IOP Conf. Series: Materials Science and Engineering **130** (2016) 012018 doi:10.1088/1757-899X/130/1/012018

Requirements to the procedure and stages of innovative fuel development

V Troyanov, L Zabudko, A Grachyov and O Zhdanova

Institution "Innovation and Technology Center by "PRORYV" Project, State Atomic Energy Corporation "Rosatom"

E-mail: tvm@proryv2020.ru

Abstract. According to the accepted current understanding under the nuclear fuel we will consider the assembled active zone unit (Fuel assembly) with its structural elements, fuel rods, pellet column, structural materials of fuel rods and fuel assemblies. The licensing process includes justification of safe application of the proposed modifications, including design-basis and experimental justification of the modified items under normal operating conditions and in violation of normal conditions, including accidents as well. Besides the justification of modified units itself, it is required to show the influence of modifications on the performance and safety of the other Reactor Unit' and Nuclear Plant' elements (e.g. burst can detection system, transportation and processing operations during fuel handling), as well as to justify the new standards of fuel storage etc. Finally, the modified fuel should comply with the applicable regulations, which often becomes a very difficult task, if only because those regulations, such as the NP-082-07, are not covered modification issues. Making amendments into regulations can be considered as the only solution, but the process is complicated and requires deep grounds for amendments. Some aspects of licensing new nuclear fuel are considered the example of mixed nitride uranium –plutonium fuel application for the BREST reactor unit.

1. Introduction

The way from the most attractive idea on modifying nuclear fuel for Nuclear Plant up to its application is very long and stretched for many years. This fact is related to so-called process of licensing the fuel with implemented modifications. The current report will touch the issues of innovative structure of fuel roads and fuel assemblies, fuel licensing requirements and regulations, the status of design-basis justification of BREST-OD-300 fuel rods and assemblies, existing calculation codes and the Comprehensive Program of design-basis and experimental justification of dense fuel as well.

2. Innovative structure of fuel rods and fuel assembly of BREST –OD- 300 Reactor Unit

The task of current report is not the discussion of reasons for developing innovative structure of fuel rods and fuel assembly of the BREST-OD-300 reactor unit. We restrict ourselves to listing the features that distinguish the fuel rods and fuel assemblies from the known and currently used design of fuel assembly of BN-600 reactor and BN-800 as well.

Structural material for fuel rod cladding and end pieces, as well as all fuel assembly elements is a ferritic-martensitic class steel called EP 823. This steel has never been used in fast reactors except for some experiments in the BOR-60 (cladding of several experimental fuel rods with oxide fuel) and BN-600 (fuel assembly hexagonal wrappers in small quantities).

The pellet column is MNUP. Before starting work on "PRORYV" Project it has been tested several MNUP fuel rods in the RR BOR-60 core , but these experimental fuel rods have differences that do not allow considering them fully referent, the test parameters varied from the parameters of operation of the fuel rods in BREST-OD-300. Great experience of uranium nitride fuel in RR BR-10 application is an important fact confirming, in principle, operation of the fuel rods, but it cannot be considered as a referent in relation to the fuel rods with MNUP.

The diameter of the fuel rods of two types (9,7mm and 10.5mm), as well fuel rod pitch in a grid (near 12.7mm), are new and based on requirements to the reactor neutronic characteristics. These characteristics are important for the thermal hydraulics, and require the development of new designs not only the fuel rods but fuel assemblies as well. The initial idea to use ribbed fuel rod cladding with self-spacing function inside hexagonal fuel assembly can was almost unrealizable because that has not been found possible industrial production of fuel rod cladding with high edge which provides a space for fuel rods distancing. It should be noted that from the licensing of ribbed fuel rod cladding point of view, developers would be faced with a set of additional problems in terms of design-basis, and, especially, in the experimental justification of fuel rods because it would be difficult to carry out the reactor testing of fuel rods of such a geometry.

