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Analysis for Emergency Operating Procedures of Bohunice V-1 NPP

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1. Introduction

Significant attention is paid to safe operation of Bohunice V-1 NPP, equipped with two VVER-440/230 units of older concept. During last decade a complex reconstruction of V-1 NPP was performed. Original design of the plant was upgraded essentially. Besides number of other measures, emergency core cooling system (ECCS) was reconstructed completely: original high-pressure injection system was separated from purification system, discharge lines were re-routed and new powerful low pressure ECCS, having relatively high cut-off head (an thus partly replacing accumulators), was added. Consequently, maximum design basis accident was increased significantly. On the secondary side, new steam dumps to atmosphere were added on each of six steamlines upstream the isolation valves. The reconstruction was completed by the preparation and approval of new safety analysis report and actualisation of PSA level 1 study. The results of this study clearly demonstrate significant achievements in the field of safety enhancement. Preparation of PSA level 2 will start in few months. Another important step in the nuclear safety upgrading of V1 NPP is the development of new symptom based emergency operating procedures (EOPs), which started in 2001. In this contribution, current status of work on the preparation of EOPs is described. Attention is paid especially to analytical support for EOPs – development of RELAP5 input data deck and example of its on-transient level validation. Finally, a concrete example of support calculation for the development of EOPs strategies – analysis of feed & bleed scenario – is presented.

2. Development of EOPs

Similarly to other NPPs operated in Slovakia (Bohunice V2 and Mochovce NPP) equipped with newer VVER-440/V213 reactors, Westinghouse approach was adopted for V-1 EOPs, too. Since the EOPs development for VVER-440/V213 units started several years earlier, experience gained on these units was used in the development of V-1 EOPs. The same experts from Bohunice NPP, which took recently part on the EOPs development for V-2 NPP, are today involved in the development of EOPs for V-1 NPP.

At first, major design differences between V-1 and V-2, which could have an impact on EOPs development, were identified. The impact of V-1 design features on EOPs strategies and procedures was analysed. All available accident analysis and engineering studies were summarised. Based on this information, the estimated strategies for particular procedures were specified. Since these strategies have to be further verified, a list of required analysis for the next phase of EOPs development program was prepared. Technical support organisations were subcontracted by utility in order to perform these accident analyses.

3. Preparation of REALP5 model

For this purpose of support calculation for EOPs development, new six-loop model based on RELAP5/Mod3.2 code was developed, verified and validated. The main goal was to develop the plant model applicable for simulation of broad spectrum of accidents, expected response of the reactor protection, control systems and anticipated operator interventions. Therefore, besides detailed nodalization of the primary and secondary systems, significant attention was paid also to the modelling of core kinetics, ECCS, auxiliary systems, reactor protection, ESFAS and plant controllers (make-up and let-down, PRZ heaters and spray, FW controller, reactor power controller, steam dumps to condenser and atmosphere, etc.). Final nodalization is shown on Fig. 1. Standard methods and criteria were used to check the geometrical fidelity and the model performances in steady-state conditions. Transients recorded during start-up experiments, performed after completing the plant reconstruction, were used for on-transient level validation and tuning of the developed model. An example of on-transient level validation is shown on Figs. 2.1-2.4.

One of the main particularities of the high pressure ECCS is the throttling capability of the flow through the usage of the motor operated valves in the discharge of each high-pressure pump, which are working simultaneously with minimum flow protection. This way injection capacity into primary system may be smoothly controlled by operator, which could be effectively used for primary pressure control, e.g. for the treatment of primary-to-secondary leakages accidents. This relatively complicated injection system was modelled separately (Fig. 3), taking into account complete geometry of discharge lines and operation of the control valves. Obtained characteristics were compared with available measured values (Fig.4) and incorporated in simplified form into the basic plant model, which was later used for the analysis of broad spectrum of accidents. Here, an example is presented.

4. Support analysis for Feed & Bleed procedures

The complete loss of FW leads, after depletion of SG inventory, to loss of secondary heat sink and necessity to initiate primary feed & bleed procedure as an alternative way of core decay heat removal. There are several reasons why the time margin for initiation of feed & bleed procedure for VVER-440 reactors of both generations is relatively long. The most important are as follows:

- low core power density;
- large water inventories on both, primary and secondary sides;
- cut off head of HP injection is comparable with setpoint for opening of PRZ relieve valve

Therefore, contrary to some higher powered reactors with large margin between HP injection cut-off head and operating pressure, for VVER-440 reactors there is no strong “deadline” for initiation of feed & bleed procedure, behind which it is not possible to depressurise primary system fast enough in order to achieve the HP injection before core damage occurs. Based on the analysis of VVER-440/V213 units, it is possible to initiate the feed & bleed procedure even for states approaching to saturated conditions in primary system. However, in such cases significant voiding in hot legs resulting in flow stagnation in loops can be expected during depressurization of the primary system. Consequently, HP injection into cold legs will result in formation of cold plumes in downcomer and reactor integrity could be potentially endangered.

After loss of FW, the decay heat is removed from the core to SGs either in forced or (after RCPs trip) natural circulation. The steam generated on the secondary side of SGs is removed either through steam dumps to condenser or to atmosphere, what causes continuous decreasing of secondary side water inventory. When the secondary side heat sink is going to be depleted, the decay heat results in increasing of primary temperature and pressure. As soon as opening setpoint of the PRZ relieve valve is reached, primary inventory starts to discharge into bubbler tank and later, after break of the membrane, into confinement.

