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THE THERMAL-HYDRAULIC ANALYSIS METHODOLOGY IMPROVEMENT IN FRAMEWORK OF VESSEL REACTORS LIFETIME EXTENSION

Oleksandr Mazurok
Odessa National Polytechnic University
Odessa, Ukraine

Volodymyr Kravchenko
Odessa National Polytechnic University
Odessa, Ukraine

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ABSTRACT

At the moment, Ukraine and other countries are implementing a program to extend the nuclear power plants operation over the design period, which is economically advantageous, as the level of capital expenditure required for this is significantly lower than for the new power units building. Particular attention is paid to assessing the technical condition and extending the reactor vessel operation (thermal shock assessment), as the vessel is the most expensive and complicated element of the power unit in terms of manufacturing and replacement. For this assessment, in particular, for the thermal-hydraulic analysis, special methodologies are used in the world. Analysis of the main existing methodologies of thermal-hydraulic analysis used in Ukraine and the world has shown that the practice of performing thermal-hydraulic analysis in Ukraine is ahead of theoretical knowledge, international experience and early national experience, and therefore requires optimization and improvement.

In the present work, the most relevant results of thermohydraulic, probabilistic and strength evaluations performed for SUNPP-3 were analyzed. In addition, the obtained results were compared with other results obtained for other power units recently. These estimates were performed with the purpose of the Ukrainian NPPs lifetime extension using existing methodology that can be improved with current experience.

Reason of all estimate types analysis is that the power unit extension is a complex task: thermal-hydraulic analysis of scenarios with a high probability realization is performed, and strength analysis allows estimating scenarios quantitatively and to conclude that the scenarios are representative in relation to a thermal shock or not. This can to allow reasonable to reduce the number of scenarios that require a quantitative assessment. Particular attention is paid to SUNPP Unit 3, which is now at the stage of life extension.

Methodology can be improved and after this can be used to lifetime extension of Ukrainian NPPs vessel reactors, for which the relevant work has not yet begun, as well as in the case of the repeated extension. In addition, the results of the work and the Ukrainian experience can be taken into account in the next edition of the IAEA guideline or practice.

1. INTRODUCTION

This article has analyzed the latest experience in the assessment of the reactor pressure vessel (RPV) technical condition at Ukrainian nuclear power plants (NPP). It does not make sense to use older experience, because as a result of each new calculation for each next power unit, new experience is gained, world experience, requirements and comments of the regulatory body, modernization of power units, etc. are taken into account.

Information regarding the lifetime of Ukrainian NPP with a VVER-1000 reactor is given in the Table 1 below.

Table 1. Information regarding the lifetime of Ukrainian NPP

Power Unit	**Commissioning**	**End of the design lifetime**	**Lifetime extension**
SUNPP1	1982	2013	2023
ZNPP1	1984	2015	2025
SUNPP 2	1985	2015	2025
ZNPP 2	1985	2016	2026
RNPP3	1986	**2017**	2037
ZNPP 3	1986	**2017**	2027
ZNPP 4	1987	**2018**	2028
KHNPP1	1987	**2018**	-
SUNPP 3	1989	**2020**	-
ZNPP 5	1989	2020	-
ZNPP 6	1995	2026	-
RNPP 4	2004	2035	-
KHNPP2	2004	2035	-

As can be seen from the Table 1, the lifetime of units RNPP3 and ZNPP3,4 has recently been extended. Units KhNPP1 and SUNPP3 are now in the active phase of the lifetime extension with the implementation of a large amount of modifications.

At the same time, the most up-to-date results of the calculation analysis are the work performed for the units RNPP-3, ZNPP-3,4 and KHNPP-1. Results for unit ZNPP-3 is absent in article as is provided more modern results for unit ZNPP-4. Thermal-hydraulic and probabilistic justification for these units were approved by the NPP Operator (SE NNEGC “Energoatom”) and the Regulator Body (SNRIU).

SUNPP-3, which is at the stage of preparation to lifetime extension, was selected for analysis. At the same time, the article authors took part in performing analyzes for SUNPP-3.