The design of fuel assembly is unwrapped which is similar to fuel assembly TVS-2M type for VVER-1000. But this analogy is very conditional, because the spacer grids made from steel EP 823 only visually are similar to the grids from zirconium alloy E110. Suffice to say that the strength properties of the steel EP 823, in particular the elasticity modulus, which is three times more and the yield strength, which is also 2-3 times higher, make additional difficulties during forming cells and welding of spacer grids and requirements annealed weld points within a short time after welding make difficulties in the manufacturing racks. Thermo-mechanical behavior of unwrapped fuel assemblies during operation being a part of the BREST' core have significant features, and loading factor of fuel assemblies while operating is not completely the same as in VVER-1000.

Regulation requirements [1] do not settle the required scope of design-basis and experimental justification, but specify that justification should be sufficient and included into design and engineering documentation of the developer. ("While developing the technical specification on reactor unit and (or) modifying the reactor core with the help of new structures of fuel assemblies, new fuel composition and updating the reactor control and safety system or other systems important for safety, the necessary reactor and development testing have to be performed. For proving the fulfillment of safety criteria the sufficiency of performed testing has to be shown in the reactor unit specification (cl. 2.1.4)"). In other words, in reactor plant technical specification or any reference materials, the results of design-basis justification as well as experimental ones in "sufficient" scope should be listed. According to [1], for design-basis justification only certified methods and codes can be applied, but the application area of these certified codes should spread to the conditions, structures and materials appropriate to the licensed product.

Experimental justification must be performed on the reference structure when loading parameters (power, mechanical, thermal, irradiation, corrosive loading) are equal to design ones. This requirement is easier to perform, if the changes in existing units are licensed, especially if there are appropriate experimental facilities and prototype or research facilities as well. As an example we may consider the changes in structural material in the fuel rods of operating reactors: there is a technical possibility to carry out experimental work with new fuel rods in identical conditions. If the experimental facilities are absent, as in the case of BREST type reactors (there is no lead-coolant reactor in operation as the basis), implementing the requirements of experimental justification and reference in sufficient scope becomes unfulfilled. Not only in normal operating conditions the justification shall be implemented, but in all accidents, foreseen by the project. The developer is obliged in its design documentation to justify and to set the maximum design limits for fuel rods, safe operation and operational limits. Such limits are set in [1] for RBMK and VVER fuel, as well as for BN reactor oxide fuel.

Limits have to be set and, in addition, put in [1] for MNUP. Otherwise there is a conflict with the regulations, because it is impossible to meet regulations which do not specify the limits.

3. Codes for fuel rods while operating in normal conditions and their breakdown

For a long time in various industry organizations different calculation codes for calculating the working capacity of container type fuel rods of fast reactor were created, applied and forgotten. Currently, the only code which is practically used for fast reactor fuel rods is "KORAT", developed and operated by JSC "VNIINM", the chief designer and technologist of fuel rod. Code "KORAT" has been certified by Scientific and Engineering Centre for Nuclear and Radiation Safety for calculating fast reactors fuel rods with uranium oxide and MOX fuel in normal operating conditions (stationary, transient and maneuvering modes of reactor operation).

The formulation and the calculation algorithm are using the approaches specified in [2], of which the main ones are the calculation of the plane circular cross-section of a fuel rod under conditions of plane strain, central symmetry and annular layers with different properties, that allow simulating the fuel pellet, the gap between fuel and rod cladding and fuel rod cladding. The loading is monotonous.

Independent calculating the number of plane cross-sections by the height of fuel rod provides the ability to sum the amount of axial deformation and to determine the elongation of the fuel rod. However, this calculation can be very approximate, since the model does not take into account the capacity of axial ratchet in the interaction of the fuel column and the rod cladding, just as is done in models [3].

The "KORAT" code is adapted, as far as it was possible, to calculate the fuel rods with nitride fuel. For this purpose physical and mechanical characteristics of nitride pellet and the cladding made of steel EP 823 were entered. However, in the "KORAT" code the model of the gas swelling of nitride pellet is missing. The model takes into account the temperature, density scales and structure parameters of the fuel pellet, including porosity. The model of gas emission from the fuel pellet is also absent. This is the reason why both gas emission and swelling are set by empirical dependencies in the "KORAT" code.

The “KORAT” code has no certification on calculating fuel rods with nitride fuel.

The “PINCODE” code is designed and certified in JSC "Institute of Physics and Power Engineering IPPE". The code is used for the calculation of thermo-mechanical behaviour of UOX fuel rods in normal operating conditions. The code was developed within the frame of the SVBR Project development, but for calculating fuel rods with nitride fuel the code was not used.

