Foreword

This publication is a compilation of the papers presented in 11th International Conference on WWER Fuel Performance, Modeling and Experimental Support, organized by the Institute for Nuclear Research and Nuclear Energy (INRNE) of the Bulgarian Academy of Sciences in co-operation with the International Atomic Energy Agency (IAEA), Vienna, Austria, supported by the Kozloduy Nuclear Power Plant (KNPP), the Bulgarian Nuclear Regulatory Agency, and TVEL Fuel Company, Russia.

The Conference took place in hotel Bolero, Golden Sands Resort, Bulgaria, from 26 September 2015 to 3 October 2015. It was attended by 117 participants, among them more than 100 experts and specialists from 22 countries, including representatives of 3 international organizations, 16 Russian organizations and other 36 foreign institutes, nuclear fuel plants, nuclear power plants and organizations responsible for WWER and PWR fuel design, manufacturing and research, and 3 Bulgarian organizations, working for the Bulgarian nuclear industry.

70 papers have been presented in the Conference in 6 oral and 1 poster session, covering: (1) general overview lectures; (2) fuel performance and operational experience; (3) fuel modeling and experimental support; (4) fuel safety and QA; (5) spent fuel performance and management; (6) specific issues of WWER-1000 fuel reliability. The proceedings provide Summary, Conclusions and Recommendations of the Conference, together with the full text of the presentations.

The Conference was financially supported by the IAEA and TVEL Company and the proceedings preparation and print were supported by the KNPP and TVEL.

The papers are published as provided by the listed authors. During the compilation of this publication, the text and the captions have been insignificantly edited for uniformity reasons. The information contained is, however, preserved and in the responsibility of the authors. Neither the Conference organizers, not the publisher, not any of the institutions supporting the Conference, hold any responsibility for the correctness and the use of this information.

The organizers of the Conference and the editors would like to express their thankfulness to all participants who contributed to the success of the Conference by presenting their work and by sharing their competent views. The chair persons of the sessions and panel discussions: Mr. U. Basak (IAEA), Mr. V. Molchanov (Russia), Mr. Yu. Ryabinin (Russia), Mr. D. Tonev, Mr. Y. Yankov, Mr. K. Kamenov (Bulgaria), Mr. V. Meciar, Mr. J. Klouzal (Czech Republic), Mr. A. Ugryumov, Mr. I. Vasilchenko, Mr. I. Evdokimov, Mr. P. Aksenov, Mr. Yu. Bezrukov, Mr. E. Kosourov, Mr. A. Burukin, Mr. A. Shestopalov, Mr. E. Mikheev, Mr. P. Fedotov, Mr. V. Kuznetsov, Mr. G. Shevlyakov, Mr. D. Davidov, Mrs. E. Zvir (Russia), Mr. B. Volkov (Norway), Mr. Ch. Ganguly(India), Mrs. L. Kekkonen (Finland), Mr. J. Deshon (USA), Mr. M. Dye (USA), Mr. P. Van Uffelen (Germany), Mr. J. Biro (Hungary) performed a very important work of leading the Conference sessions and panel discussions. The free international experts Mr. S. Hettiarachchi (USA) and Mr. F. Sokolov (Russia) compiled and analyzed all sessions and panel discussions notes and prepared the Conference Summary, Conclusions and Recommendations.

Special thanks are addressed to the Russian experts, whose presentations concluded the essential core of the Conference outcome.

The prominent experts: Mr. V. Molchanov, Mr. I. Vasilchenko (Russia), Mr. P. Van Uffelen (JRC,EC), Mr. J. Deshon (USA), Mr. K. Kamenov(Bulgaria) etc. gave very important overview lectures and accomplished a high level scientific guidance of the Conference.

The TVEL Fuel Company (Russia) is gratefully acknowledged for the support in providing the expert English – Russian – English cabin interpretation performed by the excellent interpreters Mr. M. Nikitin and Mr. A. Lopukhov (Russia).

The valuable work done by the IAEA staff member Mr. U. Basak (Scientific Secretaries of the Conference) during the preparation of the Conference, made possible successfully to hold the meeting and to overcome difficulties.

IAEA Technical Meeting (TM) “Achieving zero fuel failure rates: challenges and perspectives”, 1 – 2 October 2015, Varna, Bulgaria was organized in conjunction with the 11-th International Conference on WWER Fuel Performance, Modelling and Experimental Support. Conference participants were able to join IAEA TM as observers as well as the TM participants have joined in the conference. The reports presented on TM sessions are included in the Conference Proceedings too.

Summary, Conclusions and Recommendations

Prepared by S. Hettiarachchi and F. Sokolov

The 11th International Conference on WWER Fuel Performance, Modelling and Experimental Support was held in Golden Sands,Varna, Bulgaria from September 26 – October 3, 2015. The conference was organised by the Institute of Nuclear Research and Nuclear Energy (INRNE), Bulgarian Academy of Sciences, in co-operation with the International Atomic Energy Agency (IAEA). The conference was also supported by the Kozloduy Nuclear Power Plant (KNPP) and TVEL Fuel Company, Russian Federation.

This conference was the 11th in a biennial series that has been supported by IAEA and has provided a common forum for better understanding of fuel modelling and performance amongst the WWER community. There were 117 participants from 22 countries that included a large number of experts from research institutes, fuel designers and nuclear industries as well as utility and regulatory staff. The conference was well organised and well conducted.

There were 70 papers: 65 oral presentations and 5 poster papers. All the oral and poster presentations were covered in the following key areas in separate sessions:

- Fuel Performance and Operational Experience
- Improvement of Fuel Design and Operation
- Fuel Modeling and Experimental Support
- Fuel Safety and QA
- Spent Fuel Performance and Management
- Specific Issues of WWER-1000 Fuel Reliability

Four presentations were made in the Opening Session.

Mr. Tonev, Director of the INRNE, delivered the opening address welcoming the participants. He also highlighted various activities carried out in INRNE, Bulgaria. The Institute is the biggest centre in Bulgaria for scientific and application investigations in the following fields: theoretical and mathematical physics, high energy physics, nuclear physics, nuclear energy, nuclear methods and technologies, radiation environment and ecology. A new project of INRNE is to establish a centre for applied research. New accelerator – cyclotron type (24 MeV per nucleon) is already bought and INRNE Cyclotron Laboratory will ensure: a) Production of Radiopharmaceuticals; b) Fundamental Science – Chemistry, Biology, Physics; c) Training and Education in Nuclear Energy, Radiochemistry, Radio-pharmacy, Nuclear Physics and Radiation Protection.

