# Моделиране поведението на ядрено гориво за реактори тип ВВЕР-1000 с ФРАПКОН -3 при нормални експлоатационни условия

**Marina Andreeva, Totju Totev, Stoyan Stoyanov**

*Резюме: В статията са представени резултати от моделирането и оценката на поведението на ядрено гориво за реактори тип ВВЕР-1000 при нормална експлоатация, получени с помощта на интегралния код ФРАПКОН-3. Моделирането и количествените оценки са извършени в Аргонска Национална Лаборатория, САЩ.*

## Modeling of the WWER-1000 fuel-rod behavior in Steady-state condition with FRAPCONE-3 computer code

**Marina Andreeva, Totju Totev, Stoyan Stoyanov**

*Abstract: It is presented within the paper the results of the modeling and the assessment of the integral code predictions of the WWER fuel-rod behavior in steady-state condition. The assessments in this paper have used the MASSIH and ANS 5.4 subroutine in the code. The modeling and calculations have been performed with FRAPCONE-3 computer code in Argonne National Laboratory, USA.*

## I. Introduction

WWER type reactors are to a great extent similar to the western LWR's. This is especially true for WWER-1000 type reactors. At present, there are altogether 26 WWER-1000 units in operation and 11 in construction in Russia, Ukraine, Bulgaria, Czech Republic, China, India and Iran.

According to Bulgarian Nuclear Regulatory Agency's Requirements [1]:

- The system of physical barriers of any nuclear unit of a NPP shall include: the fuel matrix, the fuel cladding, the reactor coolant system pressure boundary and the reactor containment system.
- Plant safety shall be analyzed using deterministic and probabilistic methods to verify and confirm the established design basis and the effectiveness of defence in depth arrangements.
- Computer codes, analytical methods and plant models to be used in the safety analysis shall be verified and validated. Uncertainty of the results shall be quantified.

## II. Fuel Rod Modeling Computer Codes

Some of the well-known computer codes modeling fuel rod behavior in steady-state conditions, small transients and small deformation are as follow:

- GAPCON-THERMAL 2; FRAP – S3, FRAPCON-2/3 (NRC);

- ENIGMA (BNFL); TRANSURANUS (EU); FEMAXI (G); COMETHE; COPERNIC; ENIGMA; FALCON, FRAY; FEMAXI; METEOR; PIN-micro; START.

**II.I. FRAPCON-3 computer code**

The FRAPCON-3 code iteratively calculates the interrelated effects of fuel and cladding temperature, rod internal gas pressure, fuel and cladding deformation, release of fission product gases, fuel swelling and densification, cladding thermal expansion and irradiation-induced growth, cladding corrosion, and crud deposition as functions of time and fuel rod specific power.

The calculated procedure is illustrated in Figure 1, a simplified flowchart of FRAPCON-3.

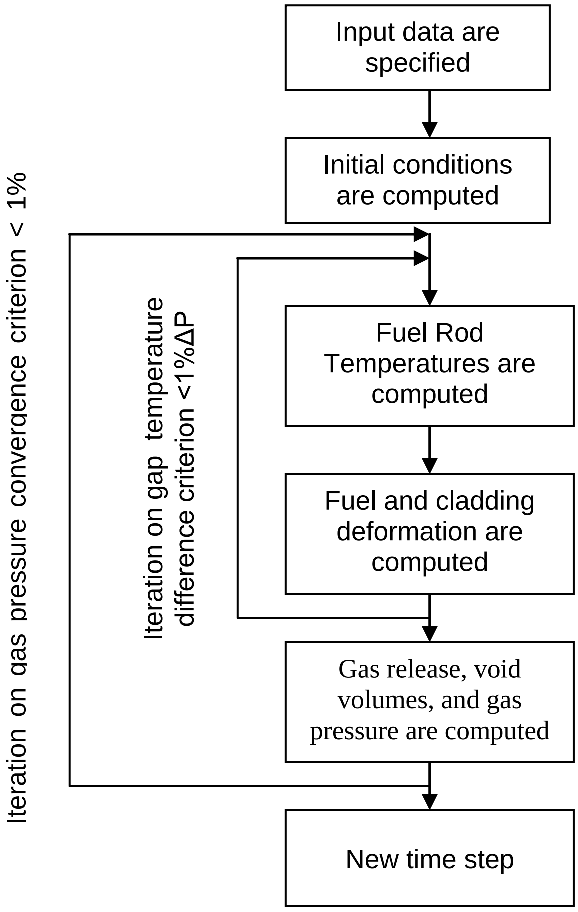


Figure 1. Simplified FRAPCON-3 Flow Chart

The calculation begins by processing input data. Next, the initial fuel rod state is determined through a self-initialization calculation. Time is advanced according to the input-specified time-step size, a steady-state solution is performed, and the new fuel rod state is determined. The new fuel rod state provides the initial state conditions for the next time step. The calculations are cycled in this manner for the user-specified number of time steps.