The code “DRAGON” is developed in IPPE. The code is focused on thermo-mechanical behavior of fuel rods with nitride fuel. For this purpose the model for calculating the swelling of the fuel core under the model of spherical gas pores is entered.

This fact distinguishes the "DRAGON code from the others, because potentially in the "DRAGON" code the swelling of the pellet column depends on the following parameters: irradiation, gas composition and gas pressure under the fuel rod cladding and fuel temperature, and the most important parameter is the stress condition in elementary volumes of the fuel pellet. This condition depends on temperature gradients and isostatic pressure caused by the interaction of the pellet column with the fuel rod cladding.

Axial ratchet is missing in the model and the algorithm of fuel rods calculation by the "DRAGON" code, but the task of introducing this model is settled. The algorithm is going to be implemented during the years 2016-2017. Furthermore, the nearest works to improve calculations of fuel rods by the "DRAGON" code cover the algorithms and techniques of statistical calculations which are taking into account the statistical uncertainty of input data and operational parameters, as well as the individual characteristics of each fuel rods in the core in the current time for determining the probability characteristics of fuel rods failures.

The “DRAGON” code is verified at problems with an analytical solution as well as on all available experimental data for testing fuel rods with nitride uranium fuel and MNUP. This code is not certified in national authorities, because for its full certification is not yet sufficient experimental data obtained in studies of nitride fuel rod prototype.

The "DRAGON" and "KORAT" codes are used while developing explanatory and calculation notes as part of the design and engineering documentation on all experimental and prototype fuel rods and fuel assemblies which are being tested in BOR-60 and BN-600 reactors within the framework of the fuel licensing program (see below). Moreover, as neither “KORAT” nor “DRAKON” is certified for MNUP justification, the application of two independent codes and commands is used as a cross-verification of calculations for increasing calculations reliability.

The "BERKUT" code developed by the Nuclear Safety Institute (IBRAE) was initially established as a mechanistic (best estimation) code for calculating nitride fuel rods with the application of precision physical models. However, the lack of fundamental data on characteristics and phenomena in nitride fuel has forced developers to change the concept. The engineering version of the "BERKUT” code has been created, but there “initial statements” are absent. At the same time, in engineering version of “BERKUT” a models of pellet column swelling and fission gas producing emission are missing. The “BERKUT” code hasn’t been applied in practice yet, because it does not represent new ideas, methodologies and algorithms. Certification of the "BERKUT" code was not performed for the reasons mentioned above.

All listed calculation codes are intended to calculate quasi-stationary loading and are used to calculate the fuel rods in normal operating conditions and deviation of normal operating conditions. Criteria of fuel rods integrity do not differ from the criteria accepted for normal operating conditions. While developing the technical specification the list of deviation during normal operating conditions is considered, but usually one self-run is made as the most hard deviation.

The calculations of accident modes, including dynamic ones, are a task for the coming period. Also it is needed to develop a methodology of MNUP justification in the modes of power cycling and, in addition, seismic loading.

An independent task is the calculation of the thermo-mechanical behavior of the fuel assembly. The fuel assembly of beam-rod model is developed in IPPE. The model rather borrows approaches and some results of the "TEREMOK" code development [4]. Currently, only engineering calculations and verification according to factory testing of fuel assembly mockups are performed.

There is no model of fuel cladding’ corrosive effect and spacer grids in the lead coolant environment, although such attempts are undertaken. Currently corrosive effect is accounted as corrections for the corrosion via the reduction of structural element thickness.

Correction for corrosion is introduced by the results of corrosion experimental studies in lead stands without reactor irradiation. The experimental possibility of determining corrosion in reactor conditions is absent.

4. The fuel rods calculations in accidental situation

Investigations on the accidental situations analysis in BREST OD-300, made by the Chief designer of reactor unit, has shown that the design accidents can lead to increasing of fuel cladding up to 1100°C . In general the “DRAGON” code could be used for calculating the stress-strain state of fuel rods in these conditions. However, these calculations are faced with the lack of information on physical and mechanical properties of fuel cladding and pellet column in same temperature level and total absence of experimental data to verify these calculations.