Mr. Basak, Scientific Secretary of the conference presented a paper entitled “IAEA Activities in the Area of Nuclear Power Reactor Fuel Engineering”. This presentation gave an overview of activities in Nuclear Fuel and Material Section of the Division of Fuel Cycle and Waste Technology and then focused on nuclear fuel performance technology related to water cooled reactors in accordance with the theme of the meeting. He mentioned that number of Coordinated Research Program (CRP) were initiated by the IAEA: FUMAC “Fuel Modelling in Accident Conditions” (2014-2018), SMoRE-2 “Accelerator Simulation and Theoretical Modelling of Radiation Effects, Phase 2” (2016-2020), ACTOF “Analysis of Options and Experimental Examination of Fuels with Increased Accident Tolerance” (2015-2019). This activity was initiated as a part of the post-Fukushima IAEA NS Action Plan to examine approaches for development of water-cooled reactor fuels with enhanced tolerance to severe accidents. One of the IAEA CRPs is on Fuel Modelling (FUMEX), aimed at better understanding of fuel behaviour during accidents through identification of best practices in application of relevant physical models and computer codes used in different Member States.

Mr. Ugrumov, staff of the JSC TVEL presented a paper on nuclear fuel for NPPs: current status and main fields of development. TVEL Company’s main product is the supply of fuel for WWER reactors and it provides a 17% share of the world nuclear fuel market. They are committed to ensuring the safety, reliability, economic efficiency and competitiveness of supplied fuel. In this regard, the main focus areas are, provision of the geometric stability of the FAs, increasing the FA operational life time, improving the FA reliability, and realisation of the safe and economically efficient fuel cycle operations up to 18 months duration. A large amount of R&D and technological advances have been made in order to manufacture new fuel compositions as well as improvements on zirconium materials. To achieve this goal, the machine equipment in the fabrication plant is being constantly renewed. Improvements are being made in rolling mills as well as on the technology of obtaining the zirconium blanks. The technology of uranium dioxide powder production and the technology of the fuel pellet fabrication are also being improved. Robotics is used for the assembly and welding of the grids and skeletons during the FA fabrication. The rigs for assembling the fuel rod bundles are also being modernized. The quality control methods used for both the zirconium blanks and the final product are also being improved.

Mr. Ryabinin, staff of the Joint Stock Company (JSC) "Rosenergoatom Organization" delivered a talk on nuclear fuel operation experience in implementing the program of power uprate at WWER NPPs belonging to JSC. He mentioned that the power uprate program of operating WWER-1000 plants was performed by Rosenergoatom using FA-2M and FAA-PLUS for 18-month fuel cycles. Their operation was justified at 104% of the rated power, and extension to 18-month fuel cycles was carried out at WWER-1000 units (except for Kalinin NPP-1). The analysis of actual performance data confirmed the efficiency of the actions implemented, and issues addressed related to the introduction of new fuel type, extended fuel cycles and spent nuclear fuel storage and removal.

Session 1 was devoted to fuel performance and operational experience. The morning session had five presentations. All the presentations mainly dealt with the experience of fuel implementation in WWER reactors – three papers were devoted to WWER-1000 and two to WWER-440. The experience of TVSA fuel implementation at Kozloduy NPP and results of the computer codes validation and utilization were discussed in the first paper. The compliance of the calculated neutron-physics characteristics with the acceptance criteria and sufficient precision of the neutronic codes were shown on the basis of various experimental data and cross verification between the codes.

The second presentation addressed the positive experience of the TVS-2M assemblies implementation at Balakovo NPP in 18 month fuel cycles, at uprated power (104%) and the usage of the axial profiled Gd-rods in order to minimize the power peaking factors and linear heat rate in the upper part in some of the fuel rods. The results of the test operation of fuel rods with different claddings, made by E110M, E125 and E635M alloys at Balakovo NPP were also provided. The recently observed problem with the "white crust" on the cladding surfaces was also discussed.

The third presentation showed the positive experience of different TVSA modifications usage at Kalinin NPP. The strategy of 18 month fuel cycles implementation at uprated power (104%) was also presented. The transition and equilibrium fuel loadings features were discussed. The implementation of burn-up measurement installation MKS-01 was presented, in order to solve the spent nuclear fuel handling and transportation issues due to the increased fuel enrichment and heavy metal mass.

The basic outcomes from the operation of 2nd and 3rd generation WWER-440 fuel at Kola NPP were presented in the fourth paper. The differences between the calculated and measured Gd burnable absorber properties during the first cycle of fuel operation were discussed. The particularities of the fuel loading patterns were also presented.

The excellent Gd-2M^{+} fuel performance in Dukovany NPP WWER-440 reactors was presented in the fifth paper of the Session 1. The enhancements of fuel design, 5-year fuel cycles and power uprate operation up to 108% were discussed. Some comparisons between the calculated and measured neutron-physics characteristics were presented in the paper.

The post-lunch session had 5 papers, including 3 papers that summarized the fuel performance and operational experience of two WWER 440 stations, namely, the Kola Nuclear Power Plant (NPP) in Russia and the Loviisa NPP in Finland. The 4th paper in this session was presented by the Nuclear Power Corporation of India Limited (NPCIL), highlighting the reactor physics computation and the actual measurement during the commissioning of the first WWER 1000 unit of Kudankulum NPP (KK1) in India. The 5th paper was presented by CEA, Cadarache, France, which covered the major features and application areas of the upcoming Jules Horowitz Materials Testing Reactor (JHR), under construction in France. The JHR would be used for irradiation –testing of nuclear fuel and core structural materials for power reactors, produce radioisotopes (Mo 99 in particular) for medical and healthcare purposes and also provide training and education in nuclear reactor technology. The time line for completion of civil work, first criticality and full operation of JHR are mid-2016, mid 2019 and end of 2020 respectively.

The paper on WWER 440 reactors at Kola NPP summarized the first, second and third generation of WWER 440 fuel assemblies that have been developed in Russia. The neutronics calculations for the same have been performed with the modern code complex of WWER Physics Department in the Institute Atomic Stations of National Research Center “Kurchatov Institute”: TVS-M, MCU/REA, BIPR-7 A and PERMAK-A.