2. THERMAL-HYDRAULIC ANALYSIS METOGOLOGY

The Guideline (methodology) IAEA-EBP-WWER-08 [1] is the main international document containing a methodology for assessing the state of the RPV under pressure (VVER and PWR types) in part of performing thermal hydraulic analysis. The manual is the fundamental document for carrying out an analysis of this type in the world.

It should be noted that, if necessary, as part of the development of the Guidelines [1], more detailed national standards were used regarding RPV integrity (in particular, thermal shock), for example, PNAEG-7-002-86 [2] and Guidelines [3, 4] .

Also, within the framework of the VERLIFE project, a methodology [5] was prepared. Methodology VERLIFE included several sections and annexes relating to the RPV integrity assessment, developed as part of the preparation of the Guidelines [1].

In addition, in 2010 was issued IAEA document IAEA-TECDOC-1627 [6], which contains both recommendations for thermal shock assessment and practical results of applying various methodologies, guidelines and standards, some of which were used in Ukraine, namely, IAEA- EBP-WWER-08 [1], VERLIFE [5] and MRKR-SHR-2004 [7]. However, the experience of Ukraine is not taken into account in [6], since at that time work in Ukraine was just begun. In addition, it should be noted that [7] contains only the methodology for performing strength calculations without any information on the thermo-hydraulic part.

Since the Guidelines [1] contains only general recommendations, based on it, a domestic methodology for assessing the strength and lifetime of WWER reactors during the operation of Ukrainian NPPs MT-D.0.03.391-09 [8] was developed, which is basic, in particular to perform thermal and hydraulic analysis for all power units of Ukrainian NPPs.

Based on the available information, including [1, 5, 8], more detailed standard and work programs for assessing the technical condition and extending the lifetime of reactor elements [9, 10, etc.] are developed, as well as other documents containing methodologies for performing thermal-hydraulic analysis for Ukrainian NPPs. This takes into account the recommendations of the United States Nuclear Regulatory Commission (USA) documents [11, 12], which contain the assessment methodology used in the United States for PWR reactors and which, in terms of thermal-hydraulic evaluation, is practically the same as in [1] .

Thus, the most relevant document for the power unit for which the justification is being carried out is the work program developed specifically for this unit. However, since the programs are based on recommendations and methodologies, the requirements of which must be fulfilled necessarily, a number of aspects are not taken into account, such as the specifics of Ukrainian NPP units, in particular, recent upgrades and accumulated evaluation experience as part of the lifetime extension.

In [13] provides a more detailed analysis of the methodologies used in Ukraine.

3. PROBABILISTIC ASSESSMENT

In conducting thermal-hydraulic calculations in support of justifying the RPV integrity, deterministic and probabilistic approaches are used to take into account both scenarios that obviously lead to thermal shock and scenarios that may not lead to serious consequences, but have a high probability of ignored.

The probabilistic approach allows a more complete analysis of the initial events that are potentially hazardous from the point of view of the RPV integrity. First of all, this is due to the multiple failures of equipments.

As part of work in Ukraine, probabilistic analysis is no less significant than the deterministic one; therefore, it is performed in parallel and is mandatory (not optional). Based on a probabilistic analysis, dominant scenarios are determined that have a high probability of realization (frequency more than 1E-08 1/year) and potentially lead to thermal shock. Then, for the list of dominant scenarios, deterministic analysis is performed (quantitatively or qualitatively).

For the analysis, the codes SAPHIRE (ZNPP4, RNPP3) and RISKSPECTRUM PSA (SUNPP3) are used, and as base, the relevant integrated probabilistic model for the entire spectrum of accidents initiating events for all modes of the reactor facility.

In Table 1 is shown results of probabilistic analysis for unit SUNPP-3 with using code RISKSPECTRUM PSA. Also results for another units were presented in Table 1 for comparison. For comparison were selected dominant scenario groups with a frequency of more than 1E-05 1/year. Groups were analyzed for different reactor facility modes. All scenarios are characterized by the following ECCS configuration: more than 2 safety train in operation.