According to [1] it is required to introduce the so-called maximum design limits (MDL) for fuel rods which are needed for justification accidental situations on reactor units. In case of increasing the limits fuel rods will be failured. Such MDL for BREST fuel rods are developed on the basis of data on long-term strength of fuel cladding at high temperatures.

The loading conditions of the fuel rod are accepted as typical, based on calculations of gas release from the fuel. Calculations are made according to the worst combination of parameters. The following data is characterized MDL: temperature 800 ^{o}C – t_1, 900 ^{o}C – t_2 , 1000 ^{o}C – t_3. In addition, MDL on MNUP maximum temperature of and the cladding deformation is introduced as well.

5. The program of experimental justification of fuel rods

The structure of works on MNUP justification is specified in Figure 1.

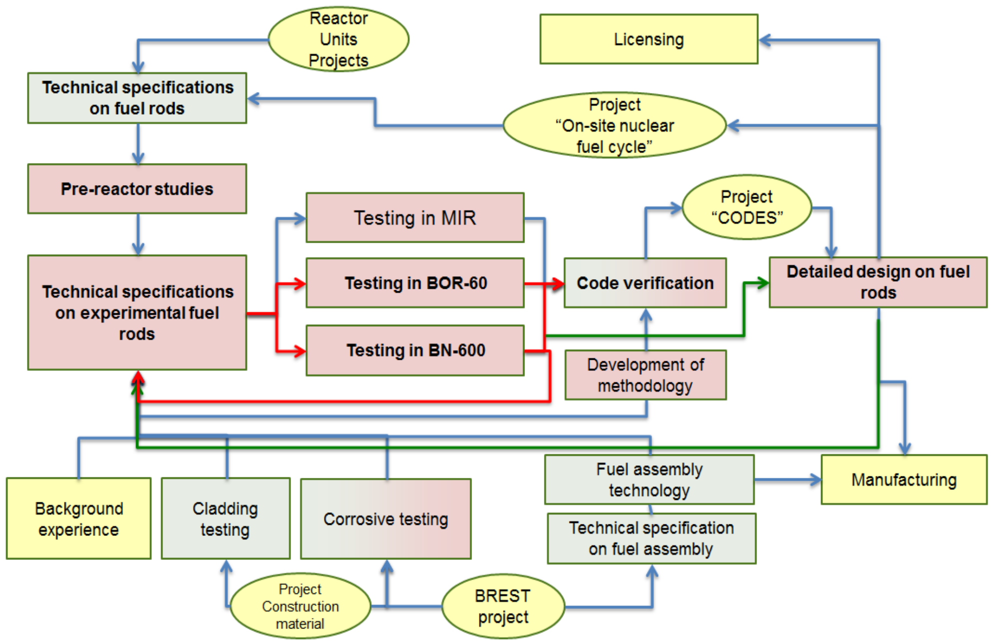

Figure 1. The structure of works on MNUP justification.

The Comprehensive Program of design-basis and experimental justification of dense fuel [5] was developed and approved on December 31st, 2013 by Dr. Pershukov, the Deputy Director General of State Corporation "Rosatom". As a result of executing the mentioned Program the working capacity of fuel rods with MNUP under main operational parameters of BREST-OD-300 and BN-1200 reactor's cores should be explained.

Part of the Comprehensive Program of design-basis and experimental justification are the following:

- pre-reactor examination of MNUP composition;
- methods and equipment for pre-reactor studies;
- development of MNUP carbothermal synthesis and basic technologies for pellet manufacturing;
- the manufacture of experimental fuel rods and fuel assemblies;
- reactor testing in BOR, MIR and BN-600 reactors;
- post-reactor examination;
- modeling and codes (corrosion, fuel rod, thermo-mechanical characteristics of active zone);
- development of structure and basic technologies of BREST fuel assembly;
- alternative structures and technologies;
- technical specifications of fuel rods (standard and experimental).

Schedule for testing MNUP in RR BOR-60 for the period 2012-2016 is given in Figure 2.