The solution for each time step consists of 1) calculating the temperature of the fuel and the cladding, 2) calculating fuel and cladding deformation, and 3) calculating the fission product generation and release, void volume, and fuel rod internal gas pressure. Each of these calculations is made in a separate subcode. As is shown in Figure 1, the fuel rod response for each time step is determined by repeated cycling through two nested loops of iterative calculations until the fuel-cladding gap temperature difference and internal gas pressure converge.

**II.2. Code structure**

The FRAPCON-3 is a large and complex that contains over 200 subroutines. These subroutines have been grouped in packages, not all of which need to be complied for every run. These packages are listed in Table 1.

| Package | Description |
|---|---|
| FRPCON | The main section of the code, including all of the thermal models; also includes the FRACAS-I mechanics model. |
| FRACAS-I | Contains the subroutines comprising the FRACAS-I mechanics model. |
| MATPRO | The MATPRO material properties package. |

Table 1. Major FRAPCON-3 Packages

Material properties of Zr-1%Nb which is used as cladding in WWER-1000 type reactors are not provided in MATPRO package, However, Zr-1%Nb alloy has comparable mechanical properties with annealed Zircaloy-4 [8,9]. Furthermore, oxidation characteristics of both alloys are not very different. Hence, Zircaloy-4 is used as a substitute for Zr-1%Nb.

The independent integral assessment of the FRAPCON-3 code represented in [3] indicates that the code is capable of analyzing commercial LWR fuel behavior and authors have been performed calculation for generic WWER model.

**III. Fuel assembly design and characteristics**

Design features important to the analysis of fuel behavior during steady state and accident conditions include the design of the fuel and cladding, the arrangement of the fuel rods in the fuel assemblies and the arrangement of the fuel assemblies in the core. As shown in Table 2, the component of the fuel and cladding is approximately the same in all of the relevant power reactor designs. For example, $UO_2$ is generally used in most of the designs, although MOX or UO2Gd2O3 pellets might be used. All of the fuel designs use fuel pellets, prepared from a sintered powder with a theoretical density above 90%. The fuel pellets typically are shaped to minimize pellet–cladding mechanical interactions and may include dished and/or tapered ends. Some of the designs use annular pellets (e.g. for WWERs) devoted, to some extent, to improving the performance of the fuel during normal and accident conditions (e.g. fission gas releases, internal pressure). The fuel cladding of the different reactor types is also more or less similar, using different alloys of zirconium. Some advanced designs may also include special coatings or different layers of cladding materials to further reduce corrosion and fuel rod failures during normal operation or accidental situations. Also, the BWR assemblies use control elements that are located outside the actual fuel assembly, while the PWR and WWER assemblies include control rods distributed within the fuel assembly. All the assemblies also include some type of grid spacers or spacer elements composed either of zirconium alloys or structural materials such as stainless steel or Inconel.

| Design parameter | PWR | BWR | WWER |
|---|---|---|---|
| Fuel material<br><br>Type<br>Active length (m) | UO2, MOX,<br>UO2, Gd2O3<br>Pellet<br>~3,6 for 3 loops<br>~4,2 for 4 loops | UO2<br><br>Pellet<br>3.81 | UO2<br><br>Annular Pellet<br>~2,4 (WWER 440)<br>~3.6(WWER 1000) |
| Cladding material | Alloys of Zr | Zircaloy-2 | Zr+1%Nb |
| Control element location | Internal | External | Internal for WWER-1000 and special assemblies for WWER-440 |
| Assembly | Rectangular fuel rod array including control rods, zircaloy spacer grid | Rectangular fuel rod array with water tubes, zircaloy spacer grids | Hexagonal fuel rod bundle, Zr-Nb spacer grids |

