UDC 621.039.058

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ANALYSIS OF EXISTING THERMAL-HYDRAULIC ANALYSIS METHODOLOGIES IN THE FRAMEWORK OF RESOURCE EXTENSION OF REACTOR PRESSURE VESSELS

О.С. Мазурок, В.П. Кравченко. **Аналіз існуючих методик теплогідравлічного аналізу в рамках робіт з продовження ресурсу корпусів реакторів**. На даний момент Україна і ряд інших країн реалізують програму з продовження експлуатації АЕС у понадпроектний термін, що економічно вигідно, оскільки рівень капітальних витрат, необхідних для цього, значно нижче, ніж при будівництві нових енергоблоків. Особлива увага приділяється оцінці технічного стану та подовження експлуатації корпусу реактора, оскільки корпус є найдорожчим і найскладнішим елементом енергоблоку з точки зору виготовлення та заміни. Для даної оцінки, зокрема, для теплогідравлічного аналізу в світі застосовуються спеціальні методики. У даній роботі проаналізовано основні існуючі методики теплогідравлічного аналізу, що застосовуються в Україні та світі, з метою визначення їх актуальності по відношенню до специфіки та поточному стану енергоблоків АЕС України Намічено шляхи оптимізації теплогідравлічного аналізу, зокрема, за рахунок обґрунтованого зменшення кількості розглянутих сценаріїв. Наукова і практична цінність даної роботи полягає в тому, що виявлені дефіцити діючих методик з урахуванням практичного досвіду дозволять розробити актуальну методику, яка дасть можливість зосередитися на більш глибокому дослідженні представницьких аварійних сценаріїв і буде застосовна для теплогідравлічного аналізу АЕС України з урахуванням наявного на даний момент вітчизняного досвіду. Методологія дослідження полягає в аналізі діючих методик з відповідною оцінкою їх застосовності і актуальності для АЕС України, а також урахуванням практики виконання теплогідравлічного аналізу під час продовження ресурсу вітчизняних АЕС. Дослідження дозволило зробити висновок, що практика виконання теплогідравлічного аналізу в Україні випереджає теоретичні знання, міжнародний і вітчизняний ранній досвід. Таким чином, вітчизняні методики вимагають актуалізації або удосконалення і можуть бути застосовані в рамках робіт з продовження експлуатації енергоблоків АЕС України, для яких відповідні роботи ще не розпочаті, а також, під час продовження повторно.

Ключові слова: продовження ресурсу, корпус реактора, теплогідравлічний аналіз, термоудар, методика

O. Mazurok, V. Kravchenko. **Analysis of existing thermal-hydraulic analysis methodologies in the framework of resource extension of reactor pressure vessels**. At the moment, Ukraine and a number of other countries are implementing a program to extend the operation of nuclear power plants beyond the design period. This is economically advantageous, since the level of capital expenditures required for this is much lower than with the construction of new power units. Special attention is paid to assessing the technical condition and extending the operation of the reactor vessel, since the vessel is the most expensive and complex element in terms of manufacturing and replacement. For this assessment, in particular for thermal analysis, in the world special techniques are applied. In this paper, we analyze the main existing methods of thermal-hydraulic analysis used in Ukraine and the world, in order to determine their relevance in relation to the specifics and current state of Ukrainian NPP units. Ways of optimizing thermal-hydraulic analysis are outlined, in particular, due to a reasonable reduction in the number of considered scenarios. The scientific and practical value of this work lies in the fact that the identified deficiencies in existing methods, taking into account practical experience, will allow developing a current methodology that will allow you to focus on a deeper study of representative accident scenarios and will be applicable for the thermal-hydraulic analysis of Ukrainian NPPs, taking into account the current domestic experience. The research methodology consists in the analysis of existing methods with an appropriate assessment of their applicability and relevance for NPPs of Ukraine, as well as taking into account the practice of performing thermal hydraulic analysis when extending the life of domestic NPPs. The study led to the conclusion that the practice of performing thermal–hydraulic analysis in Ukraine is ahead of theoretical knowledge, international and domestic early experience. Thus, domestic methods require updating and improvement and can be applied as part of work to extend the operation of Ukrainian NPPs for which the relevant work has not yet begun, as well as when renewing.