The latest design of WWER 440 fuel assembly has no jacket but includes modified frame angles and support tubes. The paper highlighted the deviations between the computed and measured values of the burnup, power and coolant water temperature distribution for odd and even cycles, the calculated and experimental concentration of boric acid in different fuel cycles and deviations between the calculated and measured values of relative power of fuel assemblies.

The two papers from Finland summarized the fuel performance and operating experience of the two WWER 440 units at Loviisa Nuclear Power Plant (NPP) operated by Fortum and the two BWR 880 units at Olkiluoto operated by Teollisuuden Voima Oyj (TVO) and described the Loviisa NPP pool side inspection facility highlighting some recent results of post irradiation examination of WWER 440 fuel assemblies. The two WWER-440 units in Finland have been in operation since 1977 and 1980. The load factors of all operating reactors in Finland have been very high (presently more than 90 %) and the fuel reliability experience has also been excellent. During the total of 71 operational cycles until the refuelling outages in 2014, there were only 34 leaking assemblies. Grid-to-rod fretting was identified as the most common failure mode, but the cause of failure for a significant number of rod failures remains unknown. This corresponds to an average fuel failure rate of $3x10^{-5}$ per year with one failed rod per assembly. The Loviisa WWER 440 fuel assemblies are currently being supplied only by JSC TVEL, Russia. The fuel assemblies have mixing vanes in spacer grids to improve coolant mixing. The average and maximum discharge burnup of Loviisa fuel has been 45 MWd/kgU and 57 MWd/kgU respectively. The integrity of fuel assemblies are evaluated by monitoring the I^{131} and Xe^{133} concentrations in the water. The Loviisa fuel assemblies are considered to be intact during in –pile operation when the I^{131} and Xe^{133} concentrations are < 2.2E2 kBq/m^3 and < 7.4E3 kBq/m^3 respectively. Even if these limits are exceeded, the fuel rods will not be considered as leaking if the I^{131} / I^{133} is < 0.1 , Xe^{133} / Xe^{135} is < 0.3 and I^{131} is < 5.0E4 kBq/m^3. Systematic pool side inspection of irradiated nuclear fuel are carried out to find out the root cause of fuel failures, if any, based on which corrective actions are taken to prevent recurrence of fuel failures , thereby improving the fuel reliability.

The NPCIL +(India) paper summarized the computations performed during the initial fuel loading and the first criticality of KK1 – WWER 1000 based on the Monte Carlo code MCU /REA 1 in order to justify initial dry fuel assembly loading (up to 60 fuel assemblies) and addressed the requirement of the atomic energy regulatory authority concerning the sufficiency of neutron monitoring and related safety issues including boron–free water flooding of the core. The computational results closely matched the experimental results based on neutron monitors.

The afternoon session also had 5 papers, with three Westinghouse papers, one from Ukraine and a paper from USA on BWR fuel performance.

Two of the Westinghouse papers dealt with WWER-1000 fuel design and optimized core loading sequence for Ukrainian reactors. The Westinghouse fuel design has been operating for 8 years with no failures. The grid strap was made of Alloy 718 and bottom nozzle tapers were also added. Both top and bottom nozzles were made of 304 stainless steel. During these developments, fuel assembly pressure drops and vibrations tests were also performed. In the core loading optimization phase, smooth dummies with top and bottom nozzles were employed. Smooth sided dummies will be used for core loading and they will be available for use when necessary. With the use of dummies, it may take longer to load the core, however, no fuel assembly damage and minimized loading trip limits and time.

The third Westinghouse paper discussed the use of LTAs with a burnup of 43 MWd/kgU. Post-irradiation examination was performed on 6 LTAs during this study. No rod damage or grid damage was observed, and the rod drop time was 2.34 seconds. The maximum bow was 9.2 mm (Design 10mm), twist was < 3 degrees, rod oxide thickness was 24 μm (maximum allowable < 50 μm). The fuel assembly growth, RCCA drag forces, oxide thickness, total fuel rod-to-nozzle gap channel closure, and fuel assembly bow data were all within the expected bounds. Safe mixed core operation was demonstrated in this paper.

The Ukrainian paper addressed some aspects of nuclear fuel utilization at Ukrainian NPPs over the past two years. The WWER-1000s were operated up to 3045 MWT. The feed water flow was increased by 1%, and asymmetric power distribution was observed during these operational efforts.

The BWR paper described the challenges faced in achieving zero fuel failures where two major crud/corrosion related events occurred in 1998 and 2003 in the USA. Outside of these, the most common failure mode is often debris related with unknown factors also contributing to some extent. There have not been any corrosion related fuel failures in the USA over the past decade. Japan has led the way by maintaining 96% of the BWRs with zero fuel leakers up to 2006.

Session 2 focus was on the improvement of fuel design and operation. Eleven papers were presented in this session that covered practically all aspects connected with Fuel Assembly (FA) design for current and modern types of WWER reactors. It was noted that there are two Research Organizations in Russian Federation (RF) responsible for FA design and justification of their reliability: OKB "GIDROPRESS", Podolsk and JSC "Afrikantov OKBM", Nizhniy Novgorod.

The reactors of AES-2006 and WWER-TOI projects developed in the 21st century are referred to as the modern WWER. These projects are the result of evolutionary development of WWER-1000 forerunner. An increase in the core height in these projects required the height of the reactor core baffle and protective tube unit to change. An increase in the fuel loading allowed the increase in reactor power to 3300 MWT with simultaneous decrease in the fuel rod linear power from 480 to 420 W/cm.

WWER-1000, AES-2006, WWER-TOI core arrangements differ in both the number of CPS CR and location of ICIS channels. As a result of improvement of the design, the fuel assemblies of new plants are stable to vibration and other operational loads.

The papers described different types of FAs that are being manufactured now and those that will be manufactured in the near future.

Design of TVS-2006 and TVS-TOI is based on the design of close prototype TVS-2 and referred to as generation 3+ and have the following basic characteristics: stability of shape due to rigid skeleton, increased loading with fuel, and application of zirconium materials within core boundaries. For project AES-2006, new FA design without fuel rod fixing in the lower grid (provides simplification of assembly at the manufacturer and assembly/disassembly at NPP) is under development.

Within several years the trial operation of the FAs with MG +ADF (mixing grid + anti-debris filter), new modifications of alloys E-110, E-635, E-125, and also NTMC (neutron and temperature measuring channels) was performed successfully.

