Acta Scientific Applied Physics

Volume 2 Issue 8 August 2022

Review Article

Analysis of the Pump Trip Transient in VVER-1000

Hussein Gamal, Aly Shaaban, Mohamed Abdulhameed, Ashrakat Helal, Menna Mohamed, and Ayah Elshahat*

Nuclear and Radiation Engineering Department, Faculty of Engineering, Alexandria University, Egypt

***Corresponding Author:** Ayah Elshahat, Nuclear and Radiation Engineering Department, Faculty of Engineering, Alexandria University, Egypt.

Received: May 13, 2022

Published: July 29, 2022

Abstract

System codes have always been used to ensure the safety of nuclear reactors in both transient and accident conditions. This paper uses RELAP-SCDAPSIM to perform a safety analysis of VVER-1000 during a reactor coolant pump (RPC) trip in the absence of the Reactor Power Limitation Controller (RPLC). Based on the single-failure criterion, The malfunction of RPLC activates the Reactor Power Controller (RPC) causing scram. Thermal imbalance and flow reversal are the most characteristic consequences of RCP trip. The interplay of these two phenomena in the existence of scram is thoroughly examined, and the effect on reactor parameters is discussed. It has been found that a new steady state is established within 400 seconds of transient initiation, and all the acceptance criteria have been fulfilled. It can be concluded that although scram is not normally effected in response to an RCP trip, VVER-1000 can safely respond to this transient in the absence of the RPLC, and maintain a steady state without any operator intervention for at least three hours.

Keywords: VVER-1000; Pump Trip; Scram; RELAP-SCDAPSIM; Safety Analysis

Introduction

In nuclear reactors, off-normal events are categorized as: (a) transients, also termed anticipated operational occurrences, and (b) accidents. The trip of one or more reactor coolant pumps (RCP) belongs to the category of transients, and is also termed loos of coolant flow transient [1]. An RCP trip can occur due to one of the following reasons [2]:

- Mechanical failures,
- Disturbances in the control system, or
- Loss of electric power.

Reduction of the coolant flow leads to an imbalance between the heat generated in the core and the heat removed which might result in the core exceeding its thermal limits. Thermal imbalance also leads to a short-term pressure increase of both the primary and the secondary circuits [1]. That's why the RCP trip transient has the following three acceptance criteria [1]:

- The probability of DNB in the core is low.
- Pressure at both the primary and the secondary systems is maintained below 110% of the design pressure.
- There is no fuel meltdown anywhere in the core.

Input description and validation

The nodalization of the RELAP-SCDAPSIM model for VVER-1000 is shown in Figure 1. The pressure vessel contains 5 fuel channels (components 512, 513, 514, 515, and 516), and one bypass channel (component 508). Each fuel channel contains an SCDAP fuel rod heat structure and an SCDAP control rod heat structure. Fuel rod heat structures are modeled using 6 radial nodes. However, control rod heat structures are modeled using only 2 nodes due to an

SCDAP limitation. Channel 516 is the hot channel, where all the calculation of this paper has been performed.

Figure 1: VVER-1000 nodalization.

The primary circuit consists of the piping system, the pumps, and the steam generator (SG) tubes. The Westinghouse pump model, built in RELAP5, is used to model the pumps, and their complete coastdown takes 104 seconds based on data recorded at KNPP [3]. A pressurizer (component 526) with four groups of heaters is also included in the model. The pressurizer (PRZ)'s spray line (component 532) is connected to loop 4, whereas its surge line (component 525) is connected to loop 3. A set of relief valves are also connected to the PRZ. The heat input of the pressurizer heaters is based on measured data and is modeled using 4 trip-controlled general tables.

The feedwater system is modeled using time-dependent volumes (components: 131, 231, 331, and 431) providing a constant mass flow with a predefined coolant temperature through valves 181, 281, 381, and 481, and is treated in the analysis as a boundary condition.

The secondary side of the SGs, and the steam lines are modeled with minimum sufficient detail. The steam line includes the common header (component 450), turbine stop valve (component 468), and steam dump to atmosphere (BRU-A) valves. Steam dump to condenser (BRU-K) valves are not modeled because the condenser is treated as a boundary condition and is replaced by a time-dependent volume.

To include a sensitivity analysis method in the model, the number of nodes for some of the primary components of Loop 4 has been changed to examine nodalization sensitivity. For instance, the cold leg in Loop 4 comprises only two nodes, as opposed to the 4-node cold legs of the other three loops.

Validation results

When checking the modeled reactor against the real VVER-1000 power plant [4,5], it is found that the primary system model shows a satisfactory degree of accuracy. On the other hand, the secondary system is only approximate. It's a non-complete loop that is based on pre-defined boundary conditions that duplicate the reactor's steady-state. Consequently, while the steady-state parameter values of the secondary circuit might be reliable, its transient behavior shall be treated with care, and shall not be used to draw conclusions concerning the reactor's transient behavior without sound judgement.