Keywords: lifetime extension, reactor vessel, thermal hydraulic analysis, thermal shock, method

Introduction. According to the “Energy Strategy of Ukraine for the period until 2035”, it is planned to keep the share of electricity production at nuclear power plants (NPPs) at the reached level – about half of the total domestic production. To accomplish this task, it is necessary to increase the safety, reliability and efficiency of existing NPPs operation, to continue NPPs operation beyond the design time, to build and commission new power units. Lifetime extension is economically advanta-

DOI: 10.15276/opu.3.56.2018.04

ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

geous, as the level of capital expenditure required for this (analyzes, modernization of reactor facility equipments and systems), significantly lower than in the building of new power units.

In framework of NPPs lifetime extension particular attention is paid to assessing the technical condition and lifetime extension of the reactor pressure vessel (RPV), since the RPV is the most expensive and complex element of the power unit in terms of manufacturing and replacement. In particular, in order to prepare the boundary conditions for the further determination of the temperature and stress fields in the RPV material the thermodynamic analysis of series of emergency scenarios leading to a RPV thermal shock, including system analyzes and analyzes of stream mixing, are performed.

Below is the analysis of the applied methodologies, in particular, comparison of recommendations and requirements of the methodologies with their practical application in performing thermal-hydraulic analysis for Ukrainian NPPs.

Analysis of recent publications and problem statement. Applied methodologies are practical documents that contain requirements and recommendations for the implementation of the analysis, therefore, studies and publications may not be directly related to the methodologies, but for performing computational analysis with their using, etc.

The main international document containing the methodology for assessing technical condition of RPV under pressure (VVER and PWR types) in the part of performing thermal-hydraulic analysis is Guideline (methodology) IAEA-EBP-WWER-08 [1]. The Guideline is the fundamental document for the implementation of this analysis type in the world. After publication in 1997, the Guideline was applied in different the IAEA thermal benchmark, and was also widely used in Member States operating VVER NPPs. After this, revision 1 of the Guideline [1] was issued in 2006.

Also, within the framework of the VERLIFE project of the fifth framework program of the European Union, in the period 2001…2003, a was developed methodology, which included several sections and annexes on the RPV integrity evaluation developed in framework of the Guideline [1] preparation. In 2008 and 2013 the VERLIFE methodology was revised and update, but for Ukraine, the 2008 version is applied [2]. Also the document IAEA-IAEA-TECDOC-1627 [3] was published in 2010, which contains both recommendations for the evaluation of thermal shock, and practical results of applying various methodologies, guidelines and standards, some of which were applied in Ukraine, namely [1, 2, 4]. However, the experience of Ukraine is not taken into account in [3], since at that time the work in Ukraine was only begun. Also, it should be noted that [4] contains only the methodology for performing strength calculations without any information on the thermohydraulic part.

As the Guideline [1] contain only general recommendations, therefore on its basis a domestic methodology of VVER RPV strength and resource evaluation in the process of operating Ukrainian NPPs MT-D.03.03.391-09 [5] was developed. Methodology [5] is basic, in particular, to perform thermal-hydraulic analysis for all power units of Ukrainian NPPs. Based on the available information, including [1, 2, 5], more detailed standard and work programs are developed for assessing the technical condition and lifetime extension of reactor elements, as well as other documents containing methodologies for performing the thermal-hydraulic analysis for Ukrainian NPPs. At the same time, are taken into account the recommendations of the United States Nuclear Regulatory Commission (USA) documents [6, 7], which contain the estimation methodology used in the USA for PWR type reactors and which, in terms of the thermal-hydraulic evaluation, practically does not differ from the procedure given in [1].

Thus, the most up-to-date document for a calculated power unit is a work program developed specifically for it. However, as the programs are based on recommendations and methodologies, the requirements of which must be fulfilled, therefore a number of issues, such as the specificity of the Ukrainian NPP power units, in particular, the recent modernization and the accumulated experience during the lifetime extension, are not taken into account.

