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Steam Line Break investigation at full power reactor for VVER-1000/V320

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HIGHLIGHTS

- In this study we investigated Steam Line Break accident at full power reactor.
- The reference power plant for the analyses is Unit 6 at Kozloduy NPP.
- The RELAP/MOD 3.2 computer code is used in performing the analyses.
- The results are used for analytical validation of EOP.

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ABSTRACT

This paper presents the results of thermal-hydraulic calculation of "Steam Line Break" analysis at full power reactor for VVER-1000/V320 units at Kozloduy Nuclear Power Plant (KNPP), done during the development of symptom based emergency operating procedures (SB EOPs) for this plant. The RELAP5/MOD 3.2 computer code has been used in performing the analyses in a VVER-1000 Nuclear Power Plant (NPP) model. A model of VVER-1000 based on Unit 6 of Kozloduy NPP has been developed for the systems thermal-hydraulics code RELAP5/MOD 3.2 at the Institute for Nuclear Research and Nuclear Energy–Bulgarian Academy of Sciences (INRNE–BAS), Sofia.

The main purpose of the analysis is to estimate the parameters of the monitored plant which are used to identify symptoms that are used by operators to identify the plant's state and the critical safety function (CSF). The results of the thermal-hydraulic analyses have been used to assist KNPP specialists in analytical validation of EOPs.

The performed analysis is based on a previously used bounding approach in analytical validation of SB EOPs. Based on this approach a list of scenarios has been performed, involving a different number of safety systems with or without operator actions. The presented thermal-hydraulic calculations of the accident scenarios involve the loss of CSF "Subcriticality" for VVER-1000/V320 units at KNPP.

1. Introduction

Emergency operating procedures (EOPs) analyses are designed to provide the response of monitored plant parameters for the identification of operators' symptoms available, timing of the loss of critical safety functions and timing of operator actions to avoid the loss of critical safety functions or core damage. The objective of analytical validation is to perform an evaluation of the EOPs in order to confirm written correctness of the procedure, and to ensure that technical and human factor concerns have been properly incorporated. The methodology, which was used in developing the symptom based emergency operating procedures (SB EOPs) for KNPP VVER-1000/V320 is an elaboration of Beelman (1999).

During the development of SB EOPs at Kozloduy Nuclear Power Plant (KNPP), a numbers of thermal-hydraulic analyses for KNPP have been performed at the Institute for Nuclear Research and

Abbreviations: AFWS, auxiliary feed water system; BRU-A, steam dump to atmosphere (SDTAF); BRU-K, steam dump to condenser; CSF, critical safety function; CV, check valve; ECCS, emergency core cooling system; EFW, emergency feed water; EFWS, emergency feed water system; EFWP, emergency feed water pump; EOP, emergency operating procedures; FWL, feed water line; FAIV, fast acting isolation valve; HA, hydro accumulator; HPSIS, high-pressure safety injection system; HPP, high pressure pump; HHPP, high high pressure pump; INRNE–BAS, Institute for Nuclear Research and Nuclear Energy of Bulgarian Academy of Sciences (Sofia, Bulgaria); KNPP, Kozloduy Nuclear Power Plant; LPP, low pressure pump; LPSIS, low-pressure safety injection systems; MCP, main coolant pump; MFWS, main feed water system; MSH, main steam header; NPP, Nuclear Power Plant; PRZ, pressurizer; RCS, reactor coolant system; SB EOPs, symptom based emergency operating procedures; SG, steam generator; SLB, Steam Line Break; SV, safety valve; VVER, water water energy reactor.

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Nuclear Energy–Bulgarian Academy of Sciences (INRNE–BAS) using RELAP5/MOD3.2 computer code (Andreeva et al., 2012; Groudev et al., 2008,2013; Pavlova et al., 2007). The scenarios, which have been developed by plant specialist at KNPP, contain failures of equipment. The purpose of the scenarios is to predict the behavior of NPP and to help correctly validate the operator action for validation and verification of EOPs.

EOPs thermal hydraulic analyses are performed for accident scenarios which involve the loss of critical safety functions (usually evaluate the accidents beyond the automatic capabilities of the engineered safety features where operator intervention is required). When performing the task to identify the scope of coverage of the EOPs, a good knowledge of the thermal-hydraulics of the plant (Groudev et al., 1999a,b) is necessary to identify the possible challenging accidents. The general philosophy of EOP analyses is described in detail in (Pavlova et al., 2008; Groudev et al., 2008).