The input deck has been validated by performing a steady-state simulation for 2000 seconds and comparing the results with measured VVER-1000 plant data found in Ref. [6]. The results of input validation are presented in tables 1-4. For the primary circuit (Tables 2 and 3), no relative error exceeds 3%. For the secondary circuit (Table 4), errors are a bit larger, but never exceed 10%. It can also be observed that the parameter values of Loop 4 are not significantly changed from those of the other three loops. Based on these results, it can be concluded that the input deck is accurately representative of VVER-1000.

Simulation procedure

A simulation of the RCP trip has been performed by RELAP-SCDAPSIM for 3 hours (10,800 seconds), and its results are compared to three benchmarks which present measured plant data during the trip of one RCP in a VVER-1000 [6], a VVER-440 [7], and a Siemens-KWU PWR [8].

Per-channel parameter	Calculated channel value					Plant value	Error
	512	513	514	515	516		
Inlet coolant temperature [∘ C]	296.9	295.5	295.9	296.4	297.0	-	-
Outlet coolant temperature [∘ C]	333.0	313.2	319.2	326.4	335.0	320 ± 3.5	1.7%
Coolant pressure [MPa]	15.85	15.85	15.85	15.85	15.85	-	-
Channel mass flow rate [kg/s]	1,971	4,351	3,235	4,070	3,693	-	-

Table 1: Per channel parameters.

Per-loop parameter	Calculated loop value				Plant value	Error
	Loop 1	Loop 2	Loop 3	Loop 4		
SG inlet temperature [∘ C]	323.8	323.8	323.8	323.8	318 ± 2	1.8%
SG outlet temperature [∘ C]	294.6	294.6	294.6	294.5	287 ± 2	2.6%
Hot leg temperature [∘ C]	323.8	323.8	323.8	323.8	- 287 ± 2	- 2.7%
Cold leg temperature [∘ C]	294.8	294.8	294.8	294.7	15.64	0.4%
SG pressure [MPa]	15.71	15.71	15.71	15.71	-	-
Hot leg mass flow rate [kg/s]	4,468	4,468	4,468	4,457	-	-
Cold leg mass flow rate [kg/s]	4,468	4,468	4,468	4,457		

Table 2: Per primary loop parameters.

Parameter	Calculated value	Plant value	Error
Reactor thermal power [MW]	3000	3000	0%
Pressurizer pressure [MPa]	15.72	15.65	0.4%
Pressurizer temperature [∘ C]	345.3	347 ± 1	0.5%
Coolant flow rate [kg/s]	17,861	17,610 ± 400	1.4%

Table 3: Parameters common to all primary loops.

Per-loop parameter	Calculated loop value				Reference value	Error
	Loop 1	Loop 2	Loop 3	Loop 4		
SG inlet temperature [∘ C]	209.7	209.7	209.7	209.7	220 ± 5	4.9%
SG outlet temperature [∘ C]	281.4	281.4	281.4	281.4	- 6.17 - 6.56	- 3.2%
SG pressure [MPa]	6.566	6.566	6.566	6.566	437 ± 30	8.4%
Main feedwater mass flow rate [kg/s]	400.2	400.2	400.2	399.5	437 ± 30	8.4%
SG outlet mass flow rate [kg/s]	400.2	400.2	400.2	399.5	2.4 ± 0.05	9.6%
SG water level [m]	2.170	2.170	2.170	2.170		

Table 4: Per secondary loop parameters.

This simulation is based on the following assumptions:

- Failure of the Reactor Power Limitation Controller (RPLC) to operate and respond to the thermal imbalance. The main function of the RPLC is to reduce the reactor power at transients without causing a scram. At base-load, the VVER-1000 operates in the T-mode (i.e. secondary circuit pressure stabilization). When the pump is tripped, a sudden pressure/temperature transient occurs, and the RPLC responds by switching to the N-mode (i.e. neutron power stabilization) [6]. A number of gray control rods is partially inserted into the core to reduce the reactor power and stabilize it at a lower level without causing a economically expensive scram [7,8]. When the RPLC fails to operate, the Reactor Power Controller (RPC) effects a scram and black control rods are fully inserted into the core [6]. This assumption is based on the single failure criterion, which is defined as follows: Given two or more systems performing the same safety function, when one of these systems fails, the remaining one(s) are capable of performing the required safety function [9]. From a calculational point of view, when scram was not modeled to occur just after the pump trip, RELAP5-SCDAP reported a termination by failure a few tens of seconds after the pump trip had been initiated, which is indicative of the onset of core meltdowm.
- There is no operator intervention throughout the whole simulation.
- Only the behavior of the hot channel (HC), and bypass channel (BC) is considered. Examination of the behavior of the latter is based on an analysis recommendation given by the IAEA in Ref. [1].
- The behavior of Loops 2 and 3 is assumed identical, as is justified perviously, and only one of them is plotted in the figures of Section 4.
- Due to the approximations used in modeling the secondary circuit, the simulation is restricted to the primary circuit. That is, a loop simulation is performed rather than a system simulation.

