IMPLEMENTATION OF THE IN-VESSSEL RETENTION STRATEGY
FOR RIVNE NPP UNITS 1,2

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Abstract

According to the Comprehensive Safety Improvement Program for Ukrainian NPPs one of the measures to be implemented is an “In-vessel retention strategy for WWER-440 (Rivne NPP Units 1 and 2)”, which has been included in the Program as one of the so-called ‘post-Fukushima’ safety upgrades based on results of the ‘stress tests’ conducted for all operating Ukrainian reactors in 2011-2012. The paper discusses the results of the first analytical studies conducted to support implementation of the strategy at Rivne NPP (RNPP). As a first step a screening of existing similar technical measures at European WWER-440 plants (Loviisa and Paks, etc) and analysis of applicability were done. Using the same calculation strategy as Paks NPP, calculations were done for specific conditions and geometry of RNPP Units 1, 2. Reactor core melting and corium relocation to the bottom was simulated with MELCOR. Flooding of gap between the reactor vessel and cavity, efficiency of external cooling and circulation of water in cooling loop were calculated with AHLET. Based on calculation results the recommendations for preliminary design of ex-vessel cooling system were proposed.

1. INTRODUCTION

Ukrainian utility, NNEGС “EnergoAtom”, which operates all 15 WWER Units, is obligated to implement the Comprehensive Safety Improvement Program for Ukrainian NPPs, – which is a set of technical measures strengthening safety of operational Units. One of the key safety measures, which have increased significance after Fukushima accident, is “in-vessel retention” of corium in case of severe accident. Integrity of reactor vessel gives a chance to significant reduction of radiological accident consequences. Other major aspects are a reduction of hydrogen generation and lower pressure in the containment. Thereby “in-vessel retention” (IVR) measure is co-jointed with two other measures: installation of passive autocatalytic recombiners (PARs) and containment filtered venting.

IVR measure has been realized on most of WWER-440 reactors in Europe. Large amount of borated water in Bubble Condenser Tower together with sufficient elevation difference allows to deliver passive system with high reliability and performance.

Analysis of IVR strategy implementation on WWER in EU countries shows that general approach was almost the same for all units: Loviisa, Paks, etc. And the most challenging problem was the sufficiency of gap between reactor vessel and concrete cavity to provide flow rate and heat sink for reactor cooling. Thus the main objective of investigation was defined as follows, to check the applicability of common approach to the real geometry of Rivne Units 1, 2 and to recommend a preliminary design of ex-vessel cooling system.

2. OTHERVIEW OF EXISTING IVR STRATEGIES AND SELECTION OF PREFERABLE DESIGN

An otherview of ex-vessel cooling system, implemented on WWER-440 in EU countries has shown two different technical designs:
— Fist one is implemented in Loviisa NPP [1], [2], and;
Second one was realized on several NPPs in Eastern Europe. Paks NPP, could be taken as a reference Unit [3].

The principal difference between these variants is a design of the biological shield below the reactor bottom. In first case, the shield is moved down by pneumatic driver to ensure enough free space for sufficient cooling. Alternative idea is to save the initial design of the shield and to install float passive valve, which would opened by itself after flooding of reactor cavity. In this case, the coolant will circulate in the gap between reactor vessel and existing thermal shield. Second case is much easy, cheaper and do not lead to additional significant exposure of staff. Respectively, it was decided, to try adopting this design for Rivne Units 1 and 2.

Principal scheme of the ex-vessel cooling system is presented below on Fig. 1. The cooling circuit consists of:

- Inlet valve located in the corridor connecting BCT and reactor unit compartments and provides water to the ventilation ducts in concrete;
- Part of the ventilation ducts isolated by u-type siphon, this predicts water junction in other compartments;
- Reactor cavity compartment;
- Float valve installed on the biological shield below the reactor bottom;
- A gap between the reactor bottom and the shield;
- The vertical heated part in the annular gap between the reactor vessel and the reactor cavity with biological shield;
- Outlet between the reactor vessel and the bearing ring.

![Schematic solution of the external cooling circuit of the VVER-440 reactor vessel and nodalization for ATHLET model](image)

3. DEFINITION OF THE PROBLEM AND NUMERICAL ANALYSIS

First of all, operability of the above described circuit depends of availability of sufficient water inventory. The analysis of geometry of reactor compartment and the vessel of Rivne NPP Units 1, 2 has been done. The minimal free volume, which should be filled to obtain stable natural circulation is 863 m³, is much less than a volume of water on trays in BCT (1463 m³).

Next step is the analysis of the key aspects of ex-vessel cooling, as follows:

- Departure from nucleate boiling ratio at the reactor bottom, and
- Sufficiency of coolant flow rate and heat sink in the circuit for reliable reactor cooling.
The following method was used to provide a reliable solution of the safety problem:

— Selection of severe accident scenarios, based on results for Paks and severe accident assessment previously done for Rivne NPP. The “Large break LOCA 2*Dn 500 with total station blackout” is the most challenging scenario leading to the earliest melting and formation of corium.

— Calculation of in-vessel phase of accident with MELCOR computer code and comprehensive analytical model of NSSS of Rivne-1 to define transient progression and timing of main events, such as start of melting and further formation of corium. Time of corium formation gives us a total decay heat power, which should have been removed by ex-vessel cooling. The results of the calculations demonstrate that the maximum power of the decay heat is about 9 MW, which correlates well with the values obtained for the Paks NPP [4], 9.63 MW for the 107% of the initial reactor power.