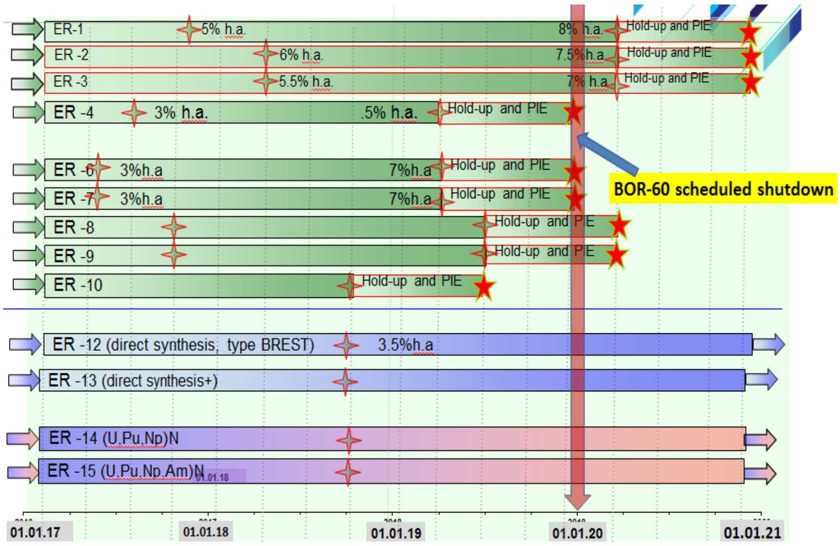

Figure 2. Schedule for testing MNUP in RR BOR-60.

In BN-600 the more meaningful testing conditions in terms of length of irradiation at steady-state operating parameters with minimum influence of transient mode and variations in power are provided.

To ensure the testing presentability in terms of burning and neutron dose and fission density experimental fuel assemblies are placed on the core periphery of BN-600 reactor. This fact leads to significant peaking factor in experimental fuel assembly (EFA). The irradiation parameters of standard fuel rods have a significant range depending on the location of the fuel assemblies. Thus, the results of experimental fuel assemblies' irradiation in conditions of significant heat gradient along the radius will provide data for reasoning the working capacity of fuel rods with various levels of linear heat generation rate, but with required fission rate.

Schedule for testing MNUP in BN-600 is given in Figure 3.

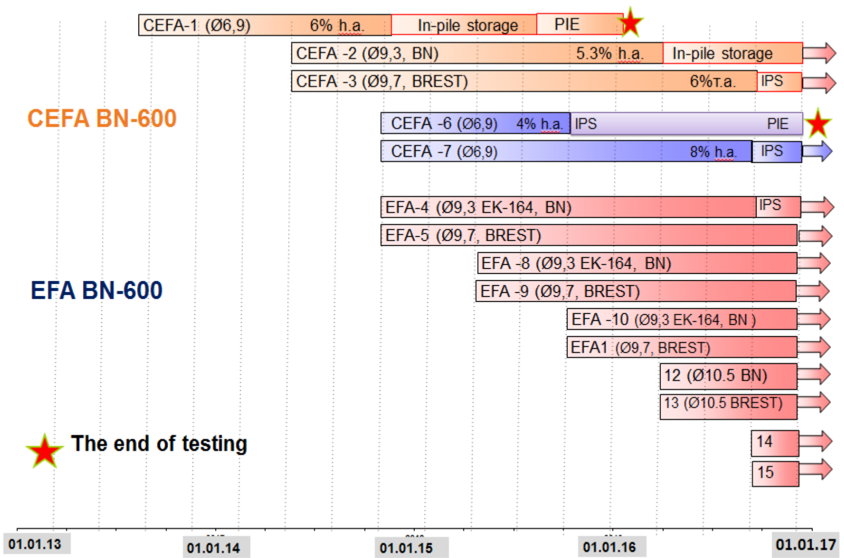

Figure 3. Schedule for testing MNUP in BN-600.

Post-irradiation studies of EAF fuel rods of BN-600 will provide the following additional data on the properties of the mixed nitride:

• by the MNUP swelling rate depending on temperature and burning;
• by the fission gas release from MNUP depending on temperature and burning;
• by the chemical interaction between MNUP with claddings EK 164 hd and EP 823 depending on burning and temperature.

