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SAFETY ANALYSIS DURING LOW-PRESSURE CASE IN VVER-1000 WITH ASYST

ABSTRACT

This study evaluates the reactor degradation progression in VVER-1000 due toa low pressure case as a result of loss of coolant accident (LOCA) and total station blackout (SBO). The simulation evaluation was carried out using the adaptive SYStem Thermal-hydraulics (ASYST) program. The low-pressure scenario was simulated by modelling 80 mm small break LOCA (SBLOCA) in the hot pressurizer leg during SBO. This double ended break size was selected to cause a significate faster depressurization during low-pressure simulation and a longer borated cold-water injection from the passive hydro accumulators (HAs). This enabled evaluating the times required to attain critical set points during the transient progression. The investigation looked into loss of coolant circulation, fuel and clad heating up, commencement of hydrogen generation, the activation of the passive safety system, changes in pressure, and primary circulation. Simulation results showed that the modelled break area was enough to allow primary loop depressurization. The results revealed that the fuel damage decreases after the introduction of HAs. Actuation of HAs at their actuation set-points provided core cooling by injecting water into reactor core. These kinds of analyses assist in estimating the time available to perform operator safety actions. This in turn aids in emergency planning and severe accident management.

I. INTRODUCTION

Severe accidents (SA) are those in which the reactor core sustains significant damage. Some of the factors that contribute to SA include station blackouts (SBO) and coolant loss accidents (LOCA) [1]. SBO accident accounts for 26% of the total core damage frequency (CDF). To limit radioactive release in such accidents, severe accident mitigation guidelines and emergency response are required [2]. The primary goal of this study was to evaluate thermal hydraulic degradation progression, primary pressure fluctuations, and hydrogen (H_2) generation in VVER-1000 during total SBO together with SBLOCA postulated case. The Adaptive SYStem Thermal-hydraulics (ASYST) application was used to perform calculations for both steady state and transient scenarios.