Results and Discussion

The results of the simulation are presented if Figures 2 13. The event sequence in the primary circuit proceeds as follows [2,6,8,10,11]:

- The RCP of Loop 1 is tripped 100 s after the beginning of the simulation, and scram is effected 1.4 s after the pump trip due to the RPC action, as explained in Assumption 1. The core power quickly stabilizes at around 3.3% of nominal power (Figure 2).
- The pump velocity decreases quickly (Figure 3) causing a corresponding flow reduction (Figure 4). The flow reduction has the following effects:
- A sudden pressure surge for a few seconds (Figure 5), which opens the PRZ relief valves very briefly (Figure 6) causing a rapid system depressurization and a decrease in coolant inventory. At the sudden pressure surge, the primary pressure never exceeds 110% of its design value, and Acceptance Criterion 2 is fulfilled.
- Flow reversal in the affected loop (Figures 4 and 7), which results in a decrease in pressure drop along the core (Figure 8), and a flow increase in the intact loops.
- Sudden increase in coolant temperature (Figures 9 and 10), which consequently leads to a sudden decrease in coolant density (Figure 11).
- Sudden increase in clad temperature (Figure 12) due to the decrease in heat removal. However, the maximum excess temperature doesn't exceed 10 ∘ C, which means a sufficient margin to boiling exists, and Acceptance Criterion 1 is fulfilled.
- Due to the pressure reduction and the temperature increase just described, the PRZ heaters are automatically activated (Figure 13), in an attempt to increase the primary pressure to near its nominal value. Although the pressure transient dies out 400 seconds after the trasient onset, the sequential turning on and off of the PRZ heaters causes the primary pressure to oscillate until it nearly stabilizes at around 5000 seconds (Figure 5). It can be observed that the primary pressure and the PRZ follow exactly the same trend.
- Upon stabilization of both reactor power and pressure, and establishment of flow reversal (which causes coolant temperature decrease due to mixing hot and cold coolant), all core temperatures decrease after their brief surge and then smoothly stabilize to new lower values (Figure 12). That is, a new steady state is established.

When these figures were compared with the plant measured data found in the benchmerks [6-8], they have shown good agreement. This verifies the correctness of both the simulation results and their interpretation. It can be concluded that in the case of one RCP trip in the absence of operator intervention, VVER-1000 is capable of reaching a safe steady-state without compromising fuel integrity, or reaching the DNB set-point.

Figure 2: Core thermal power.

Figure 3: Pump velocity.

Figure 4: Pump outlet (cold leg) mass flow rate.

Figure 5: Core pressure.

Figure 6: PRZ relief valves.

Figure 7: Hot leg mass flow rate.

Figure 8: Core pressure drops.

Figure 9: Hot leg temperature.

Figure 10: Cold leg temperature.

Figure 11: Core coolant density.

Figure 12: Core temperatures.

Figure 13: PRZ temperature which is indicative of the behavior of PRZ heaters.

Conclusion

A new steady state is established within 400 seconds since the initiation of the RCP trip, and all the RCP trip acceptance criteria have been fulfilled. The failure of the RPLC causes a scram in response to the RCP trip. However, VVER-1000 can safely respond to this transient in the absence of the RPLC, and maintain a steady state without any operator intervention for at least three hours.

Bibliography

1. International Atomic Energy Agency. "Accident Analysis for Nuclear Power Plants with Pressurized Water Reactors". *Safety Reports* 30 (2003): 65.
2. Y Rahmani. "Safety analysis of vver-1000 type reactors against pump failure accident in 2 and 4 loops". The 7th International Scientific and Technical Conference "Safety Assurance of NPP with VVER" (2011).
3. Innovative Systems Software, The KNPP VVER-1000 Sample Model.
4. ROSATOM. "The VVER Today". *Technical Report*.
5. International Atomic Energy Agency. "Status report 107: VVER-1200 (V-392M)". 1200 (2011).
6. P Grudev and M Pavlova. "Simulation of loss-of-flow transient in a VVER-1000 nuclear power plant with RELAP5/MOD3.2". *Progress in Nuclear Energy* 45.1 (2004): 1-10.
7. P Groudev and A Stefanova. "Validation of RELAP5/MOD3.2 model on trip off one main coolant pump for VVER 440/V230". *Nuclear Engineering and Design* 236.12 (2006): 1275-1281.
8. MF Lozano., *et al*. "Assessment of a Reactor Coolant Pump Trip for TRILLO NPP with RELAP5/MOD3.2". U. S. Nuclear Energy Commission, Technical Report (2000).
9. International Atomic Energy Agency. "IAEA Safety Glossary: 2018 Edition" (2019).
10. P Groudev and M Pavlova. "Total Loss of Feed Water for VVER 1000". Fifth International Information Exchange Forum (2000).
11. J Kitto and S Shultz. "Steam: Its Generation and Use, 41st ed". Babcock and Wilcox Company (2005).