Tabl.1 PSA dominant scenarios

Scenario name	**Frequency, 1/year**			
	RNPP3	**ZNPP4**	**KHNPP1**	**SUNPP3**
Small primary leak (DN<11). Nominal power	-	1.37E-04	-	**4.97E-04**
Small primary leak (DN11-50). Nominal power	9.16E-03	5.65E-03	8.38E-03	**6.86E-03**
Small primary leak (DN11-50). Hot shut-down	1.70E-04	-	-	**-**
Medium primary leak (DN50-80). Nominal	4.83E-04	4.57E-04	4.53E-04	**5.33E-04**

Scenario name	Frequency, 1/year			
	RNPP3	ZNPP4	KHNPP1	SUNPP3
power				
Medium primary leak (DN50-80). Hot shut-down	1.56E-04	-	-	-
Small leak from primary to secondary. Nominal power	1.94E-04	3.91E-03	-	**5.99E-04**
Medium leak from primary to secondary. Nominal power	9.89E-04	9.78E-04	9.10E-04	**2.92E-03**
PRZ SV opening with non-closing. Nominal power	9.95E-04	-	-	-
PRZ SV opening with non-closing. Hot shut-down	1.08E-03	-	-	-
Main steam line breaks without SG localization. Nominal power.	-	-	1.25E-03	-
BRU-A (SG SV) opening with non-closing. Nominal power	-	4.81E-04	-	-
BRU-K (or turbine stop valve) opening with non-closing and with all BZOK failure. Nominal power	-	3.20E-04	-	-
BRU-A (SG SV) and BRU-K (or turbine stop valve) opening with non-closing. Nominal power	-	2.25E-04	-	-
Loss of heat removal through a secondary with “feed-bleed” procedure. Nominal power	1.58E-04	-	-	-

From the table 1 it can be concluded that scenarios with small and medium leaks of the primary coolant, as well as leaks from the primary to the secondary, have a high probability.

However, there are scenarios that are specific to a particular unit. This is due to the use of integrated models for different codes, the specifics of each individual unit, the human factor in the simulation, etc.

Thus, it can be further argued that small and medium leaks are required to perform a thermal-hydraulic analysis with a further assessment of strength. The remaining scenarios are typical for each unit and must be performed using the current model for this unit.

Probabilistic analysis is not auxiliary and should be considered when performing deterministic analysis.

4. DETERMINISTIC ASSESSMENT

The goal of thermal-hydraulic analysis is to determine the boundary conditions (thermal-hydraulic parameters) for selected representative scenarios. The obtained results are used to conduct a strength analysis in justifying the RPV integrity. At the same time, the thermal-hydraulic analysis is performed using both results: the probabilistic and engineering assessments.

Engineering (qualitative) assessments of the initial events is performed in order to reduce the total number of calculation scenarios selected to RPV integrity justification. In the course of the qualitative analysis, possible ways of the accident process are considered taking into account the overlapping of equipment failures (personnel errors).

In practice, so it happens, but a qualitative analysis is performed from the power unit to the power unit. Due to qualitative analysis, initial events are excluded from the quantitative analysis, for example, leading to an RPV external flooding, unintended actuation of the high-pressure injection or the makeup system (based on the design features of the reactor facility and the RPV protection from cold overpressure, respectively).

Besides, during analysis ibitial events of the same group are identified (and belonging to the same frequency category), which have similar development scenarios and lead to similar consequences, and a preliminary selection of the initial event representatives and, accordingly, scenarios for the subsequent detailed modeling. However, all initial events of the group are considered (whole spectrum of leaks, etc.), representatives are not selected, but according to the results of qualitative analysis, preliminary calculations and experience of previous analyzes - the most conservative variant of the emergency scenario of each initial event is chosen. Such a calculation is made and boundary conditions are prepared for the strength assessment.