Purpose and objectives of the study. The purpose and task of the study is to analyze the existing methodologies of thermal-hydraulic analysis used in the RPV lifetime extension. The task is reduced to the analysis of international [1] and domestic [5] methodology. A their detailed analysis of the methodologies with taking into account the experience of the assessment as part of the lifetime extension, will allow developing recommendations for its improving, consisting in taking into account the current state of power units and allowing to optimize the performance of thermal-hydraulic analysis. The improved methodology will reduce the time spent on performing thermo-hydraulic analysis by reducing the required scope of work and improve the quality of the results obtained.

The objectives are defined below, and the main stages of the thermal-hydraulic analysis are analyzed: the selection of the approach to calculations, the collection of basic data, the calculation models development, the selection of calculation scenarios for analysis, the calculations execution.

The purpose of thermal-hydraulic analysis. As determined in the current methodologies, the purpose of thermal-hydraulic analysis is to determine the boundary conditions for selected representative emergency or transient regimes (scenarios). The obtained results are used to perform strength analysis in justification of the RPV integrity.

Boundary conditions are the time dependences of the following parameters for the scenarios accompanied by cooling of the coolant: the temperature field in the reactor downcomer, the heat transfer coefficients from coolant to the PPV wall and the pressure in reactor downcomer.

The execution of thermal-hydraulic calculations in justification of the RPV integrity is a complex task, including a number of engineering analyzes and calculations using computer codes. In general, the content of the work on thermal-hydraulic analysis is similar to the content of the work when performing calculations for the accidents analysis as part of the power units safety analysis and includes the following main steps: the choice of the approach to the calculations, the collection of the initial data, the development of calculation models, the choice of calculation scenarios for analysis, the calculations.

The above steps in this order are defined in the methodology [5] and were having been taken as the basis for analysis of the methodologies below.

The choice of the approach to the calculations. When executing thermal-hydraulic calculations in support of the RPV integrity justification, deterministic and probabilistic approaches are used to take into account both scenarios that will obviously lead to thermal shock and scenarios that may not lead to serious consequences but have a high probability of realization, therefore can not be ignored.

In accordance with [1] and [5], the deterministic approach is based on the selection initial events (from the events list), which represent a potential danger from the point of view of the RPV integrity, and their subsequent analysis with using a conservative assumptions. A characteristic feature of this approach is the use of the principle of single failure for safety systems (i.e., multiple failures of equipment are not considered). In general, the approach is similar to that used in the analysis of design basis accidents. In order to reduce the number of necessary calculations, a qualitative analysis is usually carried out first. Based on the results of qualitative analysis from list of initial events leading to similar consequences, are determined by event-representatives (i.e., events which lead to more adverse consequences from the point of view of the target calculation parameters, for example, to a lower coolant temperature in the reactor downcomer at high primary pressure) for their subsequent detailed modeling using thermal-hydraulic calculation codes.

The probabilistic approach allows for more complete analysis events, potentially dangerous in terms of the RPV integrity. First of all, this is due to the consideration of multiple equipment failures.

In [5] it was noted that the use of the probabilistic approach requires a large amount of labor (in comparison with the deterministic approach). Therefore, the probabilistic approach is usually applied in addition to the deterministic approach and is used after calculations using the deterministic approach in case of need for in-depth analysis. It was also noted in [2] that probabilistic analysis of the occurrence of thermal shock is regarded as additional to the deterministic.

However, in practice, in the framework of work in Ukraine, probabilistic analysis is no less significant than deterministic analysis, therefore it is carried out in parallel and is mandatory (not additional). On the basis of probabilistic analysis, dominant scenarios with a high probability of realization (frequency more than 10^{-8} 1/year) and potentially leading to thermal shock are determined. After that, for the dominant scenarios list, a deterministic analysis is performed (quantitatively or qualitatively). For the analysis, the codes SAPHIRE and RISKSPECTRUM PSA are used, and the corresponding integral probability models for the full spectrum of the accident initial events for all reactor facility modes.

ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

The initial data collection. As for analysis requires the engineering assessment execution, preparation of calculation models and the development of calculation scenarios, therefore, it is required to perform the initial data collection in an amount sufficient to complete the calculation analysis for researched the power unit. In [1] and [2], there are no specific requirements for the initial data collection. However, the methodology [5] puts forward the requirements for the development of a separate database, which should be developed directly within the framework of this task.

In practice, within the framework of work in Ukraine, separate database are not developed. The available database for the reactor facility, systems and containment (if available), the drawings of the equipment and pipelines, the operation instructions, the technological regulations of safety operation, the lists of the protections and interlocks, the neutron-physical calculations of fuel loads, technical solutions and corresponding documentation for the modernization implementation and other technical documentation. Initial data are limited to equipment and systems that are part of the modeling boundaries.

The calculation models development. To perform the thermal-hydraulic analysis appropriate calculation models for integral thermal-hydraulic codes and engineering codes that take into account mixing in downcomer should be developed. For the analysis in Ukraine, the following codes are used: RELAP5 (definition of boundary conditions with quasi-three-dimensional downcomer modeling), GRS-MIX and REMIX/NEWMIX (obtaining the specified boundary conditions for the dominant scenarios with considering mixing in downcomer). Typically, as models for the RELAP5 code are used by the NPPs operator, which are developed within the framework of the safety justification, for example, for the analysis of design accidents. The requirements to the model development, validation and documentation are similar to other types of analysis for Ukrainian NPPs.

In addition, researches on the CFD codes using for the analysis of thermal shock are performed in the world also (for example, some results are showed in [8 – 11]), but now domestic methods do not require this type of analysis.

The choice of calculation scenarios for analysis. The identification of those initial events with a potential threat to the RPV integrity to select the calculation scenarios must be performed.

In accordance with [5] a general list of the initial events specified for researched NPP is taken as a basis. This list is described and analyzed in the normative and technical documentation, including safety analysis reports. It is recommended to select all initial events and scenarios that potentially affecting to the RPV integrity from the list of initial events. In addition, according to [5] the initial events in the groups should be considered: decrease of the primary coolant reserve, increase of the coolant reserve, increase of heat removal through the secondary, external cooling of the RPV. Thus, the approach is analogous to the approach to the design basis accidents analysis. However, in the framework of Ukrainian projects realization is applied the approach given in [1, 2], where the grouping is performed in a different way, namely, the initial events are related to the groups: Loss of coolant accidents (LOCA), primary to secondary leakage accidents (PRISE), large secondary leaks (MSLB), stuck open pressurizer safety or relief valve, inadvertent actuation of the high pressure injection or make-up systems, accidents resulting in cooling of the RPV from outside. The initial that are not present in groups LOCA, PRISE and MSLB belong to the group OTHER. In addition, based on the experience of the justifications and the high probability of realization (frequency $1.73{\cdot}10^{-6}$ 1/year), the scenario “Total loss of the feed water of the steam generator (SG)” with the implementation of the procedure “feed-and-bleed”, which is characterized by a fast pressure decrease in the primary, followed by cooling from cold water from the emergency core cooling system (ECCS).

As indicated above, probabilistic analysis is performed in addition to the deterministic evaluation. As a result of the probabilistic evaluation, the list of calculation scenarios can also be complemented, for example, in the framework of the RPV integrity justification for RNPP–3, a scenario with the main steam collector break was analyzed with a failure to closing all quick-acting shut–off valve (multiple failure), characterized by a high probability of realization (frequency $1.58{\cdot}10^{-4}$ 1/year).

As an example, Table contains lists of 10 dominant scenarios groups with a high probability of realization, obtained for the SUNPP-3, with the preliminary calculation results of the reserve to brittle fracture T_{ka} (for in formativeness). Based on the presented results, it can be concluded that the dominant (more probable) scenarios may not be representative from the results of strength analysis. The applied methodologies for calculating the probability and T_{ka} are given in documents [12, 13] and [14, 15].