II. A BRIEF DESCRIPTION OF VVER-1000

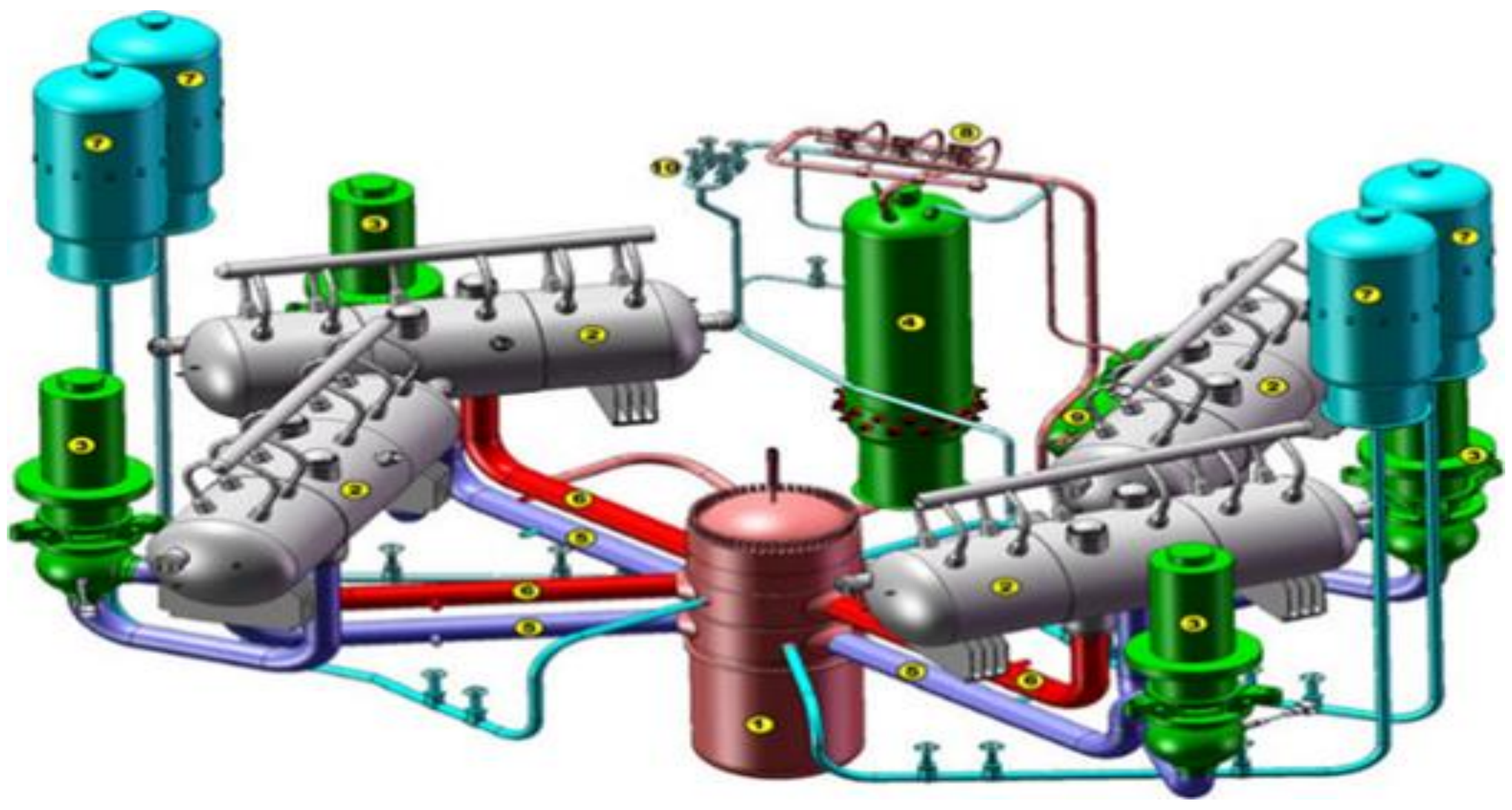


Figure 1. Primary circuit of VVER-1000 plant; (1) RPV, (2) SG, (3) RCP, (4) PRZ, (5) Cold leg, (6) Hot leg, (7) Accumulator, (8) PRZ pulse safety device valve, (9) Relief tank, (10) Injection system [3]