But, even with such an optimized grouping, a large array of scenarios is still selected for the calculation analysis, for example, a calculation analysis more than 55 emergency scenarios was performed for SUNPP-3. Such a scope of scenarios requires a lot of labor costs. At the same time, most of the scenarios are characterized by a high maximum permissible critical temperature of RPV fragility (T_k^a), therefore, due to the uniformity of power units, the list of scenarios requires optimization after a detailed analysis of the already performed calculation justifications, collecting data on the specifics of all power units, including planned upgrades, to identify differences.

4.1 Dominant scenarios

Strength specialists using linear fracture mechanics methods analyzed all the scenarios obtained using the RELAP code for SUNPP-3. For the RPV cylindrical part the following 6 dominant scenarios were selected:

- LOCA 2.1.2.2 “The primary coolant leak with an equivalent diameter 32 mm with the maximum ECCS configuration in the hot shut-down mode of reactor facility”;
- MSLB 2.2.5.2 “BRU-K opening with non-closing with the operation minimum ECCS configuration in the hot shut-down mode of reactor facility”;
- OTHER 2.4.1.10 "PRZ SV opening with non-closing with minimum ECCS configuration with closure on 5200 s in the hot shut-down mode of reactor facility”;
- OTHER 2.4.1.12 "PRZ SV opening with non-closing minimum ECCS configuration with closure on 5400 s in the hot shut-down mode of reactor facility without full blackout";
- PRISE 2.3.3.1 "Rupture of three SG heat exchanging

tubes with the minimum ECCS configuration in the hot shut-down mode of reactor facility;

- PRISE 2.3.3.2 "Disruption of three SG heat exchanging tubes with the maximum ECCS configuration in the hot shut-down mode of reactor facility".

The abbreviated names of the scenarios correspond to the real names used in the analysis and are used for ease of paperwork.

Further, the final assessment of the dominant scenarios was performed by specifying the parameters in the descending section of the reactor with subsequent strength analysis using the nonlinear fracture mechanics method.

4.2 Mixing analysis

For 6 dominant scenarios a refined analysis of the coolant mixing in the reactor downcomer on the vessel inner surface was performed. This was done for scenarios characterized by thermal stratification and flow stagnation at the reactor inlet.

Scenarios associated with secondary leaks are not characterized by flow stagnation in the emergency loop, but, on the contrary, a natural circulation loop is formed with intensive flow mixing. Therefore, for the MSLB 2.2.5.2 scenario, no mixing analysis is required, and the boundary conditions obtained by the RELAP code are applied directly for the strength analysis.

Such additional analysis is a requirement of the IAEA guideline [1].

Figures 1-5 below show a comparison of the coolant temperature near weld No. 4, obtained using the GRSMIX and RELAP codes for the dominant scenarios.