The experimental data will be obtained as a result of works. The data is needed to verify the calculation of temperature and stress-strain state of a fuel rod with MNUP for smooth cladding.

Obtained results will give an opportunity to prepare design-basis justification for making decision related to maximum valid capacity of fuel rods of BN-1200 type reactor and BREST with MNUP as well. The main problem of arranging the experimental fuel irradiation in reactor BN-600 is the need to obtain a permit for testing in Russian Federal Service for Ecological, Technical and Atomic Supervision. But the permit application means an availability of sufficient justification of EFA working capacity.

6. General approach to licensing fuel rods of Reactor Unit BREST-OD-300

The above shows that an innovative type of nuclear fuel cannot be fully experimentally proved, because it is impossible to obtain the full reference during testing the fuel rods. At the same time the question of "sufficiency" of testing in the context of the amount and testing statistical assurance appears.

Moreover, the study should be performed for all operational parameters of reactor unit based on the factors of the statistical uncertainty of operating parameters (so-called factors of overheating). This means that experimental verification should also consider the factors of overheating, but to fulfill it requirements in the experiments methodologically impossible.

The fact that such statistical deviations in testing is impossible to measure and set, thereby to confirm that they were realized in testing, because to definite the rate of these overheating factors often neither set nor measure is not possible. That is, it is impossible to experimentally confirm that these overheating factors were implemented in testing and the experimental result is corresponding to factors.

The only way of new fuel licensing is basic-design justification with the help of certified calculation codes, which verification and certification are extended to the whole range of design and operational parameters of the justified items. For this purpose, the experimental results should cover the range of required parameters; extrapolation of results must be justified and carried out with the help of proven models. In this case, it is possible separately to investigate the properties and behaviour of structural materials and pellet column, to prevent individual differences in the design of experimental items from the licensed ones and to have variations in operational parameters, so as the effect of these variations can be mathematically modeled and described by the physical models.

For example, the length of the pilot fuel rods is not a factor of primary importance, if the behavior model of plane sections of a fuel rod is verified, and the neutron flux density in experimental fuel rods may differ from the neutron flux density in a licensed item, if the effect of neutron flux on the structural material has been studied in other experiments which are equal to by this parameter to the licensed items. Thus, the physical models are formed the basis of such hypothesis. In physical models the influence of the neutron flux density is important for describing the behavior exactly of structural material, and the behavior of the pellet column is not due to the neutron flux density, but the fission density in fuel material, its temperature and structural parameters.

While verifying the behavior models of the pellet column firstly it is required to ensure the adequacy of the linear heat rate and identity patterns of the fuel composition, wherein to receive the rates of energy producing (or burn-up) which correspond to real rate.

This ideology forms the basis of the Comprehensive Program of design-basis and experimental justification, which were discussed in section 5. The influence of each factor on fuel rod performance is determined in a pre-planned testing either in the research reactor BOR-60 or in the commercial reactor BN-600.

The combination of these studies allows verifying and certifying the calculation methodology and calculation code, such as "DRAGON". The limiting parameter which will decrease the area of application (certification) of the “DRAGON” code is achieved heat generation rate (rate of burn-up). The reason is the absence of established model, which prove the influence of this exact factor on the performance of fuel rods with MNUP due to lack of data about processes associated with the evolution of the structure and accumulation of fission products, and the insufficient certainty of the extrapolation of pellet column properties.

The models associated with cracking of the pellet column and fuel rod operating in modes of power cycling will remain not fully verified. The requirement for power cycling when operating in a wide range is specified in the technical specification on BREST reactor, but targeted experimental studies in this direction have not conducted yet. This is due to the fact that neither in RR BOR-60 nor in BN-600 where MNUP is testing, such power changes are not implemented while operating. Technical capability to hold such a reactor testing is not available, and the possibility of justification of such regimes will appear only after the construction and launch of BREST-OD-300 reactor.

Thus, the lack of research reactor unit with lead coolant eliminates the possibility to verify calculation models and programs on some important operation parameters, such as corrosion of structural materials in lead under radiation or fuel rods and assemblies in modes of cycling or power transients. But the experimental demonstration reactor unit with BREST type reactor is constructing to prove the feasibility of idea of a lead reactor with nitride fuel, isn’t it?

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