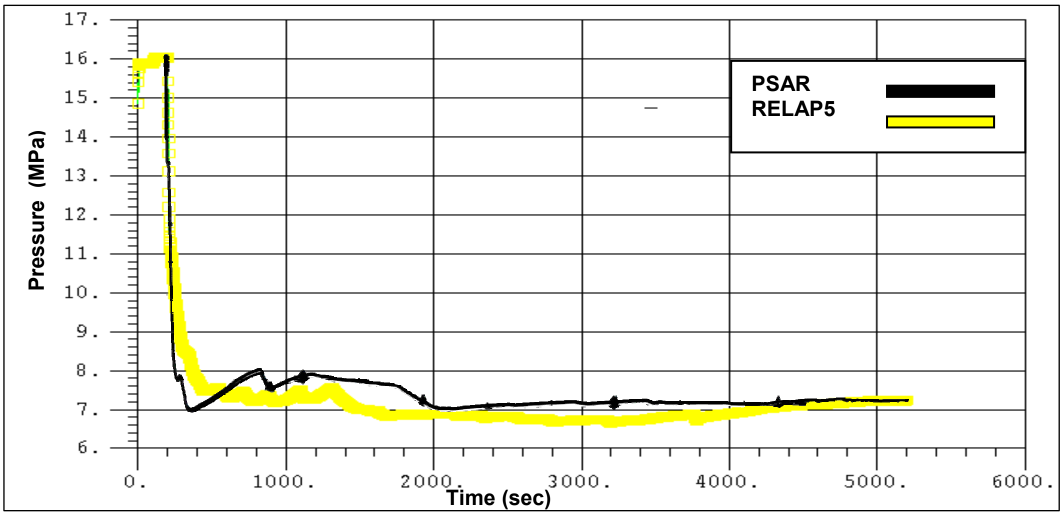

Fig. 3. Pressure at the core outlet

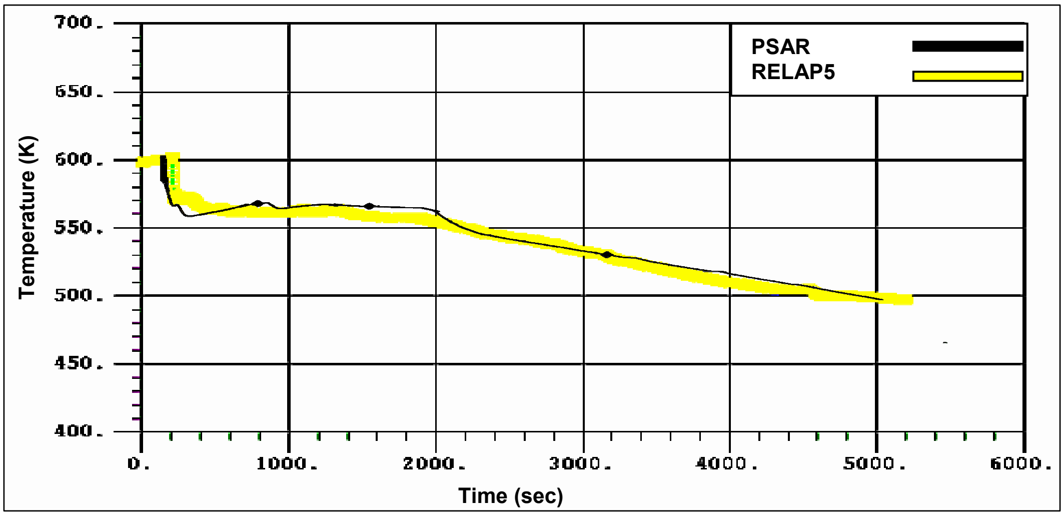

Fig. 4. Coolant temperature at the core outlet

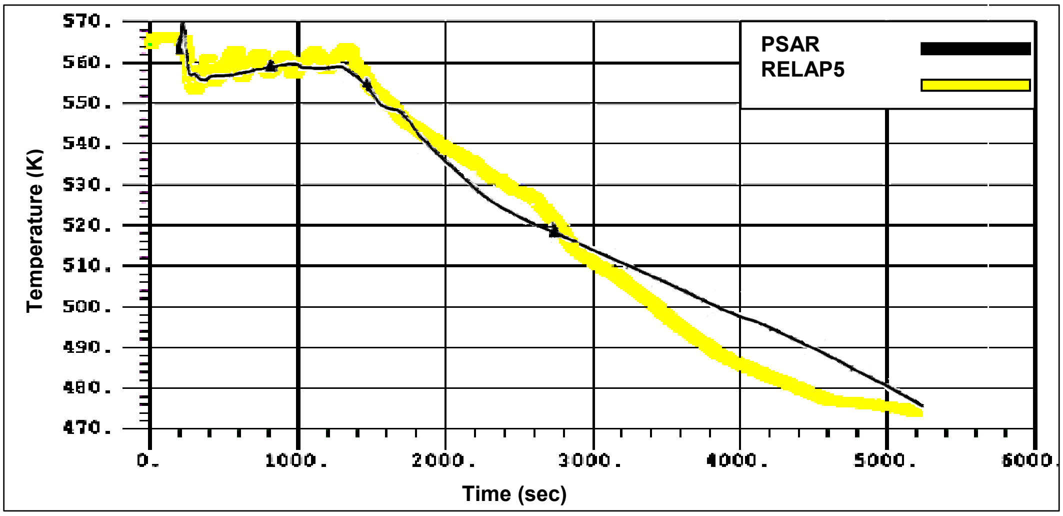

Fig.5. Coolant temperature at the core inlet

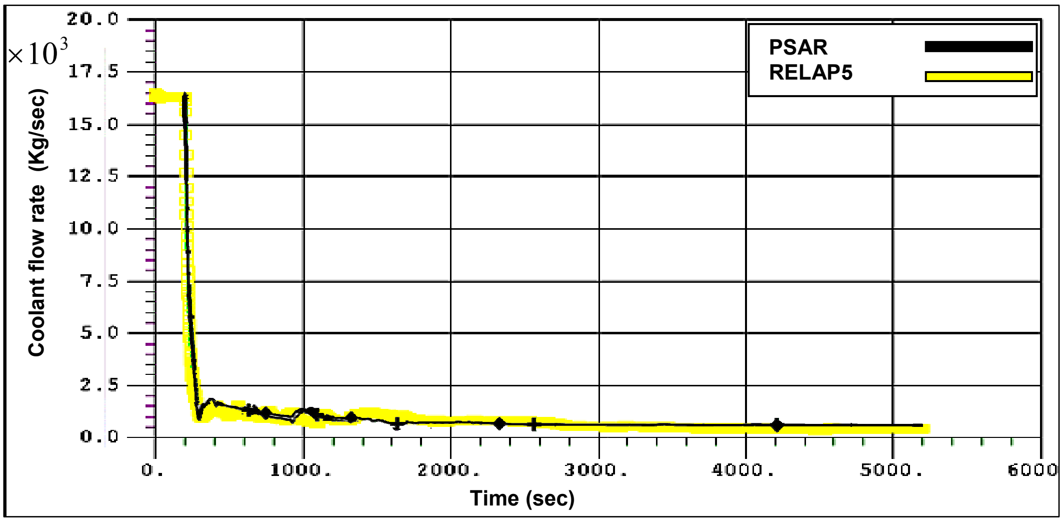

Fig.6. .Coolant flow rate at the core inlet

The comparison between the plant data and the calculated coolant flow rate at the core inlet is presented In fig 6. As seen from the figures the trends of both curves are almost the same . in this figure after end of RCP set coast down of operable loops at 87.0 s , natural circulation will be dominant flow regime to remove residual heat from the core in the following of transient .

5. Conclusions

The RELAP5 model developed for the transient analysis of the performance of VVER-1000 nuclear power plant . It has been used to predict the results obtained during the transient "Primary-to-secondary reactor coolant leakage" as a result of tube rupture with 100 mm equivalent diameter break" in the steam generator .

In this analysis, The results show that after initiating of the accident behavior of the main plant parameters during the transient are changed as the following procedure:

1. pressure in primary coolant system decrease from 16 MPa to approximately 7.1 MPa
2. temperature at the reactor outlet decrease from 598 k° to 500 k°
3. the coolant flow rate at the reactor outlet decrease from 16400 kg/sec to lower than 1000 kg/sec .

Although, Calculation deals with consideration of the SGTR accident is very conservative version of damage, but according to the present results " considered the safety systems" acceptance design criteria is met.

In PSAR of Bushehr NPP, DINAMIKA-97 code was used to evaluate SGTR accident. According to the results obtain in this report , RELAP5 predicts correctly the behavior of main plant parameters in comparison with the reported PSAR data.

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