— Definition of heat flux from corium to the coolant through the reactor vessel during in-vessel phase of severe accident was done using AIDA module of ATHLET-CD 3.1A. The AIDA model considers a segregated melt pool consisting of a lower oxide layer with decay power and an upper metal melt layer enclosed in a hemispherical reactor bottom. Moreover, considering the impossibility of modeling the semi-elliptical bottom of the WWER-440 reactor, the configuration of the melt pool with the minimum area equivalent to the hemispherical reactor bottom is simulated. The power of decay heat was taken from MELCOR calculations. At the outside surface of the reactor bottom model, constant third-type boundary conditions are set, simulating the heat transfer conditions to the coolant of the ex-vessel natural cooling circuit (the heat transfer coefficient for bubble boiling is 10,000 W/m²/K, the coolant temperature of the main flow is 120°C). This approach is similar to the assessment of ex-vessel cooling of the VVER 440 reactor of the Paks NPP [5], [5].

— Considering the significant uncertainties of calculations of key physical phenomena under severe accident the special uncertainty and sensitivity analysis was performed using SUSA 4.0 [6]. 34 initial parameters of AIDA model with most potential influence on calculation results were selected, including thermal and mechanical properties of materials of corium, type of mathematical models, ratio of zirconium oxidation and others. Sample ranking and preliminary analysis of the distribution of hypotheses, considering recommendations of SUSA developer [7] and [8], were done. 100 input decks for AIDA calculations were made by SUSA for combinations of parameters. SUSAs random number generator is based on the Monte Carlo method and the boundaries and the nature of the distribution of hypotheses. 100 calculations are sufficient to obtain a qualitative and reliable sensitivity analysis based on Wilks distribution [9] for a two-sided statistically tolerant interval with an upper and lower limit of 95% / 95%. To analyze the ranking of variable model parameters, the Spearman rank correlation coefficient [10] is used in SUSA. In this case, the application of the GRS method, included in the SUSA software, has a number of significant advantages in comparison with the methods of stochastic approximation in the analysis of uncertainty and sensitivity [11].

— The ex-vessel cooling system is simulated with AHLET code to check the operability natural circulation and sufficiency of flow rate for cooling.

4. RESULTS OF ANALYSIS

The maximum heat loading on RPV from melt pool is the most significant result of the analysis due to it defines heat transfer conditions on outer surface of RPV and, primarily, it could lead to the thermal failure of RPV. Thus, the analysis of sensitivity and uncertainty of maximum heat flux on the outer surface of RPV was done with SUSA.

Results of the analysis as a probability density function are presented on Fig. 2. It is easy to see; that maximum density of heat flux is less than 750-800 kW/m² is in good agreement with results of a probabilistic study of the melt behavior for the Loviisa NPP [12]. The most conservative values of the critical heat flux are obtained in experimental study on SBLB test facility [13]. Therefore, it could be concluded, that departure from nucleate boiling ratio resulting from comparison of the mentioned above values of the heat flux and critical heat flux, is sufficient.
A sufficiency of heat transferred from outside surface by natural circulation of ex-vessel cooling was checked with ATHLET 3.1A code. Fig. 1 presents the nodalization scheme of ATHLET model of ex-vessel cooling circuit. Model consist of following TFOs: BOX-SG – time dependent volume simulating boundary conditions for coolant inlet; VENT-CH – ventilation duct, located in concrete wall of reactor cavity; A-004 – lower part of reactor cavity compartment with float valve; CYRC-CH1,2,3 – the heated part in the gap between the reactor vessel and the biological shield; CYRC-CH4 – a gaps between blocks of biological shield surrounding inlet and outlet of primary system pipes; BOX-SG2 – time dependent volume simulating boundary conditions for coolant outlet.

The distribution of heat flux density at the reactor pressure vessel (RPV) obtained by results of uncertainty and sensitivity analysis for the maximum thermal load in the peak heat flux region is set as a boundary condition for AHLET calculations. The influence of a gap between RPV and thermal shield is checked by variant calculations with design gap and a gap reduced due to the possible thermal and mechanical deformation of RPV.

In both cases, the results of calculations confirm the reliability of heat removal and integrity of RPV. As expected, the behavior of a two-phase flow in the gap with variable geometry and coolant intensive boiling has an oscillating nature. This is intensified by the considerable unevenness of the heat load in the heated area, which leads to the irregularity of the vapor fraction in the flow, and, taking into account the interfacial friction, the outlet of generated vapor through the narrowest point of the circulation circuit (see Fig. 1). The maximum temperatures of RPV surface were obtained in points with the narrowest gap (see Fig. 1), due to a short-term transition to the dispersed flow of the two-phase coolant. Nevertheless, this does not lead to a noticeable increase in the surface temperature of RPV in those points.

As it shown on Fig.3, the process is quasistationary, the average value of heat transfer coefficient in area with average heat load (nodes 30÷35) is set at approximate value of 9000 ÷ 12000 W/m²/K. Maximum values of heat transfer coefficient were obtained for maximum heat flux areas (nodes 40÷50). The average value of coolant flow rate for two-phase circulations is equals to ≈150 kg/s (see, Fig. 3).
Fig. 3. ATHLET calculation results: A – heat transfer coefficient at the outer surface of the reactor vessel in a section with an average heat load; B – coolant mass flow rate in elevating circuit

5. CONCLUSIONS AND OPEN QUESTIONS

Analysis of IVR strategy implementation on WWER in EU countries the detailed consideration of the structural and schematic features of the reactor facilities for the RNPP-1,2 were done, and applicability of EU approach was confirmed. Numerical analysis of ex-vessel cooling system shows the sufficient DNBR and coolant flow is enough for reliable cooling of the reactor.

Open questions are:

Is a subcriticality of melted fuel will be maintained in case of relocation of fragments and possible in-vessel cooling (reflooding)?

What could happen in case of inadvertent operation of ex-vessel cooling system under normal operation of reactor facility? Would it leads to the brittle fracture of the reactor vessel?

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REFERENCES