Lists of 10 dominant scenarios groups with a high probability of realization

№	Scenario Group / Frequency, 1 year	Description	Scenario	T_{ka}, °C
1	PTS-1N-A1/$8.44\cdot10^{-5}$	Large primary leakage DN90-350 with operation 3 train of HPIS (high-pressure injection) and LPIS (low-pressure injection) system	LOCA 2.1.6.3	47.7
2	PTS-2N-1/$5.33\cdot10^{-4}$	Medium primary leakage DN50-90 with operation only 1 train of HPIS	LOCA 2.1.3.1	49.77
3	PTS-3N-1 (reactor facility mode: “on power”)/$6.86\cdot10^{-3}$	Small primary leakage DN11-50 with operation 3 train of HPIS	LOCA 2.1.2.1	30.14
4	PTS-3S-1 (reactor facility mode: “hot shutdown”)/$7.58\cdot10^{-5}$			
5	PTS-3N-4/2.76×10^{-5}	Small primary leakage DN11-50 with operation 1 train of HPIS	LOCA 2.1.2.2	88.2
6	PTS-4N-1/$4.97\cdot10^{-4}$	Small primary leakage DN<11 with operation 1 train of HPIS due to failure of the make-up system	LOCA 2.1.1.1	178.65
7	PTS-6N-1/$5.99\cdot10^{-4}$	Small leakage from primarty to secondary with failure of: make-up system, emergency SG localization, heat removal through secondary	PRISE 2.3.1.2	62.94
8	PTS-7N-11 (reactor facility mode: “on power”)/$2.92\cdot10^{-3}$	Medium leakage from primary to secondary (break of several heat exchange tubes) with operation 3 train of HPIS at the initial accident stage and the successful switching of to the recirculation of redundant HPIS train after the cooldown through secondary	PRISE 2.3.2.2	36.43
9	PTS-7S-11 (reactor facility mode: “hot shutdown”)/$3.16\cdot10^{-5}$			
10	PTS-7N-12/$9.73\cdot10^{-4}$	Medium leakage from primary to secondary (SG collector cover opening) with operation 3 train of HPIS at the initial accident stage and the successful switching of to the recir-?ulation of redundant HPIS train after the cooldown through secondary	PRISE 2.3.3.3	45.06

Based on the presented results, it can be concluded that the dominant (more probable) scenarios may not be representative accordance to the strength analysis results. For example, for the scenario LOCA 2.1.1.1 (with a high probability of implementation $4.97\cdot10^{-4}$ 1/year) a high value of T_{ka} (178.65 °C) was obtained.

In addition, it is stated in [5] that in order to reduce the total number of calculations, a preliminary qualitative (engineering) analysis of the initial events selected to the RPV integrity justification should be performed. Possible ways of accident taking into account the equipment failures and staff mistake are considered in the qualitative analysis. In practice, this is the case, but a qualitative analysis is performed for every next NPP unit, as a result of which the part of initial events are excluded from the quantitative examination, for example, PRV external cooling, inadvertent actuation of the high pressure injection or make-up systems (based on the reactor facility design features and the availability of RPV cold overpressure protection, respectively).

ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

In addition to this, the initial events of the same group are identified during the analysis (and belonging to the same frequency category). These groups should have similar chronology of scenarios and lead to similar consequences. In this case, a preliminary selection of the initial events-representatives and, accordingly, scenarios for subsequent detailed modeling is performed. However, in fact, all the initial events of the group (all dimensions of leaks etc.) are actually considered, representatives are not selected, but based on the results of qualitative analysis, preliminary calculations and the previous analyzes experience, the most conservative scenario – for each initial event, such a calculation is issued and initial data are prepared for the strength evaluation.

But, even with such an optimized grouping, a large array of scenarios is still selected for the analysis, for example, for the SUNPP-3 was performed a calculation analysis of 55 accident scenarios. Such a scope of scenarios requires a lot of labor costs, although the result is a single number of dominant scenarios. At the same time, most scenarios are characterized by a fairly large T_{ka}, therefore, in view of the power units uniformity, the list of scenarios requires optimization after: a detailed analysis of the already completed justifications, data collection on the specifics of all power units to identify differences, including planned modifications. For example, Fig. 1 and 2 show the results of the preliminary strength evaluation of the above scenarios in the definition of T_{ka}.

