THERMAL-HYDRAULIC CALCULATIONS FOR TECHNICAL CONDITION AND LIFETIME EXTENSION ASSESSMENT OF SUNPP-3 REACTOR

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ABSTRACT

This paper presents the results of thermal-hydraulic calculations performed as part of the technical condition and lifetime extension assessment of SUNPP-3 reactor. Based on the recommendations of the IAEA and the results of previously performed computational analyzes, the main conservative assumptions for the analysis of the selected computational scenarios were identified. These assumptions, in the form of initial and boundary conditions, were used in thermal-hydraulic analysis analyzes. When choosing the initial and boundary conditions for each scenario, were used the results of preliminary variant computational analyzes, engineering assessment of the initial and boundary conditions influence to the results, as well as experience in performing similar work. Thermal-hydraulic analyzes were performed using a computer code RELAP5 using a reactor model with a detailed breakdown of the downcomer and engineering code GRS-MIX to account for stratification in reactor downcomer. The results of analyzes showed that the selected initiating events lead to an intensive cooling down of the reactor pressure vessel and the internals. For scenarios leading to asymmetrical cooling, a different configuration of the ECCS was considered. For scenarios leading to a general cooling down of the coolant, the rupture location was chosen in such a way as to obtain maximum cooling down when the ECCS pumps operate in non-emergency loops. As a result of performing thermal-hydraulic analyzes of the initial events were determined boundary conditions for strength analysis performing. These boundary conditions were distribution of the pressure, the coolant temperature and the heat transfer coefficients.

KEYWORDS
Reactor pressure vessel, thermal shock, thermal-hydraulic, lifetime extension, downcomer

1. INTRODUCTION

At the moment, Ukraine continues to extend the life of nuclear power plant (NPP) units. At the same time, a full cycle of analytical justifications is performed. Result of each new calculation for each next power unit, new experience is gained, world experience, requirements and comments of the Regulatory Body (SNRIU), modernization of power units, etc. are taken into account.
Information regarding the lifetime status of Ukrainian NPP with a VVER-1000 reactor is given in the Table I below.

Table I. Information regarding the lifetime of Ukrainian NPP (status on June 10, 2019)

Power Unit	Reactor facility (reactor) type	Commissioning	End of the design lifetime	Lifetime extension
RNPP1	V-213 (VVER-440)	1981	2010	2020
RNPP2	V-213 (VVER-440)	1982	2011	2021
SUNPP1	V-302 (VVER-1000)	1982	2013	2023
ZNPP1	V-320 (VVER-1000)	1984	2015	2025
SUNPP2	V-338 (VVER-1000)	1985	2015	2025
ZNPP2	V-320 (VVER-1000)	1985	2016	2026
RNPP3	V-320 (VVER-1000)	1986	**2017**	2037
ZNPP3	V-320 (VVER-1000)	1986	**2017**	2027
ZNPP4	V-320 (VVER-1000)	1987	**2018**	2028
KHNPP1	V-320 (VVER-1000)	1987	**2018**	-
SUNPP3	V-320 (VVER-1000)	1989	**2020**	-
ZNPP5	V-320 (VVER-1000)	1989	2020	-
ZNPP6	V-320 (VVER-1000)	1995	2026	-
RNPP4	V-320 (VVER-1000)	2004	2035	-
KHNPP2	V-320 (VVER-1000)	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 SUNPP-3. Thermal-hydraulic and probabilistic justification for these unit were approved by the NPP Operator (SE NNEGC “Energoatom”) and the Regulator Body. The article authors took part in performing analyzes for SUNPP-3.

2. METHODOLOGY BASE

The main international document containing the methodology for assessing technical condition of reactor pressure vessel (RPV) under pressure (VVER and PWR types) in the part of performing thermal-hydraulic analysis is Guideline (methodology) IAEA-EBP-WWER-08 [1]. This Guideline was prepared within the framework of International Atomic Energy Agency (IAEA) technical cooperation projects RER/9/035 and the Extrabudgetary Program on Safety of NPPs with VVER and RBMK. The Guideline is the fundamental document for the implementation of this analysis type in the world.

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 [2] was developed. Methodology [2] is basic, in particular, to perform thermal-hydraulic analysis for all power units of Ukrainian NPPs.

Based on the available information, including [1, 2, 3], more detailed typical and work programs are developed for assessing the technical condition and lifetime extension of reactor elements [4, 5 etc.], 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. But, 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.

However, it requires the development of a new methodology or the improvement of the existing methodology to take into account the specifics of Ukrainian NPPs, which is described in more detail in [8].

3. 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 (HTC) from coolant to the RPV 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 [2] and were having been taken as the basis for analysis of the methodologies below.

4. SCENARIOS GROUPING

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 cannot be ignored.

4.1 Deterministic approach

In accordance with [1] and [2], 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.
In accordance with [2] a general list of the initial events specificed 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 [2], initial events that included in typical groups from the analysis of design basis accidents should be considered.
However, for SUNPP-3 is applied the approach given in [1, 3], 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, 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). Groups with inadvertent actuation of the high pressure injection or make-up systems and accidents resulting in cooling of the RPV from outside were excluded from consideration due to the fact that they do not lead to thermal shock because the technical features of the reactor facility. Based on the above, the Ukrainian methodology [2] should be improved in future, as noted in more detail in [8].

4.2 Probabilistic approach

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 [2] it was noted that the use of the probabilistic approach needs a lot of work (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 [3] 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). Based on probabilistic analysis, dominant scenarios with a high probability of realization (frequency more than 1E-08 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 RiskSpectrum PSA is used, and the corresponding integral probability model of SUNPP-3 for the full spectrum of the accident initial events for all reactor facility modes.
Also were taken into account the requirements of the emergency instructions for the accidents liquidation and accidents in the format of symptom-oriented accident instructions that contain procedures for managing processes with the possible occurrence of the RPV thermal shock. As a result, the quantitative assessment of the groups and the frequency of their implementation was carried out. Taking into account the instructions for the accidents liquidation in the format of symptom-oriented accident instructions, it was possible to expand and detail the list of groups, and to identify groups that more accurately describe the reaction of the power unit to the initial events of the accident. These groups were considered qualitatively or quantitatively in thermal-hydraulic analysis.
Lists of 14 dominant scenarios groups for SUNPP-3 with a high probability of realization (contribution more 0.1%, Figure 1) is given below in the Table II below. Profile of dominant scenario groups for SUNPP-3 is shown on Figure 1.