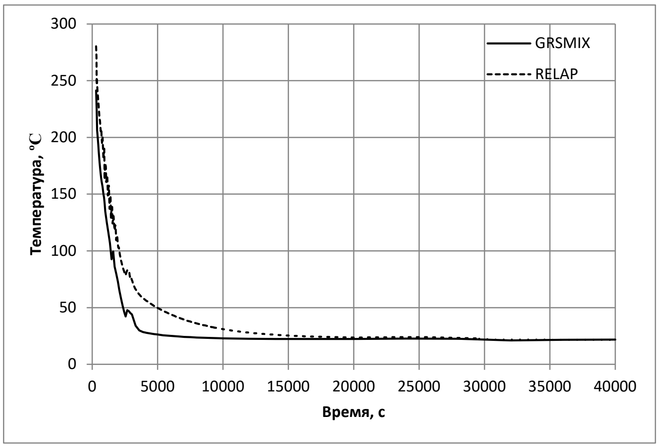

Fig.1 - Coolant temperature near the weld no. 4 for the scenario LOCA 2.1.2.1

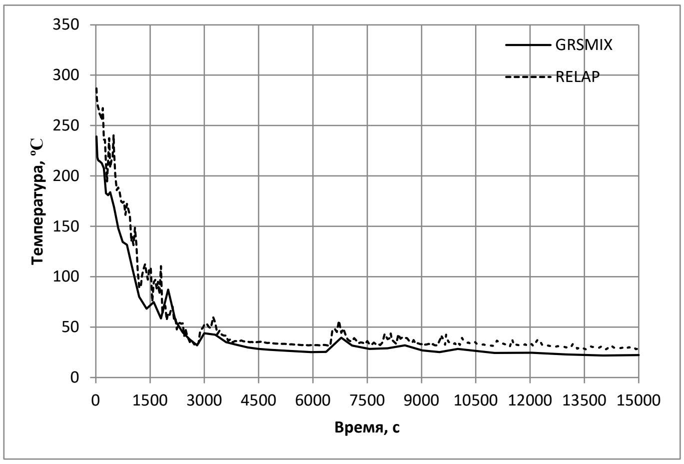

Fig.2 - Coolant temperature near the weld no. 4 for the scenario LOCA 2.1.5.4

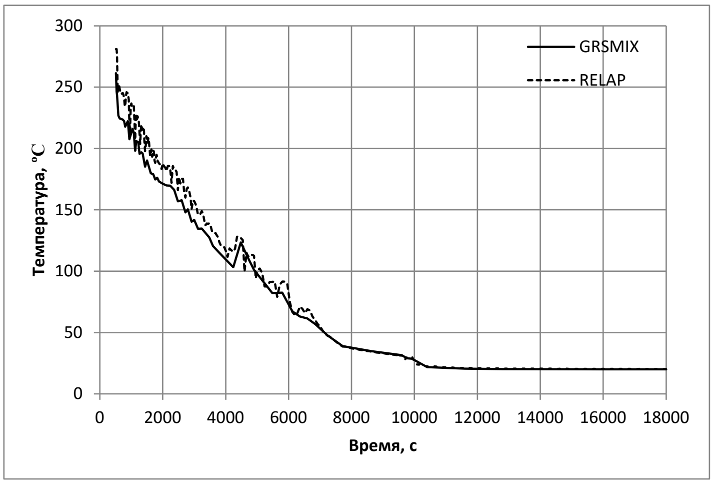

Fig.3- Coolant temperature near the weld no. 4 for the scenario PRISE 2.3.2.2

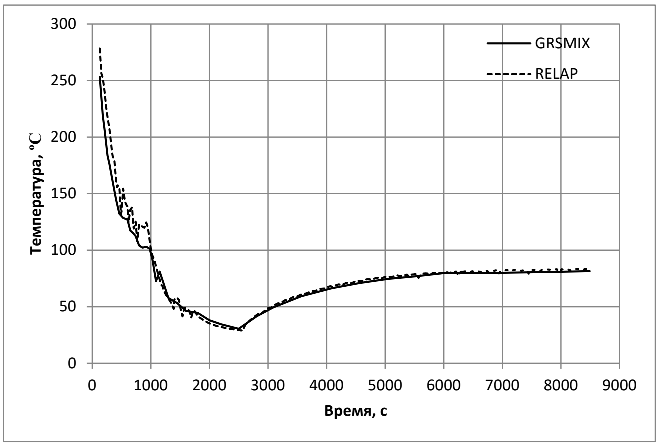

Fig.4 - Coolant temperature near the weld no. 4 for the scenario OTHER 2.4.1.2

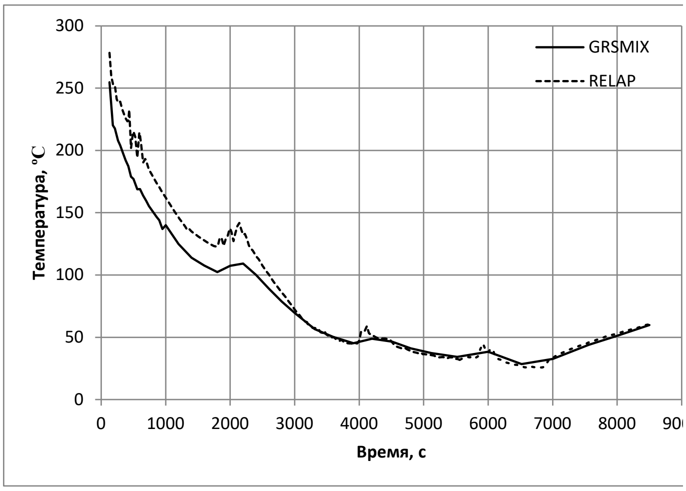

Fig.5 - Coolant temperature near the weld no. 4 for the scenario OTHER 2.4.1.5

Based on Figures 1-5, it can be concluded that the updated values of the parameters in the descending section of the reactor are more conservative in relation to thermal shock.