Experience since 2003 from the beginning of implementation of the shape-stable fuel assembly prototypes (TVS-2 and TVS-2M) improved the core technical and economic indices which allowed power ascension to 104 % and implementation of 18 months fuel cycles resulting in increased capacity factor which is very important.

It can be shown that at the nominal power of 110 % the DNBR (departure from nucleate boiling ratio) increases from 1.32 to 1.49 due to the mixing grid (MG) effect. Therefore, mixing grids may be considered in the new designs.

Presently, the technical characteristics of TVS-2M, and also new designs have been in great demand. Construction of NPPs with such new fuel designs are planned for Hungary, Finland, Iran, Egypt, China, Bangladesh, etc.

The immediate tasks of improving the designs involve the development of fuel assemblies without fastening the fuel rods to the lower grid, and also fabricating fuel assemblies with increased fuel loading. The papers also outlined the ways of further core modernization and developing new kinds of fuel for NPPs for regional power demand (e.g. WWER-300, WWER - 600).

Some papers addressed the evolution of Fuel Assembles TVSA-12PLUS, TVSA-4 for WWER-1000 reactors. The main trend in the development of this fuel involves increased uranium load (with longer fuel stack) and 12 spacer grids (with 340 mm span). TVSA-12PLUS comprises 12 spacer grids, a fuel column of 3680 mm in height, annular fuel pellet dimension of 7.6/1.2 mm. Top nozzle springs are made of wire 5.1 mm in diameter. TVSA-4 is of the same design except for the fuel pellets. TVSA-4 is equipped with solid pellets of 7.8 mm in diameter. TVSA-12PLUS and TVSA-4 have 3 mixing grids and improved geometric stability. They are supplied with debris-filters and anti-vibration grids for better resistance to debris-related failures and fuel rod vibrations. The refuelling outage may be shortened due to increase in vertical travel speed from 0.6 m/min to 4 m/min. Technical solutions and structural components applied in TVSA-12PLUS and TVSA-4 designs have been validated by their 4-5 year operating experience with these modifications.

The operational experience of TVSA FAs was a key area in papers presented in this session. The first batch of this fuel was loaded into the core of Kozloduy NPP unit 5 in 2005. Since that time about 840 FAs have been manufactured for this Utility, and over ten years of operation there was only one leaking fuel assembly. This is the lowest recorded failure rate among the nuclear utilities operated with TVSA fuel. Among the PWR/BWR reactors, a similar low failure rate was demonstrated in Japanese NPPs compared to nuclear utilities in Europe or the USA. The next generation of fuel planned for Kozloduy NPP is TVSA-12. This fuel has increased uranium content (3530 mm long fuel stack with solid pellets 7.8 mm in diameter) and 12 spacer grids (with 340 mm span). Transition to TVSA-12 is aimed at improving operational characteristics, better fuel utilization, safety, as well as substantial improvement of fuel thermal hydraulics. Implementation of this advanced fuel will enable subsequent power uprates up to 104%.

Session 3 addressed fuel modelling and experimental support which started with two papers on modelling and two papers on two important experimental installations for in-pile testing of WWER fuel and clad materials.

The first paper summarised the model developments undertaken by the TRANSURANUS user group in the frame of the FUMAC co-ordinated research project of the IAEA, which has a strong focus on loss of coolant accidents (LOCA). The models presented dealt with inner-side oxidation of the cladding after burst, and with burst of fission gas release from the high burnup structure under LOCA conditions. In addition, the present paper illustrated the impact thereof on the basis of a few separate effect tests as well as integral tests from Halden and ANL that are considered in the frame of FUMAC. Finally, the perspectives for code improvements were outlined such as the transient cladding creep and further refinement of the porosity evolution in the high burnup structure, along with plans to couple the fuel performance code to core and system codes for a refined safety analysis.

The second paper presented the experimental capabilities and in-reactor tests to investigate the degradation of fuel and structural materials under various operational and accident conditions in the research reactor HBWR (HRP, Halden, Norway). The investigations aimed at the justification of some operational limits for the existing LWR fuel as well as performance of innovative fuel and cladding materials. The paper focused on the investigation of the effect of the following many factors on fuel and structural materials under irradiation:

- Fuel thermal conductivity degradation;
- Fission gas release with burnup;
- Lift-off effect with internal pressure exceeding system pressure;
- Irradiation assisted corrosion of cladding materials;
- Secondary degradation effects on fuel rods damaged during normal operation;
- Fuel fragmentation and relocation under LOCA simulation for high burnup fuel;
- Effect of short term dry-out on high burn-up fuel rod integrity.

The paper also presented the layouts of test rigs, described in brief test techniques applied for fuel rods equipped with gauges for in-pile measurements of temperature, pressure and elongation. The paper finally showed some examples of the experimental results as well as interim and post-irradiation examinations. The developed experimental technical equipment enabled a wide scope of experiments to be performed at the HBWR reactor aiming at the justification and licensing of fuel and structural materials to be operated under both normal and accidental conditions.

The third paper described the in-pile test methods of mechanical properties of fuel and structural materials available at JSC “SSC RIAR” and applied in research reactors MIR, SM and RBT-6 (Dimitrovgrad, Russia). The paper also presented information on their further development and some experimental data obtained with the use of these methods. Among them are creep tests, stress relaxation tests, creep-rupture tests, strain tests of materials used in the nuclear power engineering under different loading types (tension, compression, pressurization, bending, twisting) and conditions (steady-state, incremental, cyclic, and more sophisticated) in water (non-boiling, boiling, supercritical), liquid metal and gas environments. Due to the availability of several research reactors and a wide range of instrumented irradiation rigs, it is possible to simulate operating conditions and predict performance of different reactor materials to show feasibility of the existing and innovative projects of nuclear power engineering. These methods have potential that will be considerably extended after the completion of their development program in the near future. Further development of the mechanical test methods presupposes creation of technical means to be used in the long-term testing of structural materials under cyclic loads, pre-irradiation and creep testing of high burnup fuel from the water-cooled power reactors.