Table II. Lists of 14 dominant scenarios groups with a high probability of realization

№	Name of Scenario Group / reactor facility mode	Description	Frequency, 1 year
1.	PTS-1N-A1 / «on power»)	Large primary leakage DN90-350 with operation 3 train of HPIS (ECCS high-pressure injection) and LPIS (ECCS low-pressure injection) system	8.44E-05
2.	PTS-1S-A1 / «hot shutdown»)		1.51E-05
3.	PTS-2N-1 / «on power»)	Medium primary leakage DN50-90 with operation 3 train of HPIS	5.33E-04
4.	PTS-2S-1 / «hot shutdown»)		1.18E-05

№	Name of Scenario Group / reactor facility mode	Description	Frequency, 1 year
5.	PTS-3N-1 / «on power»)	Small primary leakage DN11-50 with operation 3 train of HPIS	6.86E-03
6.	PTS-3S-1 / «hot shutdown»)		7.58E-05
7.	PTS-3N-4 / «on power»)	Small primary leakage DN11-50 with operation 1 train of HPIS	2.76E-05
8.	PTS-4N-1 / «on power»)	Small primary leakage DN<11 with operation 1 train of HPIS	4.97E-04
9.	PTS-5N-1 / «on power»)	Failure of SG feeding from main and emergency feed-water systems with initiation of 1 SG from deaerators and cooling with maximum speed	1.51E-05
10.	PTS-6N-1 / «on power»)	Small leakage from primary to secondary with failure of: make-up system, emergency SG localization, heat removal through secondary	5.99E-04
11.	PTS-7N-11 / «on power»)	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	2.92E-03
12.	PTS-7S-11 / «hot shutdown»)		3.16E-05
13.	PTS-7N-12 / «on power»)	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 recirculation of redundant HPIS train after the cooldown through secondary	9.73E-04
14.	PTS-7S-12 / «hot shutdown»)		1.05E-05

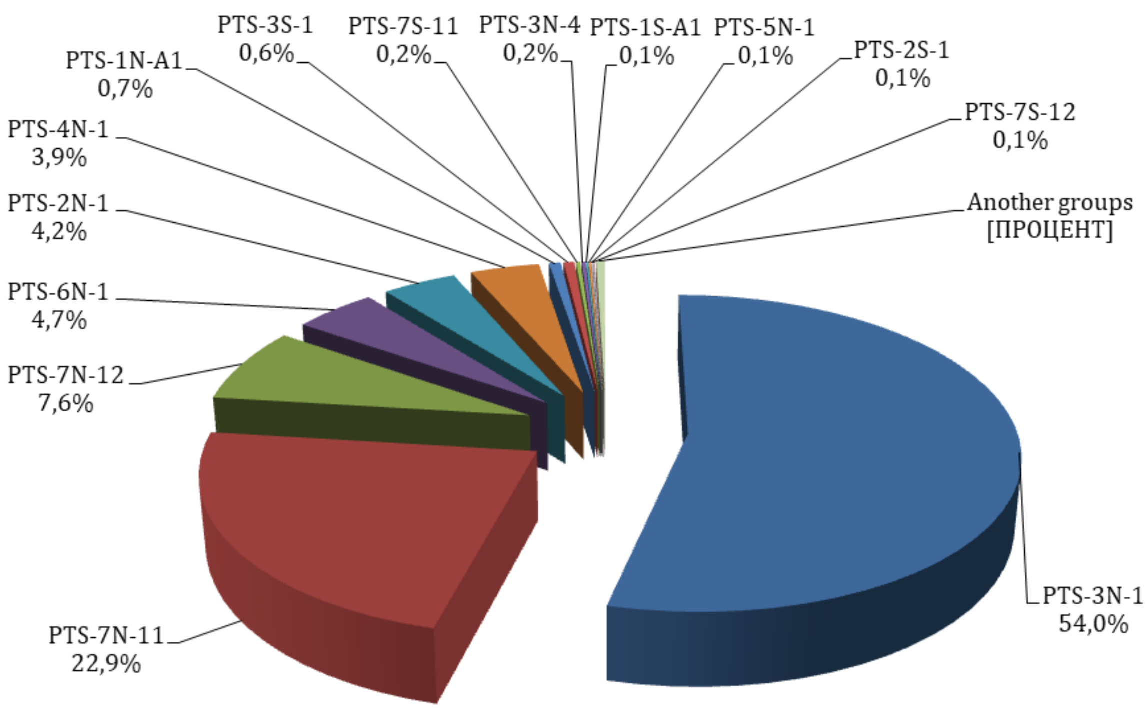


Figure 1. Profile of dominant scenario groups

5. THERMAL HYDRAULIC ANALYSIS

5.1 System analysis

Thermal-hydraulic calculations for SUNPP-3 are performed by using the computer code RELAP5/mod3.2, approved for use by the Ukrainian NPPs Operator to justify the NPPs safety. The calculation code RELAP5 is widely used to make justifications for the NPPs safe operation in Ukraine as a whole and to obtain boundary conditions for performing strength calculations. At the same time, it should be noted that the code RELAP5 has limitations in terms of correct coolant stratification modeling in the main coolant pipes and formation of cold coolant stripes and tongues in the reactor downcomer, therefore there was a need to perform mixing analysis that also is required in [1].