4.3 Final results

In any case, the above results of the thermal-hydraulic analysis make it possible to evaluate the influence of the parameters of the coolant during the course of emergency processes only at an estimated level. Quantitative assessment can be performed only by performing a strength analysis using the methods of nonlinear fracture mechanics. Table 2 presents the results of determining T_k^a in the zone of RPV weld No. 4 for SUNPP-3. Additionally, as a demonstration, these results were compared with current results obtained for other power units of Ukrainian NPPs.

It is obvious that the scenarios for different units are slightly different, but in general, the basic scenarios initial and boundary conditions are consistent and selected conservative. Additionally, Table 2 provides information on the ECCS configuration and the initial capacity of the power units, since the residual energy release residual energy release and the ECCS flow rates are decisive during accidents.

Tabl.2 Results (Tka) for dominant scenarios

Scenario name	**Tк^a, °C**			
	RNPP3	**ZNPP4**	**KHNPP1**	**SUNPP3**
Small primary leak DN32. Maximum ECCS. Hot shut-down	-	100.66	-	**75.87**
Large primary leak DN105. Maximum ECCS. Hot shut-down	-	-	68.6	-
Large primary leak DN125. Minimum ECCS. Nominal power	71.5	-	-	**94.2**
BRU-K opening with non-closing. Minimum ECCS. Hot shut-down	108.2	114.67	-	**114.15**
Main steam line breaks without SG localization. Maximum ECCS. Nominal power	-	-	79.1	-
Small leak from primary to secondary (1 SG tube break). Minimum ECCS. Hot shut-down	74.4	-	-	-
Small leak from primary to secondary (3 SG tubes break). Maximum ECCS. Hot shut-down	-	77.05	-	**69.54**
Small leak from primary to secondary (3 SG tubes break). Minimum ECCS. Hot shut-down	-	77.32	-	-
Medium leak from primary to secondary (collector break). Maximum ECCS. Hot shut-down	-	-	71.4	-
PRZ SV opening with non-closing immediately. Maximum ECCS. Hot shut-down	-	-	71.2 (without closing)	**56.92 (with closing at 2570 s)**
PRZ SV opening with non-closing immediately. Minimum ECCS. Hot shut-down	68.4 (with closing at 3600 s)	50.64 (with closing at 5400 s without blackout)	-	**55.90 (with closing at 5800 s)**

From the Table 2 it can be concluded that the representative scenarios are associated with PRZ SV opening with non-closing, with small and medium leak of the primary coolant, as well as with the leak from the primary to the secondary.

Large primary leaks and all secondary leaks are not representative. For the representative of the secondary leak group (BRU-K opening with non-closing), were obtained value of Tka is more 100 °C. Large primary leaks did not become dominant scenarios, as they are characterized by a rapid primary pressure decrease up to the ECCS (lower pressure injection) operating conditions at a pressure of about 23 kgf/cm^2 or less.

5. CONCLUSIONS

The article presents the results of probabilistic and deterministic analysis performed for SUNPP-3. These results were compared with the most relevant results for other Ukrainian NPPs.

This confirmed the need to perform a probabilistic assessment using current probabilistic models. At the same time, the probabilistic assessment is mandatory (not auxiliary) and allows for the identification of scenarios with a high frequency of implementation. These scenarios must be analyzed deterministically.

The deterministic analysis led to the conclusion that some groups of scenarios are not representative and can be excluded qualitatively at the grouping stage.

The findings can be applied to make changes to the current national methodology and the IAEA guidelines.

NOMENCLATURE

Tka - maximum permissible critical temperature of RPV fragility

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