The fourth paper dealt with a new model for burst release in the TRANSURANUS code that is attributed to micro-cracking along grain boundaries during temperature transients, both up and down. It is built on the mechanistic model for gaseous swelling and release that has been implemented in the BISON code as well. The micro-cracking along fragile grain boundaries, observed both in-reactor irradiation and post-irradiation annealing of UO2 fuel, is interpreted in the new model as a reduction of the grain-face gas inventory and storing capacity during transients. The process is modelled through an empirical temperature-dependent function. The model also includes a micro-crack healing process, which gradually restores the original grain-face gas storing capacity. The new model was originally implemented in the fuel performance code BISON. Its implementation in the TRANSURANUS code and subsequent application to the simulation of some experiments of the OECD/NEA database were also presented.

The pre-lunch session had 4 papers, dealing with thermal hydraulics, fuel rod performance enhancement, high burnup fuel rod testing and materials research on zirconium alloys used for fuel rod production.

The thermal hydraulics paper described the development of a 2D module for thermomechanical code based on the COBRA code. The module considered factors such as, convection to liquid, nucleate boiling, transition boiling, departure from nucleate boiling, cladding rewetting, and superheated steam. The model was validated using a 3.6 m long, 19 rod electrically heated bundles.

The paper on fuel rod performance enhancement described some aspects of modelling and fuel inspections at Temelin NPP. It was identified that pellet-cladding gap increases due to fuel swelling and cladding creep. As a result, fuel temperature, local pressure and thermal expansion increases imposing mechanical load on cladding if the gap is closed. Also, the rod length increases with burnup up to 54 MWd/kgU. Crack initiation and propagation was modelled using ABAQUS code.

Several enhancements in the fuel rod performance modelling have been made at ÚJV with ALVEL in the area of assessment of the risk of the fuel rod failure due to the pellet cladding interaction. New model of the creep of E110 cladding has been implemented into the TRANSURANUS code and applied to the fuel rod analysis at Czeh NPPs in order to improve the predictions of the pellet-cladding gap closure and stress relaxation in the cladding. The model has been validated to the data from the OECD IFPE database, and also using the results of the PIIP of TVSA-T at Temelin NPP. The last point is especially important since this analysis provided direct confirmation of the code applicability to the plant using a complex statistical analysis.

The next paper described the testing of high burnup VVWER-1000 rods in MIR reactor under DBA and RIA conditions. Time history of gas release and fuel enthalpy was measured during the study. The peak temperature at the centre of the fuel was measured, as well as the fission gas release. The maximum fuel enthalpy was 3.8 to 4.4×10^5 J/kg, the peak temperature at the center of fuel was 1410 -1450°C, and the fission gas release was up to 20%.

The paper on Zircaloy material study, investigated the grain size of Zircaloy by TEM. Uneven hydrogen distribution up to 300 to 400 ppm was observed. Nano-hardness of the oxide films, and their electrical properties and impedance was also measured. It was recognized that E-110 does not have sufficient hydrogen for re-orientation during transfer to dry storage.

The remaining 6 papers of session 3 were carried forward to the morning session of the following day (i.e. 4^{th} day of the conference).

Two papers from VNIINM dealt with fracture toughness and radiation creep of UO_2 fuel pellets with various grain sizes. Radiation creep was evaluated for larger grain (> 25 μm) pellets at temperatures of 650 to 1050°C. The results were compared with standard pellets of 10 to 20 μm size.

A Hungarian paper investigated the hydrogenation and high temperature oxidation of zirconium claddings. Isotherm steam oxidation was performed at 1000 °C and 1200 °C on E110G samples with 300 ppm and 600 ppm hydrogen content. Similar oxidation kinetics was found in the case of as received and hydrogenated claddings. The higher hydrogen content resulted in a more brittle behaviour. The experiments showed that hydrogen from the normal burnup process would have a negative effect on the mechanical properties of the E110 and E110G fuel cladding in case of a LOCA event.

A Finish paper was presented on multiphysics simulations of fast transients in WWER-1000 and WWER-440 reactors. FINIX is a recently developed fuel behaviour module that is designed to provide “simple but sufficient” descriptions of the most essential fuel behaviour phenomena in multiphysics simulations. It was stated that using dedicated fuel behaviour models in multiphysics simulations can reduce conservatism and uncertainties.

R&D possibilities with nuclear fuel cladding were addressed in a paper from Czech Republic. The availability of 10 MWT reactor with BWR and PWR loops having chemistry control was described. Activity transport and fuel cladding corrosion can be investigated in this facility including PIE. The facility has hot cells and the laboratory is expected to start in 2017.

A paper from Poland described modelling of helium release from the highly burned fuel during annealing and the impact on its migration in the uranium dioxide fuel during neutron irradiation. B He release occurred during annealing at > 600°C, and the fuel porosity was 15%.

Session 4 of the conference dealt with the fuel safety and QA. Five papers were presented in total.

The first three presentations from VNIINM gave important information about the most recent analytical and experimental support of the licensing of the fuel manufactured by JSC TVEL for the Design Basis Accident (DBA) conditions. Namely, the important topics of fuel pellet fragmentation and relocation during the Loss of Coolant Accident (LOCA), fuel rod failure in the Reactivity Initiated Accident (RIA) and the influence of the pre-accident hydrogen content in the cladding on the post-LOCA cladding embrittlement were addressed.

The first paper presented the integral LOCA experiment performed on the very high 72 MWd/KgU burnup WWER-1000 rod segment in the MIR reactor. Test parameters (peak cladding temperature, rod internal pressure, heat-up rate) were selected to conservatively bound any WWER-1000 core. Experiments have shown that the fuel rod relocation and fragmentation occurs in the WWER fuel at this burnup. Simulation of the test was performed by Rapta 5.2 code with the new models for the gas flow in the rod and for the fuel relocation. The application of this model in the licensing analysis was validated.

The second paper described conservative approach to derive the cladding failure criterion (radially averaged fuel enthalpy as a function of the burnup) for the E110 clad fuel rods. Tests were performed in the BIGR pulse reactor in capsule with water at atmospheric pressure. These conditions are bounding to the WWER RIA scenarios. In these tests, two mother rods were used, one with solid pellets and sponge based E110 cladding and the other with annular pellets and standard E110. All rods, even those refabricated from the high burnup rod showed sufficient plasticity in the cladding.

The third paper described in detail the experimental methods used to study the impact of the pre-LOCA hydrogen content on the post LOCA ductility of the cladding. The results have shown that up to 350 ppm there is very limited or no impact of hydrogen, only after this threshold the ECR limit has to be decreased from current 18%. It must be stressed that all WWER rods with E110 cladding up to 72 MWd/kg were shown to contain only up to 100 ppm hydrogen and the experiments thus confirm the validity of current analysis.