The calculation model was updated for the current time to take into account the current state of the power unit and the performed modernizations (in frame of preparation to the lifetime extension). Also, a breakdown of the reactor downcomer was performed, conservative conditions of cooling water temperature were modelled, etc. Detailed information on modeling and validation is given in [9, 10]. Since in 2018, the core of SUNPP-3 reactor was fully loaded with fuel assembles produced by company Westinghouse, therefore, the hydraulic and neutron-physical part of the model was refined (in previous models, fuel assembles manufactured by the Russian company TVEL was modeled).

As a result, full-scale calculation of the 54 selected scenarios was performed. The boundary conditions after RELAP5-calculations were analyzed by linear mechanics methods and 6 dominant scenarios for cylindrical part of the reactor were defined. These dominant scenarios characterized by the smallest margin to the brittle strength. Since the code RELAP5 is integral and does not allow modelling of the coolant stratification, then for these scenarios a mixing analysis in downcomer was performed. In addition, representative scenarios were chosen for reactor nozzle zone, but are not presented in this article.

5.2 Mixing analysis

Mixing analysis was performed using engineering code GRS-MIX. Code GRS-MIX was developed by Gesellschaft fur Anlagen-und Reaktorsicherheit (GRS) and represents a set of correlations and equations based on experimental data obtained from the UPTF-TRAM experimental facility. The code allows to determine the physical parameters necessary for the strength analysis of dominants scenarios above (boundary conditions).

Using the code GRS-MIX, the parameters near the inner surface of the RPV were refined in the areas that are more interesting for research. These were areas with the formation of cold stripes and tongues. In this case, for code GRS-MIX a smaller grid was used than for the code RELAP. To demonstrate this, the Figure 2 shows both grids for codes RELAP and GRS-MIX.

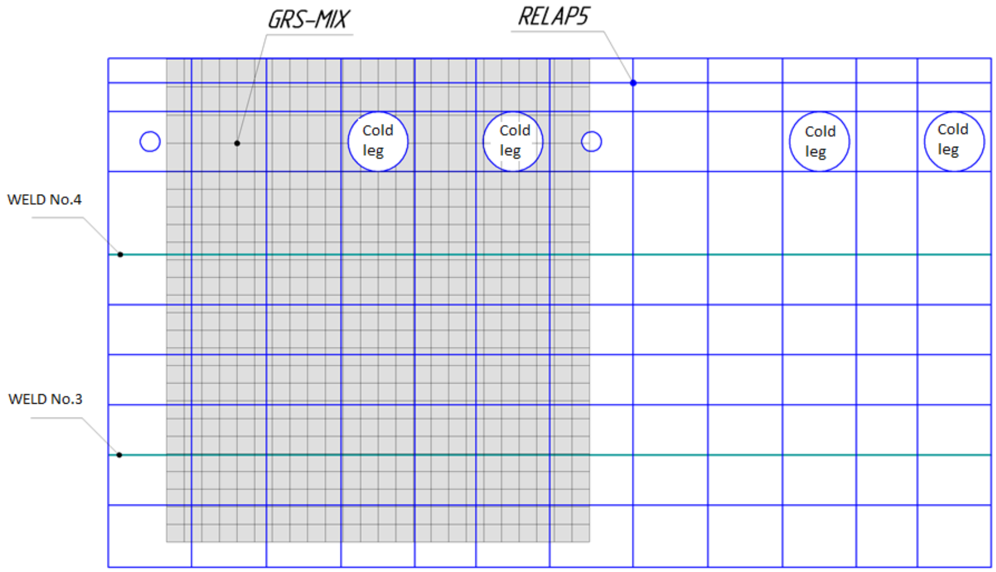


Figure 2. Grids for codes RELAP and GRS-MIX

Special CFD-codes were not applied.

5.3 Analysis results

Below are the results of computational analysis using the integral code RELAP5 and the mixing code GRS-MIX for the dominant scenarios. The figures 3-26 below show graphs of changes in pressure and minimum temperature in the reactor downcomer, as well as temperature and heat transfer coefficient at the level of weld No.4. The weld No.4 is located opposite the reactor core and is maximally loaded due to fluence. The comparision graphs show the time interval, which is characterized by flow stratification during ECCS pump operation.
In all scenarios are modelled main conservative boundary conditions such as different ECCS configuration, personnel actions for obtaining worst conditions, minimal temperature of ECCS water, maximum capacity of ECCS pumps, blackout etc.
As you can see, the mixing analysis made it possible to take into account the stratification of the flow in the reactor downcomer and to prepare more realistic and accurate boundary conditions (parameters in downcomer) for the strength assessment of the brittle strength of the RPV. In addition, it meets to the IAEA requirements and to international practice.

5.3.1 Primary coolant leak with an equivalent diameter of 32 mm with ECCS maximum configuration in the “hot shutdown” mode

Leak DN 32 from a cold leg No. 2 with full blackout is formed on 0 s. Pumps TQ13,23,33D01 begin to feed primary on 268 s after the primary pressure decreases to pumps operating conditions (less ~110 kgf/cm^2). The pressurizer is emptied on 380 s. The operator starts pumps TQ14,24,34D01 on 900 s manually. The feeding of boric acid solution from pumps TQ14,24,34D01 is stopped due to tanks emptying on 8540 s. After that, as shown in Figures 3 and 4, parameters in the downcomer are stabilized with combinations of high pressure (~103 kgf/cm^2) and low temperature (~21 °C) that are the thermal shock conditions. In Figures 5 and 6 is showed comparison of temperatures and heat transfer coefficients near weld No. 4 that were received in codes RELAP and GRS-MIX.

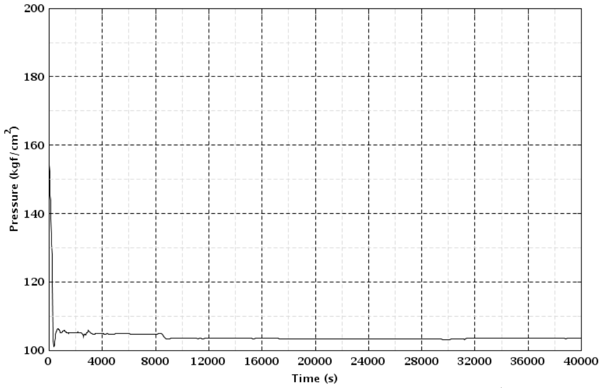


Figure 3. Coolant pressure in reactor (RELAP)

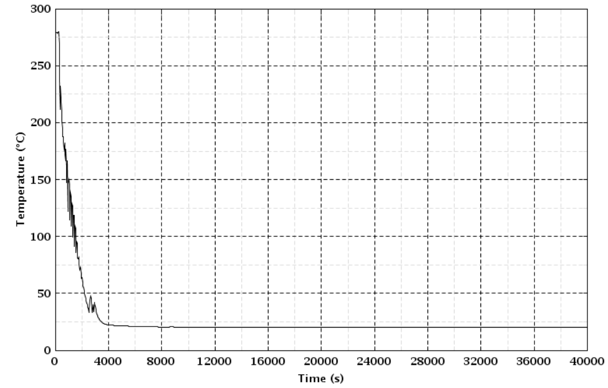


Figure 4. Minimum coolant temperature in reactor downcomer (RELAP)

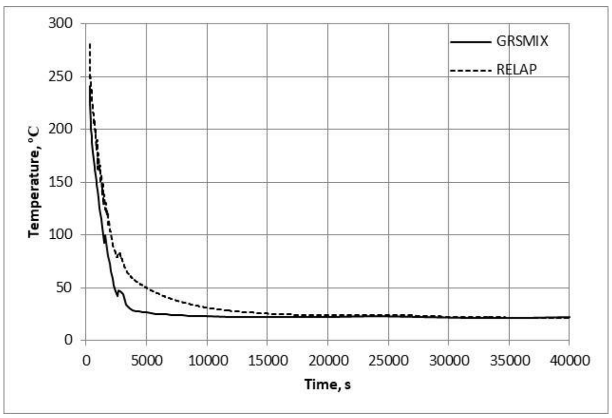


Figure 5. Coolant temperature near weld No. 4

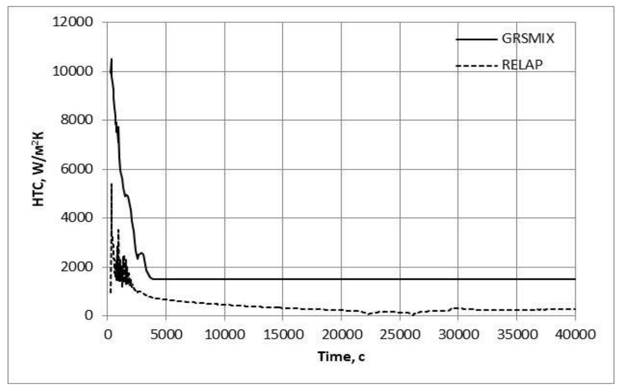


Figure 6. Heat transfer coefficient near weld No. 4

5.3.2 Rupture of the HPI injection pipeline DN125 with the ECCS minimum configuration in power mode

Rupture of the HPI injection pipeline DN125 on cold leg No. 3 is formed on 0 s. Pump TQ23D01 begin to feed primary on 10 s after the primary pressure decreases to pump operating conditions (less ~110 kgf/cm^2). The feeding of boric acid solution from the ECCS passive hydro accumulators №3,4 begins on 90 s when the pressure in the primary circuit drops to the operating conditions (less ~60 kgf/cm^2). The pressurizer is emptied on 310 s. The operator starts pump TQ24D01 on 900 s manually. Pump TQ22D01 begin to feed primary on 940 s after the primary pressure decreases to pump operating conditions (less ~23 kgf/cm^2). The feeding of boric acid solution from pump TQ24D01 is stopped due to tanks emptying on 8540 s. Fast temperature drop on first accident phase (~0-3000 s) lead to thermal shock. After ~3000 s, as shown in Figures 7 and 8, parameters in the downcomer are stabilized with combinations of pressure (~4 kgf/cm^2) and low temperature (~21 °C). In Figures 9 and 10 is showed comparison of temperatures and heat transfer coefficients near weld No. 4 that were received in codes RELAP and GRS-MIX.

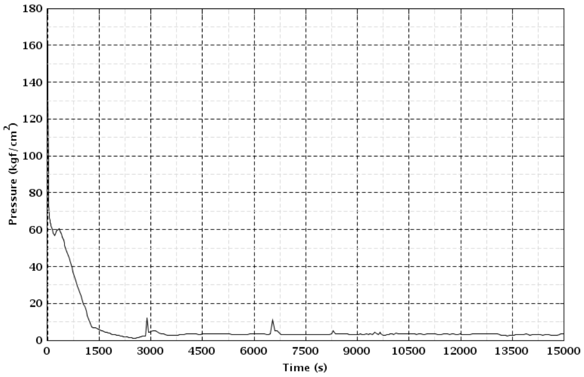


Figure 7. Coolant pressure in reactor (RELAP)

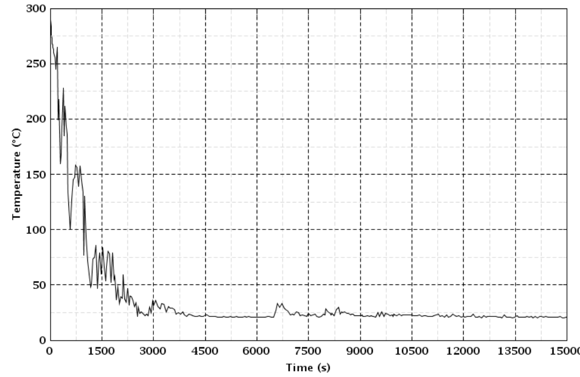


Figure 8. Minimum coolant temperature in reactor downcomer (RELAP)

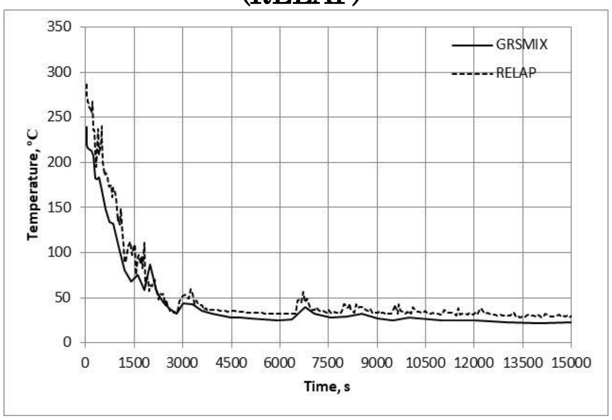


Figure 9. Coolant temperature near weld No. 4

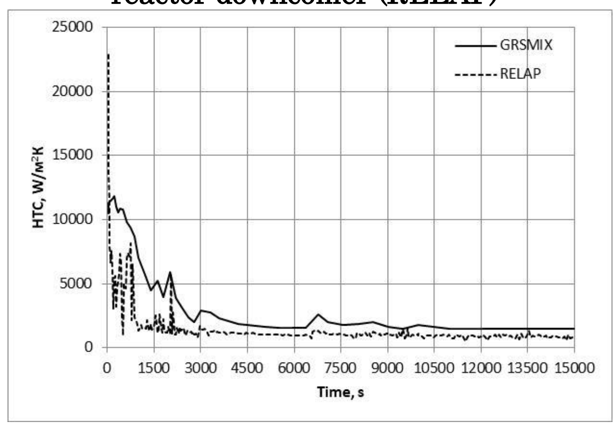


Figure 10. Heat transfer coefficient near weld No. 4

5.3.3 Unintentional opening of fast-acting steam dump valve (BRU-K) with the ECCS minimum configuration in the "hot shutdown" mode

Unintentional opening of fast-acting steam dump valve (BRU-K) with full blackout is formed on 0 s. Main steam isolation valves (MSIV) are closed on 97 s (is modelled failure to close the valve on the steam line №1, conservatively). Pump TQ14D01 begin to feed primary on 97 s after the signal of technological protection. After reducing the level in SG-1 to 1.5 m and 1.35 m emergency feed water pumps (EFWPs) №1,2 begin to feed SGs on 460 and 480 s respectively. Pump TQ13D01 begin to feed primary on 1530 s after the primary pressure decreases to pump operating conditions (less ~110 kgf/cm^2). But the primary pressure increases and the pump TQ13D01 reduce mass flow and therefore operator starts the make-up system pumps on 2060 s to restore the level in pressurizer on 5400 s. The feeding of boric acid solution from pump TQ14D01 is stopped due to tanks emptying on 7700 s. The primary coolant temperature is decreased as a result of the rapid heat sink increase through SG-1. The entry of cold coolant into the sector of downcomer causes an uneven temperature difference between the downcomer sectors with the cold "sector" formation. To end of calculation parameters in the downcomer are characterized by combinations of high pressure (~171 kgf/cm^2) and temperature (~90 °C) that are the thermal shock conditions (Figures 11 and 12). In Figures 13 and 14 is showed temperatures and heat transfer coefficients near weld No. 4 that were received in codes RELAP (code GRS-MIX is not used as stratification conditions in loops is absent).

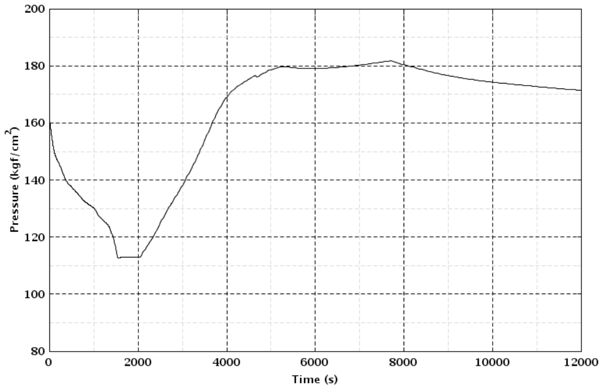


Figure 11. Coolant pressure in reactor (RELAP)

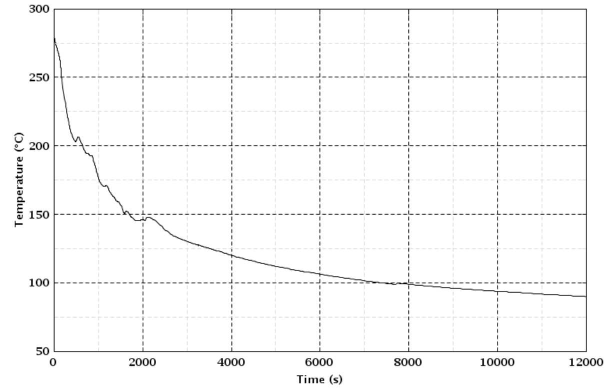


Figure 12. Minimum coolant temperature in reactor downcomer (RELAP)

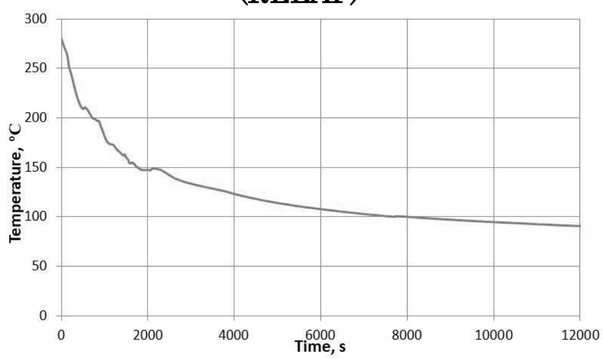


Figure 13. Coolant temperature near weld No. 4

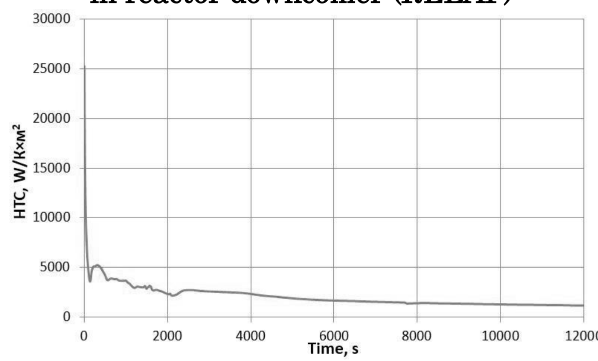


Figure 14. Heat transfer coefficient near weld No. 4

5.3.4 Rupture of three heat exchange tubes with the ECCS maximum configuration in the “hot shutdown” mode

Rupture of three heat exchange tubes of SG-4 with full blackout is formed on 0 s. On 310 s the pressure in the SG-4 increase to the settings (~73 kgf/cm^2) for opening of fast-acting steam dump valve (BRU-A). Pumps TQ13,23,33D01 begin to feed primary on 490 s after the primary pressure decreases to pumps operating conditions (less ~110 kgf/cm^2). The operator starts pumps TQ14,24,34D01, turns on the non-emergency BRU-A in the cooling mode (speed 60 °C/h), combines all emergency gas removal line on 900 s manually. On 1060 s the pressure in the SG-4 increase to the settings (~86 kgf/cm^2) for periodic opening of SG-4 safety valve. MSIV of non-emergency SGs are closed during 3360-3440 s. When the certain conditions are achieved, the pressurizer safety valves (SV) begin to operate in the mode of protection against cold overpressure for primary pressure decreasing. The feeding of boric acid solution from the ECCS passive hydro accumulators begins on 9680 s when the pressure in the primary circuit drops to the operating conditions (less ~60 kgf/cm^2). Pump TQ12,22,32D01 begin to feed primary on 9850 s after the primary pressure decreases to pump operating conditions (less ~23 kgf/cm^2). Fast temperature drop on first accident phase with high pressure (before ~9850 s) lead to thermal shock. After ~9850 s, as shown in Figures 15 and 16, parameters in the downcomer are stabilized with combinations of pressure (~30 kgf/cm^2) and low temperature (~20 °C). In Figures 17 and 18 is showed comparison of temperatures and heat transfer coefficients near weld No. 4 that were received in codes RELAP and GRS-MIX.

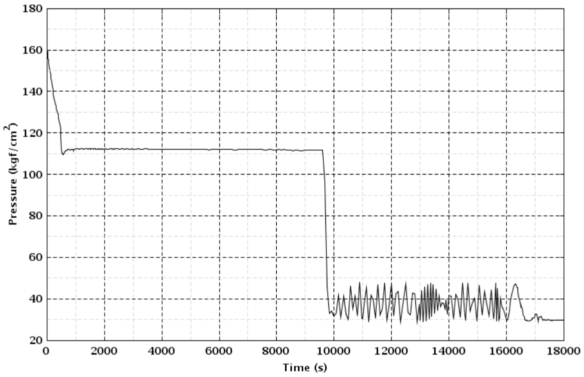


Figure 15. Coolant pressure in reactor (RELAP)

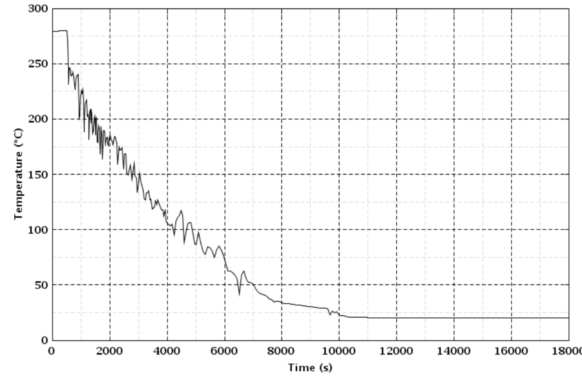


Figure 16. Minimum coolant temperature in reactor downcomer (RELAP)

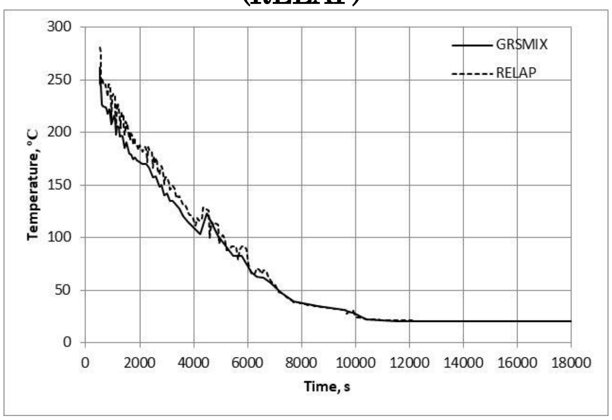


Figure 17. Coolant temperature near weld No. 4

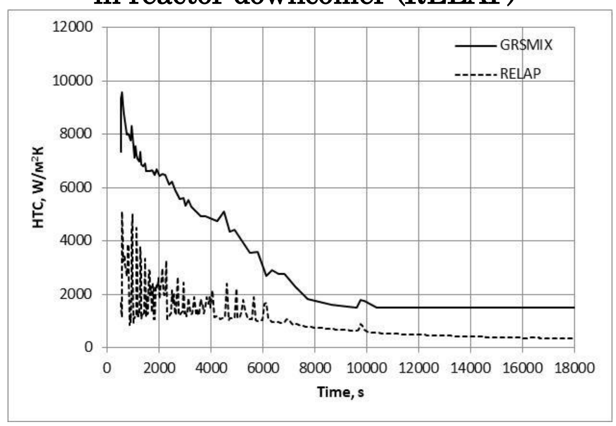
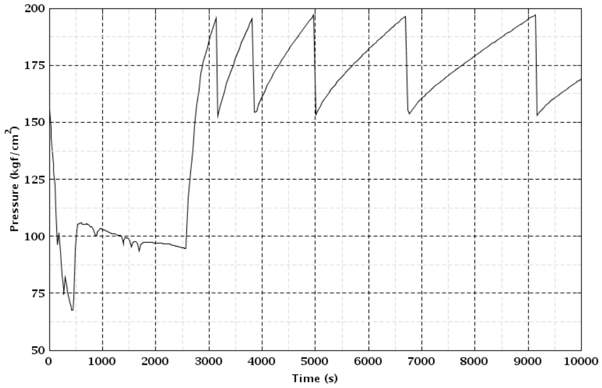


Figure 18. Heat transfer coefficient near weld No. 4

5.3.5 Unintentional opening of the pressurizer SV and closing after 2570 s with the ECCS maximum configuration in the "hot shutdown" mode in the "hot shutdown" mode

Unintentional opening of the pressurizer SV with full blackout is formed on 0 s. Pumps TQ13,23,33D01 begin to feed primary on 101 s after the primary pressure decreases to pump operating conditions (less ~110 kgf/cm^2). The operator starts TQ14,24,34D01 pumps on 900 s manually. The operator closes opened pressurizer SV on 2570 s manually (before conditions for SV operating in the mode of protection against cold overpressure), that leads to a rapid primary pressure increase. As a result, pumps TQ13,23,33D01 is stopped (2610 s). The feeding of boric acid solution from pump TQ14,24,34D01 is stopped due to operator action accordance to emergency procedures from 2670 to 2870 s. On 3130 s the primary pressure increase to the settings (~194 kgf/cm^2) for periodic opening of pressurizer SVs. As shown in Figures 19 and 20, combinations of high pressure (more 150 kgf/cm^2) and low temperature (from ~25 to ~100 °C) after pressurizer SV closing is characterized as thermal shock conditions in the downcomer. In Figures 21 and 22 is showed comparison of temperatures and heat transfer coefficients near weld No. 4 that were received in codes RELAP and GRS-MIX.



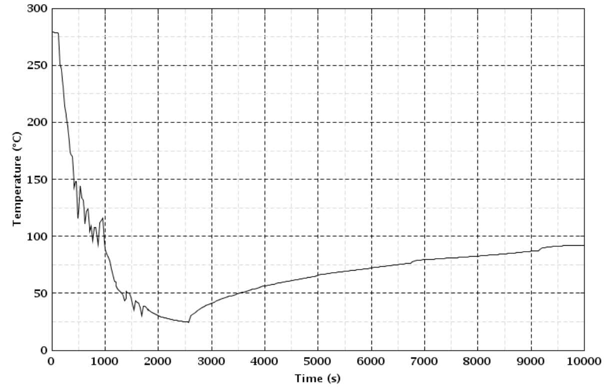
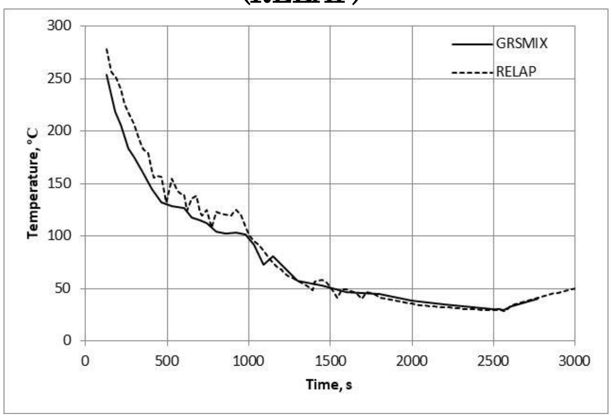


Figure 19. Coolant pressure in reactor (RELAP)

Figure 20. Minimum coolant temperature in reactor downcomer (RELAP)



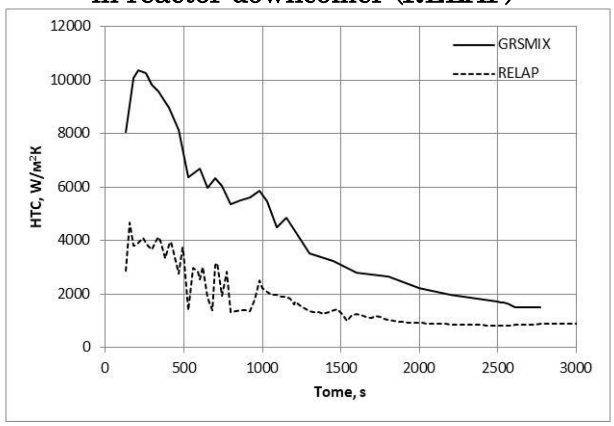


Figure 21. Coolant temperature near weld No. 4

Figure 22. Heat transfer coefficient near weld No. 4

5.3.6 Unintentional opening of the pressurizer SV and closing after 5800 s with the ECCS minimum configuration in the "hot shutdown" mode in the "hot shutdown" mode

Unintentional opening of the pressurizer SV with full blackout is formed on 0 s. Pump TQ13D01 begin to feed primary on 101 s after the primary pressure decreases to pump operating conditions (less ~110 kgf/cm^2). The operator starts pump TQ14D01 on 900 s manually. The feeding of boric acid solution from the ECCS passive hydro accumulators №1,2 begins on 3280 s when the pressure in the primary circuit drops to the operating conditions (less ~60 kgf/cm^2). The operator closes opened pressurizer SV on 5800 s manually (before conditions for SV operating in the mode of protection against cold overpressure), that leads to a rapid primary pressure increase. As a result, the feed from hydro accumulators (5960 s) and pump TQ13D01 (6920 s) is stopped. On 8080 s the primary pressure increase to the settings (~194 kgf/cm^2) for periodic opening of pressurizer SVs. The feeding of boric acid solution from pump TQ14D01 is stopped due to tanks emptying on 8540 s. As shown in Figures 23 and 24, combinations of high pressure (more 150 kgf/cm^2) and low temperature (from ~25 to ~50 °C) after pressurizer SV closing is characterized as thermal shock conditions in the downcomer. In Figures 25 and 26 is showed comparison of temperatures and heat transfer coefficients near weld No. 4 that were received in codes RELAP and GRS-MIX.

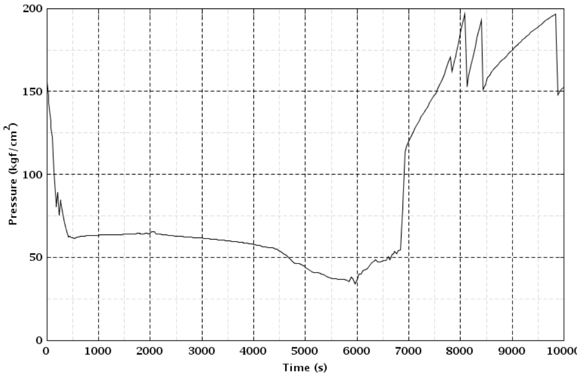


Figure 23. Coolant pressure in reactor (RELAP)

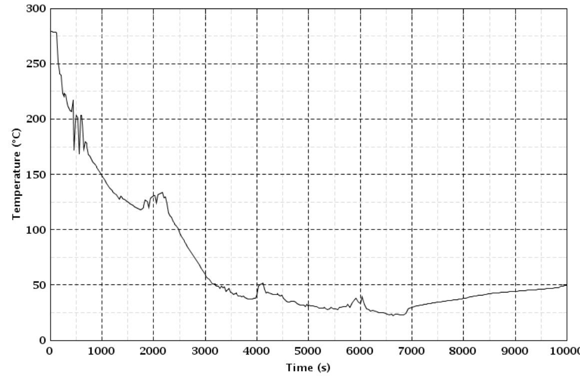


Figure 24. Minimum coolant temperature in reactor downcomer (RELAP)

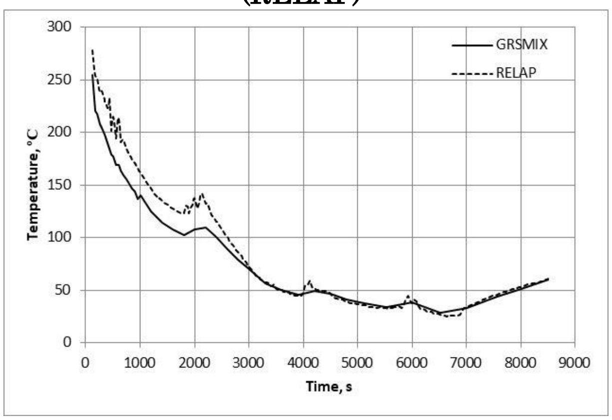


Figure 25. Coolant temperature near weld No. 4

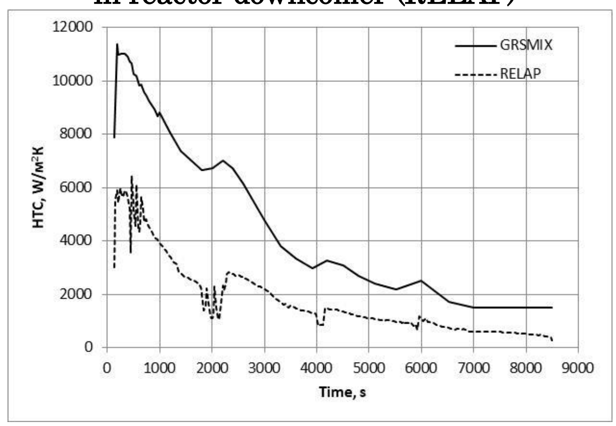


Figure 26. Heat transfer coefficient near weld No. 4

6. CONCLUSIONS

The article demonstrates the results of all stages of the analysis, which are described in more detail in the extensive technical reports approved by the NPP operator and the Regulatory Body.

Thermal-hydraulic analysis for SUNPP-3 reactor lifetime extension was carried out taking into account the requirements of the IAEA guidelines and international experience, which are considered in the typical and working programs that are special for one NPP unit. New domestic methodology should be developed or the existing Ukrainian methodology should be improved.

As presented in article due to probabilistic analysis, preliminary qualitative thermal-hydraulic analysis and variant thermal-hydraulic calculations, 54 scenarios were selected. The boundary conditions after RELAP5-calculations were analyzed by linear mechanics methods and 6 dominant scenarios for cylindrical part of the reactor were defined. Since the code RELAP5 is integral and does not allow modelling of the coolant stratification, then for these scenarios a mixing analysis in downcomer using code GRS-MIX was performed. The mixing analysis made it possible to take into account the stratification of the flow in the reactor downcomer and to prepare more realistic and accurate boundary conditions (parameters in downcomer) for the strength assessment of the brittle strength of the RPV.

For each scenario, the chronology of events and key parameters are shown to demonstrate that they lead to thermal shock conditions.

NOMENCLATURE

TQ12,22,32D01 – pumps of first, second and third ECCS LPI trains;

TQ22 – pump of second ECCS LPI train;
TQ13,23,33D01 – pumps of first, second and third ECCS HPI trains;
TQ13D01 – pump of first ECCS HPI train;
TQ23D01 – pump of second ECCS HPI train;
TQ14,24,34D01 – pumps of first, second and third ECCS HPI trains (emergency injection part);
TQ14D01 – pump of first ECCS HPI train (emergency injection part);
TQ24D01 – pump of second ECCS HPI train (emergency injection part).

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