Characteristic feature of the scenarios with loss of FW is continuous water level decreasing on the secondary side of SGs, resulting in decreasing of flooded heat transfer area between primary and secondary system. In typical nodalization of horizontal SG, the total number of heat exchange tubes is split into several layers. Performance of such kind of nodalization in conditions with secondary side water level decreasing was studied in [1], [2]. From these results it follows, that instead of smooth reduction of flooded heat transfer area between the primary and secondary system there is stepwise reduction with the number of "steps" equal to the number of layers of heat exchange tubes, used in nodalization. In order to describe properly the course of primary temperature during this process, special refined nodalization of SG tubing consisting of 26 layers.

Several different variants of loss of FW with application of feed & bleed strategy was analysed up to now for the development of new EOPs. These analyses were focused mainly on the finding the criteria for the initiation of the feed & bleed procedure. In near future it is intended to analyse also the cases with restoration of heat sink after initiation of feed & bleed procedure or the possibility to use low-pressure injection (cut-off head of LP pumps is 3.3 MPa) in the case of unavailability of HP injection. Here, only one example is presented. Main features of this case are as follows:

- primary temperature is maintained as long as possible via steam dump to condenser (later via steam dump to atmosphere) in order to maximise the time margin to membrane break of bubbler tank (e.g. in order to minimise plant damage for the cases when secondary heat sink is restored);
- reaching of subcooling margin 20 °C on the core exit was considered as a criterion for initiation of feed & bleed procedure;
- minimum feed (1 of 4 HP pumps) and bleed (PRZ relieve valve only) configuration was considered.

It was assumed that before the initiating event (total loss of FW) the plant was operated on nominal parameters. The decay heat after reactor scram was conservatively modelled using ANS-73 curve. Otherwise, the best estimate approach was applied. First operator interventions were assumed 10 minutes after initiating event. The further details of the scenario are as follows:

$t_0 = 0$ s - total loss of FW => pressure drops in FW header below 5 MPa => trip of both TG => coast down of the 4 RCPs supplied directly from TG generator => reactor power controller decrease the power to 23% of nominal value

$t_1 =$ ... - first order scram (signal “drop of Δp or drop of RCP speed on 3 or more RCPs”)

$t_2 = 220$ s - automatic back-up power supply for 4 RCPs

$t_3 = ...$ - trip of all RCPs, blockade of steam dumps to condenser

$L_{SG} < -500$ mm - operator is keeping primary temperature using 2 steam dumps to atmosphere on main steam header

- primary pressure is controlled by PRZ heaters

$t_4 = ...$ - loss of ability to keep primary temperature (both steam dumps into atmosphere fully open and $L_{SG} <$ ...mm)

$t_5 = ...$ - initiation of feed & bleed (start of 1 HP pump, 30 s later opening of PRZ

$T_{hot\_leg} = T_{sat} - 20°C$ relieve valve)

$t_6 = ...$

$T_{hot\_leg} = T_{sat} - 50°C$ - end of calculation

The results obtained are illustrated on Figs. 5.1-5.8. After reactor trip, the decay heat is at first removed from the core to SGs in forced convention regime. Later, when the water level in SGs drops below - 500 mm (from nominal level), RCPs are tripped by operator and natural circulation is set in primary system. After blocking of steam dumps to condenser (high water level in the condenser) the operator is using the steam dumps to atmosphere in manual regime for keeping the primary temperature. The continuous decrease of flooded heat transfer area is compensated as long as possible by increased opening of steam dumps to atmosphere. When the upper rows of SG tubes become uncovered, the steam generated in lower part of SGs is superheated on the upper rows, what further increases the effectiveness of the exploitation of secondary heat sink. After depleting the SGs water inventory, the primary temperature and pressure start to rise. About 5.5 hours after initiating event the set point for PRZ relieve valve opening is reached and since this time till the initiation of feed & bleed procedure is the valve cycling. At first, steam is discharged through the relieve valve into bubbler tank and the PRZ water level is rising. Later, the PRZ is completely filled and saturated water is discharged. When the subcooling margin on core exit drops to 20 °C, feed & bleed procedure is initiated. After full opening of PRZ relieve valve the primary pressure drops to saturation value, what causes flashing in hot regions of primary system. Steam bubble appears in upper plenum and upper part of SG tubing. However, there is no core uncovery and the core cooling is ensured reliably during the whole accident. Furthermore, the natural circulation in loops is still preserved and cold water injected from HP pump is mixed with ambient coolant. About half an hour after initiation of the feed & bleed procedure, HP injection exceeds the coolant losses through PRZ relieve valve and the mass inventory in the primary system starts to rise. The analysis was terminated when the subcooling margin on core exit reached 50 °C.

From the results it follows, that despite late (about 8 hours after the initiating event) initiation of the feed & bleed procedure, resulting in drying of SGs and low margin (20 °C)to saturated condition on core exit, and despite minimum feed (1HP pump) and bleed (PRZ relieve valve only) configuration, the procedure was successful. Since the natural circulation was not interrupted during depressurization of the primary system, the danger of PTS with potential impact on the RPV integrity was minimised.

Abbreviations

ECCS	emergency core cooling system
EOPs	emergency operating procedures
ESFAS	emergency safety feature actuation system
FW	feedwater
HP	high-pressure
PRZ	pressurizer
PTS	pressurized thermal shock
RCP	reactor coolant pump
RPV	reactor pressure vessel
SG	steam generator

Symbols

L	water level
p	pressure
t	time
T	temperature

References

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[2] P. Matejovic, L. Vranka, M. Bachraty: Support analysis for emergency operating procedures of VVER-440/V213 units, Fifth International Seminar on Horizontal Steam Generators, Lappeenranta, Finland, 2001