2. Description of the Kozloduy NPP and RELAP5 model

The reference power plant for this analysis is Unit 6 at Kozloduy NPP site. This plant is a typical VVER-1000 Model V320 pressurized water reactor (Groudev et al., 1999a). The basic design of a VVER-1000 plant comprises: a pressurized water reactor of 3000 MW thermal power with 163 hexagonal fuel assemblies in the core, and 10 absorbing rod banks, located in 61 fuel assemblies; four primary loops and one turbogenerator producing 1000 MW of electric power. The reactor vessel has four inlet nozzles of Ø 850 mm and four outlet nozzles of Ø 850 mm to connect to the primary loops. There are also four inlets of Ø 280 mm for safety injection of boron solution to the upper and lower plena in case of primary loss of coolant. Each loop includes one main circulation pump and a horizontal U-tube steam generator (SG). The behavior of the horizontal SG is very different compared to Western-style vertical SG (Groudev et al., 1999a). For example, the secondary side of the horizontal SG contains much more water and the response of loss-of-feedwater transients is slower. Steam Generators play a very important role in the safe and reliable operation of VVER power plants. They determine the thermal-hydraulic response of the primary coolant system during operational and accident transients. There are three different feedwater systems on secondary side: main feed water system (MFWS) with two turbine-driven pumps; auxiliary feed water system (AFWS) for normal start up, shutdown and cooldown; emergency feed water system (EFWS) with three trains or 3 × 100% redundancy important for this analysis. The emergency core cooling system (ECCS) consists of high-pressure safety injection system (HPSIS), low-pressure safety injection systems (LPSIS) with 3 × 100% redundancy and four hydro-accumulators. All elements of the primary circuit are situated in a steel-lined, cylindrical, prestressed concrete containment vessel. Systems and equipment of KNPP, Unit 6 operate according to the design requirements for corresponding level of the reactor power (Groudev et al., 1999a).

RELAP5/MOD3.2 computer code model has been used to simulate the VVER-1000/V320 NPP model (Groudev et al., 1999b). The model has been developed at INRNE–BAS for analyses of operational occurrences, abnormal events, and design basis scenarios. The RELAP5 nodalization schemes of the plant used in the analysis are presented in Figs. 1–3. The actual four-loop system has been modelled by four single loops for primary and secondary sides. The model provides a significant analytical capability for the specialists working in the field of NPP safety. In the RELAP5 model for VVER-1000/V320 NPP are included: reactor vessel; core region represented by three channels; pressurizer system including heaters, spray and relief valves; safety system–low pressure injection pumps. In the model also is presented a make up/drain system including connection (control) with pressurizer. Secondary side is developed too and represented by eight SG safety valves, four BRU-A valves, BRU-K valves, steam pipe lines (including main steam header) and turbine including regulating valve in front of the turbine. The horizontal steam generator (SG) has been modeled. A separator model and the perforated sheet have been modeled in SG model, too. Main cooling pump (MCP) has been developed using homologous curves of real pumps.

3. Event description

The consequences of a Steam Line Break (SLB) considerably depend upon several system parameters: initial power level; location of break; size of break; safety systems that are operational; control systems that are operational; and possible other failures that could occur.

For a small SLB the secondary system would indicate an increase in load with a resulting decrease in primary system average temperature and pressure. Due to the apparent increased load, the steam flow from the faulted steam generator would be increasing. Due to the increased steam flow, the feedwater control valves would modulate to a more open position in an attempt to maintain steam generator water level. As a result, the main feed flow in faulted SG would be increased. Another indication of this type of break would be a decreasing water level in the condenser hot well.

The least likely and most severe of the postulated loss of secondary coolant events is the total break size ID 580 mm upstream of the fast acting isolation valve (FAIV). For large secondary break, an immediate decrease in pressure in faulted steam line occurs. The low steam line pressure setpoints ($P_{SG} < 4.9$ MPa and $\Delta t_{S(I-II)} > 75\,^{\circ}$C) are reached in approximately 5–10 s, which results in reactor scram. This yields a turbine trip. In coincidence with rapidly decreasing steam line pressure, the primary experiences a decreasing average coolant temperature and decreasing primary pressure. The primary system transient follows the reactor trip. The same signals, $P_{SG} < 4.9$ MPa and $\Delta t_{S(I-II)} > 75\,^{\circ}$C, actuate the safety injection systems. The important system parameter trends for this break are an uncontrolled pressure decrease in faulted SG and this SG is completely depressurized. The other symptoms include decreasing of steam generator water level of faulted SG and initially decreasing of primary pressure and temperature. A rapid, extensive primary system cooldown occurs. As the primary system temperature drops, the heat transfer to the steam generator (faulted SG) and the primary system cooldown rate will be reduced. This trend will continue to the point where the primary system water volume shrinkage (caused by the cooldown) is overcome by the make up system flow rate. This results in the primary system pressure and pressurizer level restoration. Depending on the initial conditions of the systems and the size of the break, one of two conditions will be reached on the blowdown. The first is when the steam generator blowdown is essentially completed and further cooldown of the primary system is controlled by the auxiliary feedwater flow. The second is when the primary temperature is reduced so far that the heat transfer to the secondary side matches the heat generation in the primary system, which results in a stabilized primary temperature. The primary system transient following reactor trip and safety injection initiation is dependent upon the initial power level prior to transient initiation. For a SLB from full initial power, the primary average temperature and pressure will initially decrease below no load temperature, after which decay heat generated in the core will immediately begin restoring primary temperature and pressure.

4. Initial and boundary conditions

The initial and boundary conditions of important plant parameters and systems are as follow: