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Regulatory Experience in Licensing of Alternative Supplier Fuel

At present, the diversification of nuclear fuel is ongoing at Ukrainian NPPs. The Ukrainian regulatory authority and SSTC NRS, as a technical support organization, take active part in the licensing process for introduction of new fuel assembly types. In the framework of this activity, the technical review of justification material is performed in various nuclear safety issues including a wide range of verifying calculations. Neutron kinetic, thermal-hydraulic and fuel thermal-mechanical models are developed for technical review of justification material such as safety analysis of mixed cores. The features of the developed models for safety analysis of new fuel implementation and results of their application for verifying calculations are presented in this paper. The results cover such aspects as preparation of few-group cross-section libraries with features of new FA types, calculations of transitional mixed fuel loadings up to stationary ones, verifying safety assessment for some design-basis reactivity-initiated accidents, criticality of fuel management systems and thermal-mechanical fuel behavior.

Keywords: nuclear fuel, WWER, reactor core, mixed cores

The need for search of alternative nuclear fuel suppliers for the Ukrainian nuclear power plants is strongly associated with the economic and energy independence. Diversification of fresh nuclear fuel supply is one of the highest priorities for the most complete and optimal use of installed power of Ukrainian nuclear power plants. On the one hand, it positively influences the growth of fuel efficiency, as it meets customer requirements on the compatibility of nuclear fuel and the competing suppliers (manufacturers) are interested in improving the quality and performance of their fuel. On the other hand, competition between suppliers (manufacturers) promotes the establishment of a market (grounded) price for the supply of fuel [1], first at all, due to the competition between fuel suppliers. In addition, diversification of nuclear fuel supply complies with provisions of the European Energy Security Strategy. It says "...an overall diversified portfolio of fuel supply is needed for all plant operators..."

1. Diversification of Nuclear Fuel Supply at Ukrainian NPPs

Implementation of nuclear fuel supply diversification for Ukrainian NPPs started soon after signing the Agreement between the Government of the United States of America and the Government of Ukraine concerning the Ukraine Nuclear Fuel Qualification Project on 6 June 1998. The Westinghouse Company won the tender for nuclear fuel supply and technology of its design in Ukraine. An important role in the tender results was that the Westinghouse Company had already designed the fuel for WWER-1000 reactor at the Temelin NPP (Czech Republic) before the tender. The choice of Westinghouse as a potential alternative supplier of nuclear fuel for Ukrainian NPPs best meets the diversification objectives.

The Ukraine Nuclear Fuel Qualification Project provided development and introduction of alternative nuclear fuel, which would be compatible with the TVEL design fuel under operation of mixed cores. The South Ukraine Nuclear Power Plant (SUNPP) Unit 3 was chosen for test operation of Westinghouse nuclear fuel.

The project was implemented at SUNPP Unit 3 in two main phases [2]:

- six Lead Test Assemblies (LTA) were manufactured, supplied and operated at SUNPP Unit 3;
- for the second phase, a reload batch of 42 Westinghouse fuel assemblies (hereinafter WFAs) was produced and loaded for WFA test operation at SUNPP Unit 3.

During the project implementation also:

- the compatibility of Russian TVEL and Westinghouse design fuel assemblies was justified;
- LTAs were licensed and the Ukrainian Regulator Authority issued permission for their trial operation;
- six LTAs successfully worked out four fuel campaigns envisaged by the project at SUNPP Unit 3 (17th-20th fuel campaigns from 2005 to 2010). Moreover, as part of the mixed core loading of the last 20th campaign in addition to TVEL FAs of TVS-M design and Westinghouse LTAs, 42 other TVEL FAs of TVSA design were in operation. These latter FAs are characterized by high lateral stiffness due to the inclusion of rail corners in the frame design. Average burnup of unloaded LTAs was ~43.56 GW*d/tU, while the maximum values reached 46.00 GW*d/tU;

- based on the positive results of the trial operation of six LTAs, the first reload batch of 42 WFAs was loaded for the 21th fuel campaign of SUNPP Unit 3, and WFAs were operated in surrounding of 79 TVS-M and 42 TVSA FAs during the 21th fuel campaign.

The design of the first reload batch of 42 WFAs is similar to LTAs. However, the FA design improvement and transfer of FA production from the plant in Columbia (South Carolina, USA) to plant in Vesteras (Sweden) led to some changes in the design. The main differences are as follows [4]:

- increased rigidity of the frame by mounting of medium and upper spacer grids to guide channels;
- use of gadolinium oxide (Gd_2O_3) as an integrated burnable absorber instead of the zirconium diboride (ZrB_2) thin cover layer on fuel pellets used earlier in the LTA design;
- use of zirconium alloy (Zr-1 %Nb) for all middle spacer grids instead of alloy 718 because of low neutron capture cross-section and high corrosion resistance.

Based on the positive results of the trial operation of six LTAs in the first reload batch of 42 WFAs at SUNPP Unit 3, the Energoatom Nuclear Operator expanded the use of WFAs to SUNPP Unit 2.

The second reload batch of 42 Westinghouse fuel was loaded for the 22nd fuel campaign of SUNPP Unit 3 and the total number of WFAs in the core of this unit reached 84. Moreover, the first reload batch of 42 WFAs was loaded for the 24th fuel campaign of SUNPP Unit 2. The subsequent operation of WFAs in these campaigns at SUNPP Unit 2 and Unit 3 was held without any problems.

However, some deformation of spacer grid outer straps was revealed in 2012 during core refueling of SUNPP Unit 3 after operation of the 22th fuel campaign. The systematic nature of the detected deformation was confirmed by inspection of the reload batch of 42 WFAs during scheduled core refueling at SUNPP Unit 2 in June 2012. The study of the deformation causes concluded the necessity for strengthening the WFA frames. Further expansion and operation of the WFAs delivered in 2012 and 2013 were suspended.

The next WFA modification (RWFA) was developed by Westinghouse to prevent the deformation of spacer grid outer straps. Eight spacer grids with a changed outer plate profile and thickness increased up to 0.813 mm were returned. The spacer grid material was changed from zirconium alloy to stainless steel. In addition, the design of top and bottom nozzles was changed [3]. The reload batch of improved 42 RWFA was loaded for the 25th fuel campaign of Unit 3 in August 2014.

Therefore, the reactor core of SUNPP Unit 3 during the 25th fuel campaign was loaded simultaneously with three different FA types (WFA and RWFA of Westinghouse design and TVSA of TVEL design). In addition, in the summer of 2018, the reactor core of SUNPP Unit 3 was fully loaded with FAs of Westinghouse design.

During this time, the Operator decided to extent trial operation to some other units. At other units of Ukrainian NPPs, the distribution of loaded diversified FAs looks as follows at present time:

- SUNPP Unit 2 during the 29th fuel campaign was loaded simultaneously with two different FA types (2/4 core — RWFA of Westinghouse design);
- ZNPP Unit 1 during the 29th fuel campaign was loaded simultaneously with two different FA types (1/4 core — RWFA of Westinghouse design);
- ZNPP Unit 3 during the 30th fuel campaign was loaded simultaneously with two different FA types (1/4 core — RWFA of Westinghouse design);
- ZNPP Unit 4 during the 30th fuel campaign was loaded simultaneously with two different FA types (1/4 core — RWFA of Westinghouse design);

- ZNPP Unit 5 during the 29th fuel campaign was loaded simultaneously with two different FA types (3/4 core — RWFA of Westinghouse design);

Moreover, the Operator of Ukrainian NPPs has plans to extend Westinghouse FA supplies also to Rivne NPP Unit 3.

2. Ukrainian Regulatory Authority's Approach to Independent Verifying Calculations

The State Scientific and Technical Center for Nuclear and Radiation Safety (SSTC NRS), as a technical support organization of the Ukrainian Regulatory Authority, is involved in the licensing process for the introduction of new fuel types at Ukrainian NPPs. SSTC NRS' policy of technical reviews for justification documents includes independent verifying calculations for as more as possible nuclear safety aspects in the introduction of new fuel types. SSTC NRS confirms its decisions by quantitative assessment of neutron kinetic and thermal hydraulic, operational, and radiation safety aspects, strength and design reliability issues etc. For this purpose, SSTC NRS is provided with powerful computing capabilities including a series of codes and models. For neutron kinetic/thermal hydraulic and fuel thermomechanical verifying calculations, the following codes are used [5]:

- SCALE, MCNP — criticality problems, spent fuel;
- HELIOS — preparation of few-group cross-section libraries;
- DYN3D — core safety, transients, accidents;
- TRACE, RELAP, ATHLET — system thermal hydraulic codes;
- TRANSURANUS — fuel pin behavior;
- DOORS — flux/fluence on pressure vessel.

The established licensing practice for the introduction of new fuel types at Ukrainian NPPs envisages the following independent verifying calculations:

- neutron multiplying properties and few-group cross-section library preparation;
- neutron kinetic characteristics and characteristics of transitional and stationary loading;
- thermal hydraulic reliability of fuel pins in normal operation and in accidents;
- criticality of fuel storage and transport systems;
- estimation of thermomechanical behavior of fuel pins;
- estimation of effect from the introduction of new fuel types on neutron fluence at reactor pressure vessel.

In the development of computer models, the features of mixed cores should be taken into account in each area of analysis.

3. Development of Computer Models for Safety Analysis of New Fuel Implementation

3.1. Few-Group Cross-Section Library Preparation

In the context of few-group cross-section library preparation, the following differences between all used fuel assembly types must be taken into account:

- differences in fuel pellet geometry — absence/presence of central hole, outer diameter, etc., and related issues such as choice of fuel temperature for depletion calculation;
- differences in structural material and its geometry (spacer grid, guide tube, strengthened corner for TVSA fuel assembly);
- radial profiling of fuel with regard to enrichment, burnable absorber, type of burnable absorber (integrated or covered).

For this purpose, the 2D fuel assembly model for the HELIOS code with highly detailed structural elements

Fig. 1 — Difference between multiplication factors calculated by HELIOS and PHOENIX for five RWFA assembly types

is used. The developed models allow considering all the above-mentioned features for each fuel assembly type and preparing a highly accurate two-group cross-section library.

In the framework of technical review of RWFA introduction at SUNPP Unit 3, a few-group cross-section library was prepared for 5 RWFA types (beside the libraries that had been already prepared for WFA and TVSA types) with different ^{235}U enrichment and radial profiling with account of all the above-mentioned features.

Comparative analysis of RWFA multiplication properties in burnup between the calculations performed with the HELIOS models and PHOENIX models (presented in safety justification materials) shows good agreement. The maximum difference between the results does not exceed $\Delta K_{inf} = 250$ pcm (Fig. 1).

Taking into account additional heterogeneity of mixed cores in view of neutron kinetic aspects, such advanced features were included into the few-group cross-section library as assembly discontinuity factors and sub-library for accounting the spectral history effect with the methodology using Pu-239 as history indicator [6].

In the framework of few-group cross-section library preparation, the effect of corners at neighboring FAs on pin power in the peripheral row was studied as well. The results of these studies were used for assessment of margin sufficiency in pin power distributions during burnup of mixed cores [7].

3.2. Core Modelling

The design differences between the spacer grids and top and bottom nozzles resulted in different hydraulic resistance coefficients for each type of FAs. The difference between the total hydraulic resistance coefficient for RWFA and TVSA fuel assemblies amounts to ± 36 %. It caused redistribution of the coolant flow rate in the mixed core, which is needed to take into account in analysis of core kinetic characteristics and fuel pin cooling.

SSTC NRS uses the DYN3D nodal code for neutron kinetic/thermal hydraulic process simulation in the reactor core. It is used for assessment of core kinetic characteristics in steady states, Xe transients and fast transients and for pin power calculations and for core burnup calculations.

Fig. 2. Pin power peaking factor Fq and limits for FA types of the 25th fuel loading

Owing to the DYN3D flexible capabilities, the computer model for the WWER-1000 mixed core allows the following features to be taken into account:

- individual hydraulic resistance coefficients for each type of fuel assembly;
- individual thermal physical properties (thermal capacity and conductivity) of pin materials (fuel and cladding) in the heat transfer model for each type of fuel assembly;
- different uranium masses for each type of fuel assembly;
- irregular axial meshing of the core model caused by axial profiling (due to the use of blankets in WFA and RWFA designs);
- differences in fuel pin geometry — inner/outer diameter of fuel and cladding, gas gap width between them for each type of fuel assembly.

With use of the developed DYN3D model, the reactor core kinetic characteristics for SUNPP Unit 3 were calculated for the 25th up to the stationary 36th fuel campaigns. The 25th, 26th and 27th campaigns were mixed cores with different FA types. The goal of the performed calculations is to confirm the correctness of the neutron kinetic characteristics presented in safety justification materials and confirm the compliance with limits that are defined by the Ukrainian regulation “Fuel Handling. Refueling in WWER-1000 Reactor. The Nomenclature of Operational Neutronic Calculations and Experiments” [8]. The analysis of mixed core characteristics placed greater focus on compliance with the following parameters:

- peaking factors, linear pin power;
- assembly and pin averaged burnups;
- reactivity coefficients;
- working group of CRs, scram efficiency, etc.

Taking into account the differences in hydraulic resistance coefficients of each type of FAs, some limits were established stricter than in the regulation [8], such as pin power peaking factor Fq. Moreover, these factors are changed for each next fuel campaign as a function of the number of the operating assemblies of each type in the mixed core. Change of the calculated pin power peaking factor Fq and limits for the first transitional (25th) fuel loading with three different FA types are presented in Fig. 2. For further transitional fuel loads, the difference between power peaking factor Fq limits for different FA types is decreased.

3.3. Accident Analysis

At the present time, the best estimate computer codes combined with conservative initial and boundary conditions (combined analysis) are used for DBA analysis within SAR in Ukraine. Regarding RIA, an approach has been developed to implement all significant conservative initial and boundary conditions into the realistic model of the reactor core. The conservative values of such parameters as

- reactivity coefficients,
- efficiency of CRs and scram weight,
- characteristics of most loaded fuel pin,
- and thermal hydraulic characteristics

are introduced in the developed models for the purpose of DBA analysis. Taking into account this approach for independent verifying calculations, the conservative initial and boundary conditions were chosen so as to cover all FA type limit requirements. Consequently, this caused more conservative results than the ones presented in safety justification materials. Thus, for the most representative initial event of RIA group, control rod ejection, the maximal values of fuel and cladding temperature determined in the verifying

Fig. 3. Maximal fuel temperature in control rod ejection by verifying calculation

Fig. 4. Maximal cladding temperature in control rod ejection (outer surface temperature) by verifying calculation

calculation are significantly higher than the ones in justification material (Fig. 3, Fig. 4). For example, the maximal cladding temperature in verifying calculation is greater even for outer cladding surface temperature due to greater conservatism for gas gap conductivity. The discrepancies in the results are caused by differences of the computational model, leading to different characteristics of initial states. In addition, there is a slight difference in the conservative assumptions in the verifying and justification models. The most valuable include different axial power profiles in initial state and thermal conductivity of the gas gap between fuel and cladding. Nevertheless, the acceptance criteria are fulfilled even under such conservative initial and boundary conditions.

3.4. Thermomechanical Behavior

The verifying calculations for analysis of thermomechanical fuel behavior in RWFA introduction were performed both for normal operation and for accidents. The history of linear power change in each fuel pin for all transitional fuel loads was taken into account. Additional engineering factor $K_{eng} = 1.2$ was applied to pin linear power. The main attention was paid to the pins with the maximal linear power and with the maximal burnup. The following parameters and characteristics were analyzed during verifying calculations:

- maximal fuel pellet temperature;
- maximal cladding outer surface temperature;
- radiation growth of fuel pins;
- outer cladding radius of fuel pins;
- inner gas pressure;
- fuel enthalpy;
- tangential stress in cladding;
- axial deformation of fuel and cladding;
- oxide layer thickness in cladding outer surface of fuel pins.

This set of parameters allows making conclusions on the correctness of the results presented in justification materials. The results of verifying calculation showed only slight divergences from the results in justification materials: for example, ≈ 0.5 MPa in gas pressure into the pin or ≈ 100 eC in fuel pellet temperature or normal operation modes. These divergences were caused by differences in both models and the library of independent thermomechanical codes and in linear powers used as initial data for this kind of calculations. Accordingly, the fulfilment of acceptance criteria was confirmed for normal operation.

For verifying calculation of fuel thermomechanical characteristics in accidents, an initiating event "control rod ejection" was chosen as the most representative for RIA group. The history of linear power change and all conservative initial and boundary conditions were obtained from the corresponding scenario that was analyzed within accident assessment with use of the DYN3D code. For this set, the results of verifying calculations also showed slight divergences from the results of justification materials: ≈ 70 °C in maximal fuel pellet temperature and ≈ 12 J/g in maximal mean-radial fuel enthalpy. The more conservative results of the verifying calculations are due to the above-mentioned differences in the computational models. More significant divergences in the maximal cladding outer surface temperature (≈ 260 °C) were caused by different approaches to DNB calculation. The additional reason for the analysis of thermomechanical fuel behavior in accidents is assessment of characteristics with use of more precise thermomechanical models. It concerns parameters such as the maximal fuel and cladding temperature and fuel enthalpy, which had been estimated with use of more simple

Fig. 5. Maximal mean-radial fuel enthalpy in control rod ejection by verifying calculation

thermomechanical DYN3D models within accident assessment with use of the DYN3D code. This kind of model refinement could decrease the conservatism of results (Fig. 5).

3.5. Estimation of Effect on Vessel Neutron Fluence

Another important issue of verifying calculations is to estimate the effect of neutron fluence on the reactor pressure vessel. The drawback of the submitted justification materials on RWFA introduction at SUNPP Unit 3 is the absence of direct assessment of this effect. Instead, a comparison of power distribution of peripheral FAs (that give maximal contribution to vessel neutron fluence) was presented. The decrease in power of new FA types in the peripheral row was found, as evidence of decrease in the growth rates of vessel neutron fluence. In the framework of technical review, verifying calculations on the effect of transition from WFA to RWFA fuel cycle concerning vessel neutron fluence were carried out. The calculations were performed with use of the full-scale model for the DOORS package and with account of pin-by-pin neutron sources in accordance with recommendations [9].

The results of the verifying calculations showed 5–7 % decrease in the growth rates of vessel neutron fluence. However, taking

Fig. 6. Campaign-averaged vessel flux in the most loaded axial position

into account fuel campaign duration in transition to RWFA fuel cycle, the assessment of campaign-averaged vessel flux is more relevant in this case. Concerning the campaign-averaged vessel flux, its decrease in the most loaded axial position amounts up to 3 % (Fig. 6).

3.6. Other Verifying Calculations

Among the wide range of verifying calculations in RWFA introduction at SUNPP Unit 3, the following aspects should be pointed out also:

- usually performed verifying calculations of the fuel criticality management system were avoided due to smaller enrichment and different neutron multiplying properties of new fuel (RWFA);
- verifying calculations of isotopic composition with account of the conservative engineering margin allowing the application of the burnup credit approach for analysis of criticality safety in transport, storage and treatment of spent fuel were performed;
- verifying calculations of residual heat of spent fuel were performed to assess the residual heat increase due to transition to RWFA fuel. However, concerning RWFA introduction at SUNPP Unit 3, verifying calculations showed insignificant residual heat increase of about 0.3 %.

However, SSTC NRS models do not cover some aspects of safety analysis of mixed cores at present time. The most important of them are aspects such as effect of local crossflow on the spacer grid–rod friction wear and impact of stiffness and distortion of fuel in the mixed core.

Conclusions

1. Significant efforts are made in Ukraine for the diversification of nuclear fuel supplies from the side of the Fuel Vendor and Operator and the Ukrainian Regulatory Authority with the technical support organization. The fully loaded reactor core of SUNPP Unit 3 with FAs of Westinghouse design indicates this process is successfully ongoing.

2. SSTC NRS capabilities have significantly enhanced for a wide range of verifying calculations carried out the technical support organization of the Ukrainian Regulatory Authority within technical reviews of justification materials submitted in the introduction of new fuel types.

3. The wide range of performed verifying calculations has confirmed the correctness of the safety-related characteristics presented in safety justification materials and compliance with the established in the Ukrainian regulation.

4. Verifying calculations significantly increase the quality assurance of the technical review process as part of the licensing procedure. This provides the Regulatory Authority with reasonable assurance that the justification materials are developed properly.

Nomenclature

CR	- control rod
DBA	- design basis accident
DNB	- departure from nucleate boiling
FA	- fuel assembly
Fq	- pin power peaking factor
LTA	- Lead Test Assemblies
Energoatom	- National Nuclear Energy Generating Company "Energoatom"
NPP	- nuclear power plant
RWFA	- robust Westinghouse fuel assembly

RIA	- reactivity-initiated accident
SAR	- safety analysis report
SSTC NRS	- State Scientific and Technical Centre for Nuclear and Radiation Safety
SUNPP	- South Ukraine Nuclear Power Plant
TVSA, TVS-M	- fuel assembly of Russian TVEL design
WFA	- Westinghouse fuel assembly
WWER	- pressurized water reactor of former Soviet design

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Досвід регулюючого органу в ліцензуванні палива альтернативного постачальника

На даний час на українських АЕС триває процес диверсифікації ядерного палива. Для України питання диверсифікації постачання свіжого ядерного палива є одним із найважливіших пріоритетів, оскільки воно тісно пов'язане з економічною та енергетичною незалежністю. Крім того, наявність кількох різних постачальників палива також визначається положеннями Європейською Стратегією Енергетичної

Безпеки. Ліцензування нових типів палива, особливо коли мова йде про паливо альтернативного постачальника, є доволі складним процесом, до якого залучені як організації зі сторони виробника палива, кампанії, що експлуатують АЕС, так і зі сторони регулятора. Державна інспекція ядерного регулювання України та Державний науково-технічний центр з ядерної та радіаційної безпеки, як організація технічної підтримки, беруть активну участь у процесі ліцензування впровадження нового типу паливних збірок. В рамках цієї діяльності проводиться технічна оцінка матеріалів обґрунтування за різними аспектами ядерної безпеки, включаючи широкий спектр незалежних перевірок розрахунків. З метою забезпечення високої якості технічної оцінки матеріалів обґрунтування впровадження нового палива розроблені нейтронно-фізичні, термодинамічні та термомеханічні моделі, зокрема, для моделювання змішаних паливних завантажень. В роботі представлені особливості розроблених моделей для аналізу безпеки впровадження нового палива та результати їх застосування. Результати охоплюють такі аспекти, як підготовка мало-групових бібліотек макроскопічних перерізів, що враховує особливості нових типів тепловиділяючих збірок, розрахунки перехідних змішаних паливних завантажень, оцінка безпеки в деяких аварійних режимах, викиданням веденням позитивної реактивності, критичності систем поводження з паливом та термо-механічної поведінки тепловиділяючих елементів.

Ключові слова: ядерне паливо, ВВЕР, активна зона, змішані паливні завантаження

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Опыт регулирующего органа в лицензировании топлива альтернативного поставщика

В настоящее время на украинских АЭС продолжается процесс диверсификации ядерного топлива. Государственная инспекция ядерного регулирования Украины и ГНТЦ ЯРБ, как организация технической поддержки, активно участвуют в процессе лицензирования внедрения нового типа топливных сборок. В рамках этой деятельности проводится техническая оценка материалов обоснования по различным аспектам ядерной безопасности, включая широкий спектр независимых проверочных расчетов. С целью обеспечения высокого качества технической оценки материалов обоснования внедрения нового топлива разработаны нейтронно-кинетические, термодинамические и термомеханические модели, в частности, для моделирования смешанных топливных загрузок. В работе представлены особенности разработанных моделей для анализа безопасности внедрения нового топлива и результаты их применения. Результаты охватывают такие аспекты, как подготовка мало-групповых библиотек макроскопических сечений, учитывающих особенности новых типов ТВС, расчеты переходных смешанных топливных загрузок, оценка безопасности в некоторых аварийных режимах, вызванных ведением положительной реактивности, критичности систем обращения с топливом и термомеханического поведения топлива.

Ключевые слова: ядерное топливо, ВВЭР, активная зона, смешанные топливные загрузки

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