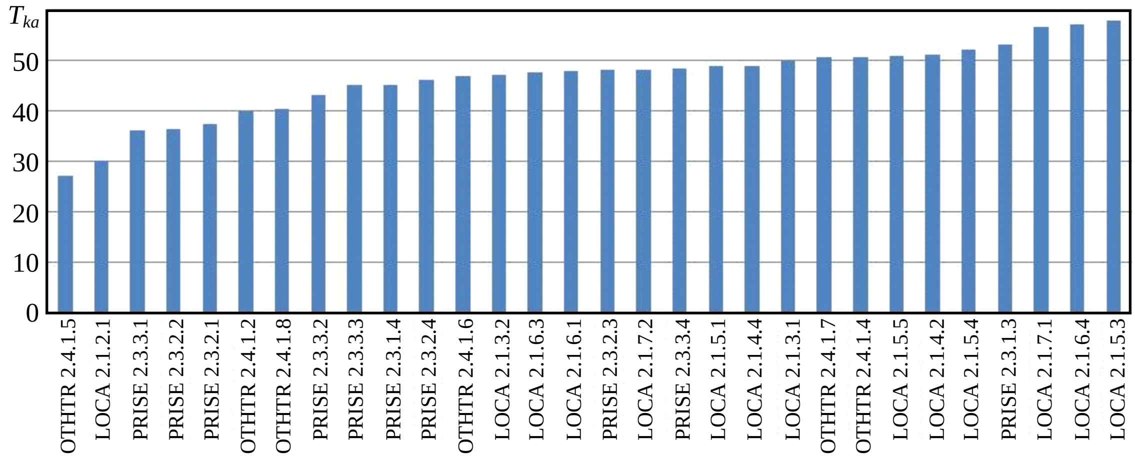

Fig. 1. Margin to brittle strength (T_{ka}), the first part of the scenarios

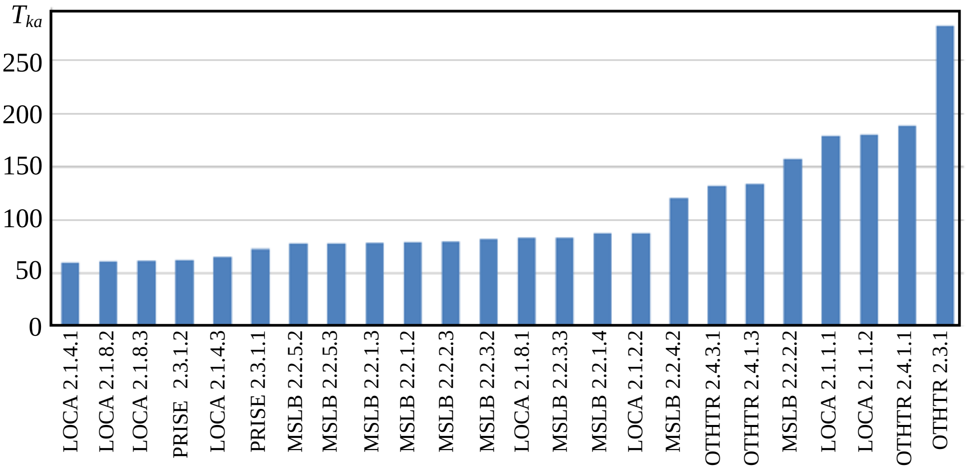

Fig. 2. Margin to brittle strength (T_{ka}), the second part of the scenarios

Shown results allow us to conclude that only 6 scenarios are characterized by T_{ka} of less than 40 °C, and scenarios of the MSLB group are not representative (T_{ka} over 78 °C, the scenario

MSLB 2.2.5.2 is the 38th in the list in ascending T_{ka} list) and can be excluded from the quantitative assessment after a detailed study.