III. MODELLING AND TRANSIENT SIMULATION

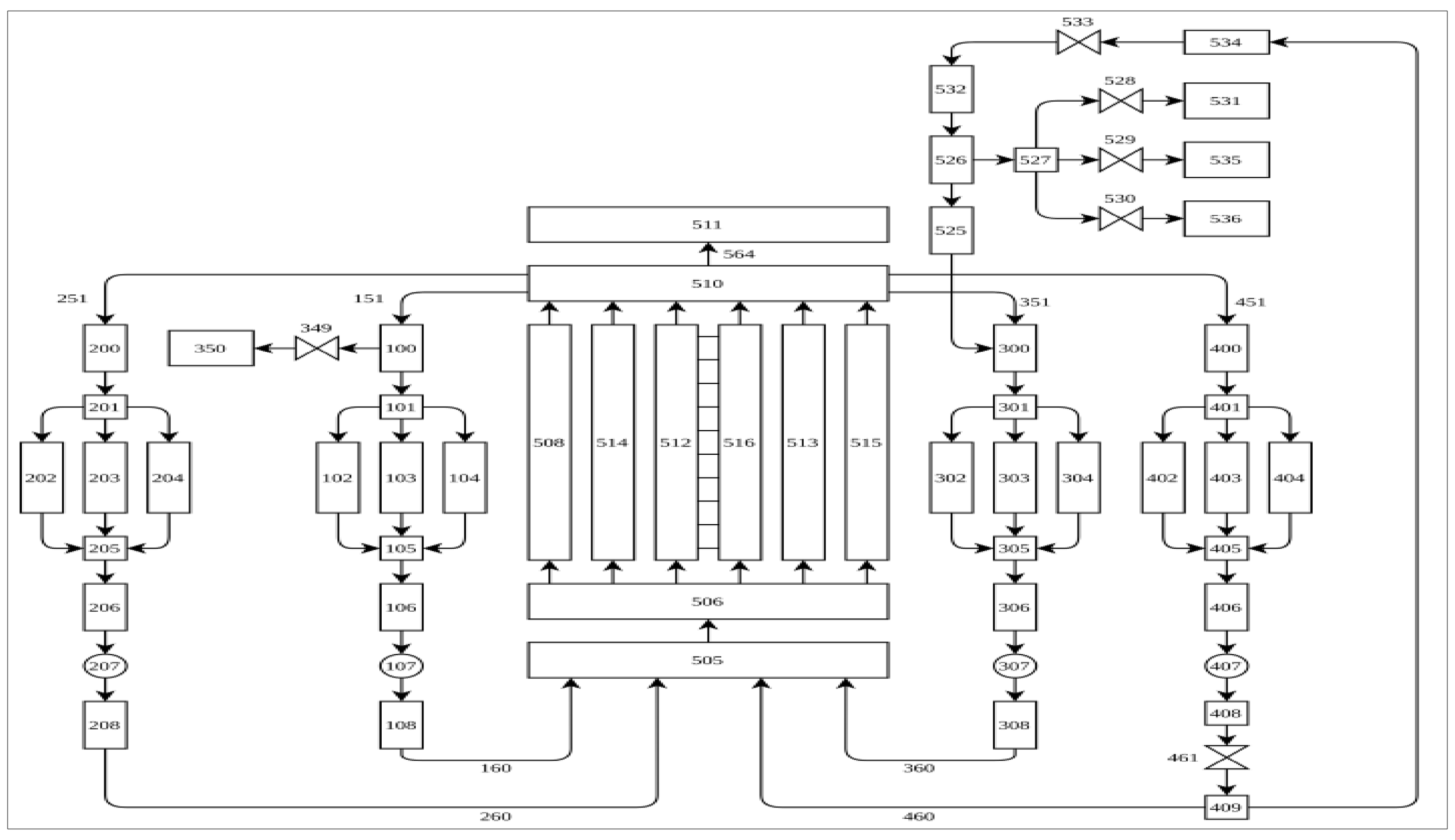


Figure 2. VVER-1000 nodalization diagram

The ASYST input was divided into four categories: hydrodynamics, heat structures, control systems, and neutronics

III. STEADY STATE CALCULATION

During the simulation, the cladding temperature in the upper node of core region attained the set point of 923 K.

This node measurement for core exit temperature in the NPP are usually located at the upper part of the fuel assemblies in the reactor core.

The simulation of stationary calculation in this work lasted 100 seconds (s). The results were equal or close when compared to the design specifications of the VVER-1000 design data [4 , 5]