The fourth paper from VNIINM presented the ramp test performed in the MIR reactor with the TVSA-alfa rod segments (solid pellet). One of the rods failed by the pellet cladding interaction. Hot cell examination has shown that the defect corresponds to the location of the missing pellet surface underneath. Two dimensional modelling by the ANSYS code (with the utilization of the START-3A) results is now being applied to assess the local increase in the stresses in the cladding caused by the pellet defect.

The last paper from CV Rez introduced the poolside inspection programme at the Temelin NPP and the role of the UJV group in the programme. Currently, Temelin is the only WWER-1000 NPP with systematic inspections of the fuel manufactured by JSC TVEL.

Session 5 was dedicated to spent fuel performance and management, and five papers were presented in this session.

The first paper (RIIAR) presented testing of WWER-1000 fuel rods of various designs and fuel burnups (burnup range of 19-72 MW.d/kgU) under dry storage conditions. The fuel rods used were of standard and improved design with higher amount of uranium in the burnup range of 19-72 MW.d/kgU. Fuel rods were tested in the helium environment at a temperature of 380°C under steady state and thermal cycling conditions. The total time of fuel rod cooling was 318 days under steady state conditions, and 277 days under non-steady state conditions (25 thermal cycles + one steady-state cycle) at a temperature of 380°C. It appeared that non-uniformity of hoop stress along the fuel rod length was caused by the contact between the cladding and the fuel pellet fragments. The impact of thermal cycling on axial strain of the fuel rod operated up to a low burnup of 20 MW•d/kgU was also evaluated. It was found out that the temperature cycling did not produce any significant effect on the changes in the hoop strain of fuel rods.

The 2nd paper (NRC "Kurchatov Institute") presented results of three dimensional calculations of the doses on outer surface of transport container TK-13 with fuel of higher uranium capacity and burnup of up to 70 MW.d/kgU. Unlike the conventional simplified models that revealed the maximum sources of radiation was localized in the central part of the FA, the three-dimensional model illustrated that the dose at the upper and lower parts may be substantially greater than in the central part. The software tools used (GRUCON SAPFIR) fully meet the modern international requirements for computational programs for complex calculations of the parameters that determine the nuclear and radiation safety.

The 3rd paper (RIIAR) presented results of post irradiation examinations of WWER-1000 spent U-Gd fuel rods with different outer diameter of pellets (7.57 and 7.60 mm) with a maximum burnup of 65 MW•d/kgU. The results showed that there is no significant difference between the elongation of U-Gd fuel rods and normal fuel rods at comparable burnups. The cladding diameter of U-Gd fuel rods and normal fuel rods was similar, and no significant difference in the fission gas release from U-Gd fuel rods was observed even with increased uranium capacity. The results of post-irradiation examination of U-Gd fuel rods showed that none of the key performance parameters (geometrical parameters, corrosion, and fission gas release) exceeded the performance criteria of the standard design.

The 4th paper (SRC RF TRINITI) was devoted to modelling of dimensional changes of spent fuel rods during dry storage. It was stressed is one of the most detrimental processes during dry storage of spent fuel assemblies. The thermal creep of fuel cladding may result in damage and loss of integrity of fuel cladding. An approach to model the anisotropic creep of fuel cladding and the overall deformation of spent fuel rods during dry storage was presented. The model was verified against experimental data for cladding tubes made of E110 alloy. A comparison was also made with RIAR data for full-scale spent WWER fuel rods tested under simulated dry storage conditions. The paper also considered normal, abnormal and accident conditions during dry storage.

The 5th paper (NR Center, of ANRA, Armenia) dealt with modelling of dimensional changes of spent WWER fuel rods during dry storage conditions. The paper described the problem of the deficiency of free cells in the spent fuel pool. The fuel assemblies with higher enrichment (3.82%) and consequently, higher burn up requires approximately twice the pre-cooling time of the spent fuel assemblies already in the pool. The possible options to overcome shortage of free cells could be compacting (re-racking) of spent fuel by applying burn up credit approach. Results of the re-racking analysis of the spent fuel pool of a WWER-440 reactor was discussed in this paper. Criticality safety analysis of the spent fuel pool was carried out by applying Actinides-only option for burn up credit approach. The MCNP6.1 code was used for developing the spent fuel pool for Model WWER-440 reactor.

Session 6 focused on specific issues related to WWER-1000 fuel reliability, and eleven papers were presented in this session.

Five papers (6.1 – 6.4 and 6.7) presented information about " Zero Failure Level" Project dealing with organization, status and tasks. The aim of the Project was to improve the reliability and the safe operation of the Fuel, i.e. FAs for WWER-1000 reactors. Three main phases of the activities: (i) determination of the current status, (ii) list of the organizational and technical measures to be adopted by the nuclear fuel developers, producers and utilities, and (iii) the adoption of measures developed in the practical work as well as in monitoring of their implementation were identified as the key tasks. A significant portion of the organizational and technical work has been completed. The main cause for FA leaking has been identified as being to debris damage of fuel rod cladding from foreign materials. Identification and analysis of dominant factors affecting the fuel failure rates in WWER-1000 units in Czech Republic, Bulgaria, Ukraine and Russia were carried out. A study on WWER-1000 fuel was performed in order to prepare recommendations and guidelines for fuel reliability and to achieve lower rates of abnormal fuel operation. The study explored the following issues: systematization of data on fuel failures over the period of 2003 to 2014, consolidation of available experience in application of inspection and repair facilities as well as the best practices employed for enhancing fuel reliability in BWRs and PWRs.

PJSC MSZ (Elektrostal) and PJSC NCCP (Novosibirsk) as a plants-manufacturer, performed the integrated analysis of fuel operation data obtained from NPPs on a regular basis to assess fuel operational results, including fuel failure statistics. The statistical analysis can be used when choosing the priority guidelines to achieve zero failure level.

The working visits by the international experts to the fabrication plants of nuclear fuels and the zirconium alloy component facility took part in the framework of the „Zero failure level“ project. The purpose of these working visits was to determine whether there are any cause-effect relationships between the production of nuclear fuel and the fuel failures as well as to identify the trends and reasons for such failures. As an outcome of the analysis of the production processes at the fabrication plants of the nuclear fuel and the zirconium alloys component facility, the system root causes affecting the failures of the nuclear fuel were not identified.