Calculations performing. First of all, the formulation of the task is carried out. Task consists in choosing an approach to carrying out the calculation analysis and determining the initial and boundary conditions. In practice, a conservative approach to performing analysis is applied in accordance with [1, 2]. However, the methodology [5] also permits the application of a more labor-intensive approach with an best-estimate (realistic) approach, which is not actually applied in the framework of the work in Ukraine.

Further, the initial and boundary conditions are determined [5]. The initial conditions include parameters that characterize the state of the power unit, which are directly measured at the station (for example, pressure, temperature, etc.), or can be calculated (for example, reactivity coefficients, regulators efficiency, etc.). In addition, the initial conditions include some design conditions, for example, the limiting values of the non-uniformity coefficients (reactor kinetics). In the conservative approach, the limiting values of the parameters (taking into account the inaccuracies of the measurement systems) from the range characterizing the normal NPP operation (i.e., operational limits) should be chosen as initial conditions.

Boundary conditions include the configuration and characteristics of systems and equipment involved in overcoming the accident consequences. In the deterministic approach, the principle of conservatism and single failure is used, while in the probabilistic approach multiple failures can be considered.

In a conservative approach, parameter values accepted as boundary conditions should lead to worst-case consequences for a given initial event, taking into account the permissible deviation of the parameter from the nominal value. Also, the loss of external sources of energy supply should be taken into account as an additional failure, if this further worsens the results of the analysis [1, 2, 5]. In particular, in accordance with [2], failures of elements of normal operation systems should be considered only for scenarios with loads leading to more conservative thermal shock conditions.

Two groups of operator actions are considered that have a significant effect on the transient process, leading to a thermal shock [1, 2, 5]. The first group is the operator's actions, which lead to an aggravation of the thermal shock conditions, the second group – leading to a softening. For the second group, it is necessary to demonstrate that the operator has sufficient information and a period of time for decision making and implementation of actions, the appropriate qualifications for the implementation of such an action. This only takes into account the actions that are prescribed by the current operating instructions. In Ukraine it is symptom-oriented emergency response instructions. It should be noted that the time of the operator is an important aspect.

Further, a calculation analysis to obtain the boundary conditions for the strength evaluation is performed. Boundary conditions are prepared in the form of tabular data in Excel format. Using these datas strength analysis is performed and scenarios, characterized by the smallest margin to brittle strength violation are determined. For scenarios characterized by a low rate of coolant natural circulation (or flow stagnation) in the main circulation pipes, a refined thermohydraulic analysis of the coolant mixing in the downcomer is performed using special engineering codes that allow obtaining improved boundary conditions on the RPV inner surface. However, such an analysis is performed only for the dominant scenarios, after which the specified boundary conditions are applied for the repeated scenario strength analysis.

Research results. The study analyzed the main stages of the thermal-hydraulic analysis, namely: the selection of the approach to calculations, the collection of basic data, the calculation models development, the selection of calculation scenarios for analysis, the calculations execution. For each stage were set out approaches to their implementation in accordance with the requirements of various methodologies, have been described the experience of using methodologies in Ukraine, were given domestic results of work and were identified methodologies deficiencies, which will allow to continue the study in terms of updating the existing methodologies or the development of modern methodologies.

Conclusions. The practice of thermohydraulic analysis performing in Ukraine is ahead of theoretical knowledge, practical experience, international experience and domestic early experience.

ISSN 2076-2429 (print)
ISSN 2223-3814 (online)

Therefore, domestic methodologies require updating or improvement. The experience of Ukraine can be taken into account in the next revision of the IAEA guidelines or practices.

At the domestic level, the methodology can be optimized by reducing the number of scenarios for which a quantitative assessment is required. Optimization can be performed through a more in-depth analysis of analyzes results for Ukrainian NPPs executed and approved by the State Nuclear Regulatory Inspection of Ukraine (SNRIU), taking into account the "feedback" between the results of strength and thermohydraulic analysis.

Up-to-date or newly developed modern methodology after the approval by SE "NNEGC "Energoatom" and SNRIU can be applied in the framework of Ukrainian NPPs units lifetime extension. This applies to units for which the relevant work has not yet been started, and also when it is renewed repeatedly.

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Received October 3, 2018

Accepted December 18, 2018