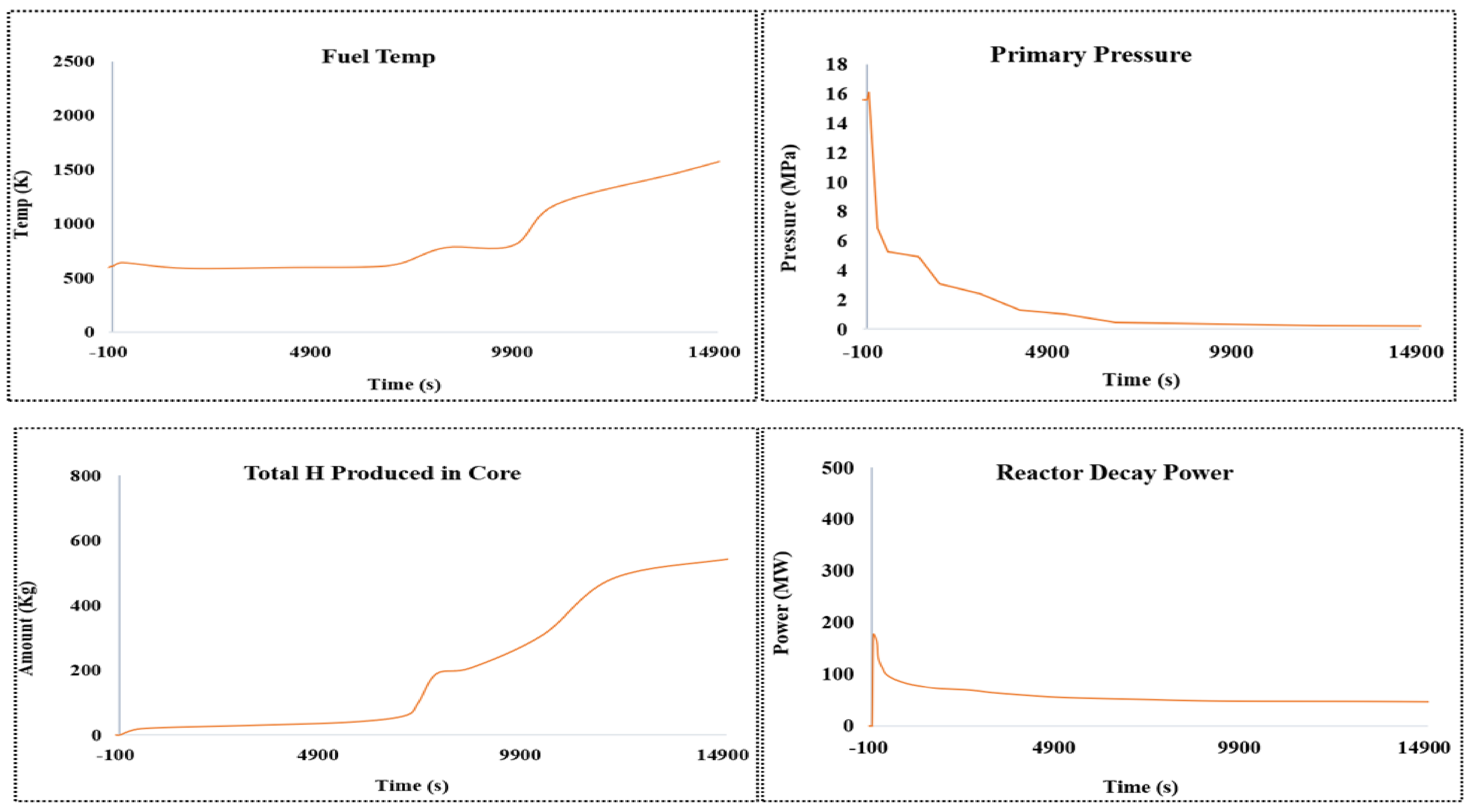
Table 1: Nodalization qualification at steady state level

	Design value	Steady State Value	Error	Acceptable Criteria
Reactor Power (MWth)	3000	3000	-	2.0%
RC Pressure (MPa)	15.7	15.7	0.06	0.1%
SG Pressure (MPa)	6.8	6.9	0.16	2.0%
Hot Leg Temp (K)	594.15	595	0.14	0.5%
Col Leg Temp (K)	564.15	565	0.15	0.5%
Core inlate flow rate (kg/s)	16,400	16,387	0.08	2.0%
Coolant level PRZ (m)	8.17	8.20	0.37	0.5%
SG level (m)	2.40	2.399	0.0416	0.2%

V. TRANSIENT RESULTS

Table 2: Sequence of main events (s)

Events	SBO with SBLOCA case with ASYST	SBO with SBLOCA case SBO with ASTEC
Transient Initiation	0.0	0.0
RCPs trip	0.0	0.0
SCRAM signal	1.6	1.6
Turbine trip	2.0	2.0
Feed water stop	5.1	5.0
SG-1 safety valve open	93.0	87.5
SG dry out	N/A	N/A
PRZ safety valves open	N/A	N/A
Loss of natural circulation	1302	1221
Core coolant exit Temp reaches 650 °C	13964	5776
HA-1 and 2 actuate	1484	1451
HA-3 and 4 actuate	1490	1464
End of calculation	15000	-



V. CONCLUSION

The 80 mm break size was big enough for fast depressurization of the primary loops. It allowed faster core degradation.

The actuation of the HAs delayed the reactor malfunctions. The loss of primary coolant circulation and core heat up were observed.

During the simulation time, it was observed that the reactor inherent control systems decreased the primary pressure to aid core reliable cooling. HAs were effective in delaying the core uncovering and the total dry out of the RPV.

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