Three papers (6.5, 6.6 and 6.7) that both Russian plants (PJSC MSZ and PJSC NCCP) manufacturer of WWER fuel presented their fuel performance results. These plants also demonstrated their process capacities, starting from UF6 procurement, its processing, and actual production of finished fuel assemblies. PJSC MSZ as well as PJSC NCCP today can manufacture WWER fuel of all possible designs and modifications. Both plants continuously adopt each other's best practices., which builds up a healthy competition, at the same time broadening the process capacities, as well as enables them to plan a more flexible production schedule. The report presented by JSC CMP (Glazov) described all the steps required to fabricate zirconium components, starting from the procurement of feed material up to rolling of sheets, tubes, bars and manufacture of the applicable parts required to manufacture fuel assemblies. Automated state-of-the-art equipment used for advanced productivity, as well as various installations able to perform numerous inspection steps to assure quality of the manufactured products was showcased. The challenges to be addressed in the nearest future were also presented. The three plants continuously improve the production engineering, reduce expenses, enhance production operations, and develop new types of products.

The paper (6.9) presented data about fuel rearrangement between sectors of 60^{o} symmetry during transportation and installation in WWER-1000 NPP reactors. Two strategies for preventing fuel bow and twist in the core were presented. The first one is rearrangement of fuel assemblies within the core between sectors of 60^{o} symmetry. The particular scheme of cross-sector fuel reshuffling should be chosen by taking into account the main bending direction of fuel assemblies. Another strategy to compensate the excessive bow is to choose the specific pattern of fresh fuel loading into the core based on higher rigidity of fresh fuel assemblies. Both these approaches are applied worldwide in LWRs.

The paper (6.10) reviewed current approaches to detect fuel failures during reactor operation and during refuelling outages in WWERs. Generally, the diagnosis of leaking fuel is performed in three steps. First, failure parameters are estimated by coolant activity during reactor operation. Second, leaking fuel is detected by sipping in the mast of the refuelling machine. Third, additional leakage test is performed in most WWER units to confirm the leak and sometimes to evaluate failure parameters (equivalent hydraulic size of the defect in cladding may be estimated during this activity). These additional leakage tests are performed in the special casks in the spent fuel pool. Uncertainties and limitations of analytical diagnosis of failure parameters during reactor operation are mainly due to a variety of operating conditions of fuel assemblies in the core (e.g. different and broader range of the heat rates, variety of fuel enrichments, and different amounts of Gd, mixed cores).

The paper (6.11) contained information on mitigation of fuel failures and perspectives of moving to 'Zero Failure Level'. It was stated that driving to 'zero' failure rate should include two kinds of efforts, 1) focus on identification of failure mechanisms, and 2) implement corrective actions. It is also important to mitigate the consequences of fuel failures if that inevitably occurs. Fuel failures entail the risks of severe secondary degradation and contamination of primary circuit due to fuel washout. Significant changes of fuel operating conditions (longer fuel cycles, higher fuel burnup, power uprate) and innovations in fuel design bear the risk of higher failure rates for some period of time. Simultaneous implementation of several innovations in one nuclear utility is not advisable since it might be difficult to identify which of the innovations affected the fuel performance positively or negatively. The disadvantage of gradual implementation of corrective actions and any significant changes in operating conditions is the long time needed to evaluate the operational experience. In some cases, it may take up to 4-6 years for each significant change to reflect on the operational performance.

The poster papers by, V.V. Likhanskii, I.A. Evdokimov, V.G. Zborovskii, A.A. Sorokin, S.A. Tokarev, SRC RF TRINITI, on „Simulation of the power profile in gadolinium-doped fuel rods with the fuel performance code rtop“, Janos Biro, MVM Paks NPP, Hungary, on „Special storage of leaking fuel at Paks NPP“, and

S.N Artamonov, V.V. Sergeev, State Scientific Centre of the Russian Federation – Institute for Physics and Power Engineering, Obninsk, Russia, S.E. Volkov, JSC TVEL, Moscow, Russia, on „Evaluated experimental database on critical heat flux in WWER FA models“, won the poster awards.

Panel Discussions: The panel discussions are an important feature of the conference and three discussions were held by primarily the panellists who were the chairpersons of the technical sessions with audience participation. Panel discussion A covered the presentations made in sessions 1 and 2, the Panel discussion B covered presentations made in sessions 3 to 6, followed by the conference overall panel discussion and conclusions.

The panel discussion A covered presentations on fuel performance and operational experience and Improvement of fuel design and operation. Following are the highlights of the Panel A discussion.

- Several speakers mentioned utilization of ^{235}U at 6 to 7 % range, however there may be licensing limit restrictions when the enrichment is above 5% ^{235}U.
- It was speculated that many plants may go up to the challenging burnups of 65-72 MWd/kgU.
- It was mentioned that if the power is increased from 104 to 110%, steam generators can handle the increased flows up to 107% power. WWER 440’s have already operated at 107, 108 and 109% power
- WWER 1000 in Bulgaria is transitioning from TVSA to TVSA-12 fuel
- Fuel grain size increase to > 25 μm is beneficial against fission gas release
- Dukovany’s reporting of no leakers in 10 years was recognized as a great achievement by all, and the need to share that experience was recognized.
- Fuel failures characterized as unknowns is discouraging. More effort needs to be made to identify the cause of failures in order to reduce the unknown factors, which may largely be due to lack of detailed inspections.
- Plants are using spacer grids, mixing grids and two types of intensifying grids
- Fuel Assembly bowing was recognized as an irreversible process
- With the mixed core experience in Ukraine, what the regulators would allow the NPPs to do remains a question, however, sharing this experience with other NPPs is desirable
- Debris leakers were recognized as a major issue. The improved fuel design features with screens added to TVSA was mentioned as one approach to minimize debris failures.
- It was recognized that FME may occur during the fabrication process and in the plant itself. It was mentioned that more and more plants are using mixing grids and debris filters.
- White crust appeared on cladding at 104% power when operational length was increased to 18 months. The crust is in the upper part of cool area. Operationally this is not an issue. Dimitrovgrad will do PIE to identify the cause.
- It was recommended to avoid using SS grids or Inconel grids. High resistance fretting wear disappeared when Inconel grids were replaced with zirconium grids
- It was mentioned that mixed oxide fuels will not improve fuel performance.
- It is important to know the corrosion margins and the grid fretting margins.

The panel discussion B covered presentations on fuel modelling and experimental support, fuel safety and QA, spent fuel performance and management, and the poster session.

Among the key topics discussed during Panel B on modelling included the utility of the TRANSURANUS code on hydrogen uptake, predicting burst pressure, swelling and LOCA simulation. Also, discussed was fission gas release during transients, coupling between fission gas release and swelling, and fuel micro-cracking. Furthermore, different aspects of fuel pin behaviour can also be studied using the TRANSURANUS code.

The ABAQUS code was used to model crack initiation and propagation.

The FINIX model was used for fast transients, and no fission gas release or plastic deformation.

The RAPTA code was used to predict thermomechanical properties and corrosion

In the area of experimental support and QA, highlights were data on fission gas release, four point bend test data on stress relaxation and creep, the grain size evaluations by using TEM and oxide film characterization by nano-hardness measurements. It was recognized that large grain size is beneficial for reduced creep as well as for better fracture toughness.

The criteria for DBA and RIA conditions will be tested at the IGR reactor. The results could be compared with data from modelling activities for DBA conditions and threshold RIA. The fuel hydrogen levels under LOCA conditions where brittle to ductility transition occurs can also be determined by modelling as well as by using ramp tests.

In the spent fuel management area, the key topics discussed included dense packaging and shields, multiscale modelling for fracture dynamics, and measuring pressure inside fuel pins.

In the fuel reliability area, discussions focused on power ramps and transients challenging bowing, debris fretting, PIE, and the new spacer grids in WWERs that eliminated fretting. Grid to rod fretting does not occur anymore with WWER 1000s. After SGs are plugged, and after maintenance, fretting issues seem to occur. In new NPPs, if clean-up is inadequate, fuel leakers are likely to occur. Foreign materials can be introduced during manufacture due to bad welding as well. Going forward, all FAs should have anti-debris filters installed.

The poster papers also covered many of the similar topics including modelling and experimental work to study fuel integrity, and also efforts to support safe operation of advanced nuclear fuel designs. Other topics included the use of mixing grids to achieve critical power levels, and also an analytical study on intensifier grids.

During the final panel discussion, the following topics were identified as critical to improve fuel performance:

- Prevention of bowing of FAs is critical.
- When new reactors are started, there is a risk of losing leak tightness.
- In improving products, human involvement is a challenge. Automation is better, but it is a slow and a gradual process.
- Power plants consider fuel as a consumable. It is up to the public to insist on fuel inspections. In WWERs, inspection of fuel is a must.
- PIE is important to identify failure causes. Some FAs failed due to vibrations.
- All FAs must be installed with anti-debris filters.
- New modelling tools used. Need some focus on accident tolerant fuels.
- Need automated outer surface examinations.
- Large grain fuels have an advantage on factors such as temperature, gas pressure etc. In-pile tests can be done with instrumented fuel rods.
- More studies on secondary hydriding will be useful.
- More research needed on experimental validation of fuel performance as well as on nucleate boiling.
- Need prediction of fuel behaviour in different operating regimes.
- High quality of the conference papers was mentioned. The conference was attended by 4 fuel suppliers, two fuel designers, and 3 fuel manufacturers.
- It was mentioned that more youth involvement is necessary.

It was recognised that the following areas need further attention:

- More information needed on high burnup fuel including FGR and PCI
- More information needed on fuel cladding properties such as swelling, creep and FGR for code improvements
- Need simulation of fuel under severe accident conditions
- Studies are needed in mixed core issues and power uprates
- Need more information on advanced fuels

Overall Conference Recommendations and Conclusions

This conference provided a unique platform for improved interaction, co-operation and understanding among the WWER fuel suppliers, designers, researchers and users. Although major improvements have been made in understanding the fuel performance, fuel design and operation, operational behaviour and modelling, fuel reliability, fuel safety, and spent fuel management, more work is needed in the areas highlighted in the bulleted items in the previous section under the final panel discussion.

1) There was useful discussion on improvements to fuel designs and axial FA profiling for increase the reactor power. Debris filters for TVSA-type fuel for WWER-1000 have proved to be an additional means to reduce the risks of debris-related failures. Nevertheless, it is recognized that they do not provide a 100% protection. Further efforts are needed to improve debris filters and enhance fuel resistivity to debris-fretting in WWERs. It is reasonable to acknowledge international experience in this area.

2) The topics of fuel modelling and computer code application covering LOCA accidents were widely discussed. The experimental capabilities and in-reactor tests to investigate the degradation of fuel and structural materials under various operational and accident conditions in the Halden research reactor (HRP, Halden, Norway) as well as the in-pile test methods of mechanical properties of fuel available at JSC "SSC RIAR" and applied in research reactors MIR, SM and RBT-6 (Dimitrovgrad, Russia) were reported. Further work was planned on the problems of modelling and experimental support for validation of the new models in case of accident conditions.

3) The spent fuel storage issue and material degradation was discussed too and paper in this area need to be encouraged. Additional work is also needed for the implementation and development of the Post-Irradiation Inspection Program (PIIP) in each country.

4) It should be recognized that fuel failures remain inevitable even if their number may be reduced and they may become less frequent. However, the challenge of fuel failure mitigation remains urgent. Severe secondary degradation of leaking fuel during operation may lead to escalation of primary coolant activity and bears the risk of fuel washout into the primary circuit.

Some countries have developed specific guidelines that are brought into action after the failure is detected (normally by monitoring the increase in coolant activity). In some instances, additional restrictions are introduced on the rate of power changes to avoid possible secondary degradation.

It is beneficial to incorporate the issues of fuel failure mitigation (including prevention from secondary degradation) into the framework of national and international 'zero failure' programs.

5) The IAEA Technical Meeting on "Achieving zero failure rates" was held in parallel with the 11th International Conference on WWER Fuel on 1- 2 October. It proceeded separately from the main Conference session 6 on "Specific Issues of WWER-1000 Fuel Reliability" where several papers were presented on "Zero Fuel Failures". Conference participants were able to join IAEA TM as observers as well as the TM members had possibility to visit the Conference sessions. Since communication and information exchange are the cornerstones of driving to zero fuel failure initiative, the events were a good opportunity to share experience on the same topic between experts.

It was recommended that this series of conferences should be continued in 2017 in Bulgaria hosted by the Institute of Nuclear Research and Nuclear Energy (INRNE) in co-operation with IAEA.