Table 2. Design parameters for various reactors

| **Parameter** | **WWER-1000** |
|---|---|
| Thermal power, MW | 3000 |
| Electric power, MW | 1000 |
| Uranium loading, t | 70 |
| Number of FA in the core | 163 |
| Number of control systems | 61 |
| Annual refueling, % core/number of FA | 33/54 |
| Maximum enrichment, wt.% U | 4.4 |
| Average discharge fuel assembly burnup, MWd/kgU | 42-44 |
| Single fuel cycle time, eff. Hours | $(7\text{-}8).10^3$ |
| Maximum power of FAs, MW | 27 |
| Coolant rate, m/s | 6.0 |
| Coolant temperature at the outlet of maximum power density a FA, C | 335 |
| Coolant temperature in the core, C | 30 |
| Coolant pressure, MPa | 16 |
| Linear heat generation rate, W/cm<br>Medium<br>Maximum | <br>167<br>448 |
| Maximum outside temperature, C | 352 |

Table 3. Main design parameter WWER-1000 core and fuel operation

| **Parameter** | **WWER-1000** |
|---|---|
| Assembly geometry | Hexagonal |
| No of rods per assembly<br>- Fuelled<br>- Unfuelled | 331<br>312<br>19 |
| Overall assembly length, mm | 4570 |
| Overall assembly width, mm | 234.5 |
| Lattice pitch, mm | 12.75 |
| Fuel element length, mm | 3840 |
| Fuel column length, mm | 3530 |
| Rod outside diameter, mm | 9.1 |
| Pellet length, mm | 9-12 |
| Pellet outside diameter, mm | 7.57 |
| Central hole diameter, mm | 2.35-2.6 |
| Pellet-cladding gap, mm | 0.15-0.26 |
| Pellet density, $g/cm^3$ | 10.4-10.8 |
| Helium pressure, MPa | 2.0-2.5 |
| Average linear fuel rating, kW/m | 16.7 |
| Peak linear fuel rating, kW/m | 44.8 |
| Maximum fuel temperature, C | 1667 |
| Cladding material | Zr-1%Nb |
| Cladding thickness, mm | 0.63 |
| Average discharge burnup, MWd/kgU | 55 |
| Maximum assembly burnup, MWd/kgU | 60 |

Table 4. WWER design data

## IV. Results

The steady state FRAPCONE results for WWER predictions shown on Figures 1-11 include plenum pressure, fuel stack axial extension, fuel centerline temperature, gap gas pressure, cladding axial extension, cladding hoop strain, effective cladding stress, cladding outside temperature, and fission gas release

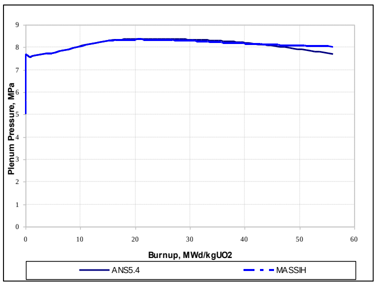

Figure 1. Plenum Pressure

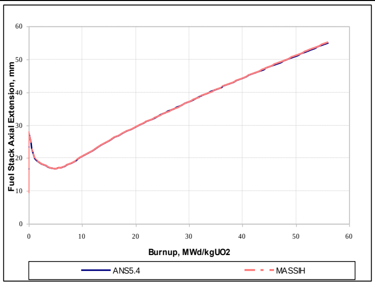

Figure 2 Fuel Stack Axial Extension

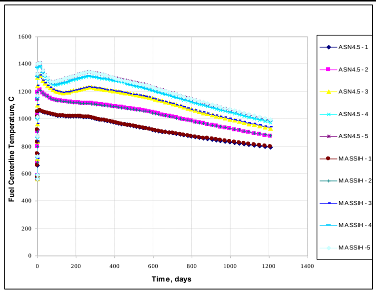

Figure 3 Fuel Centerline Temperature

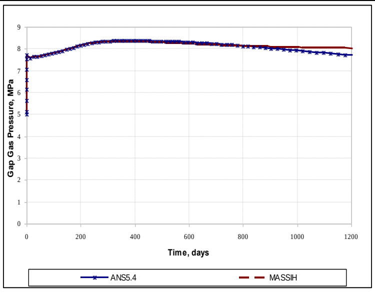

Figure 4 Gap gas pressure

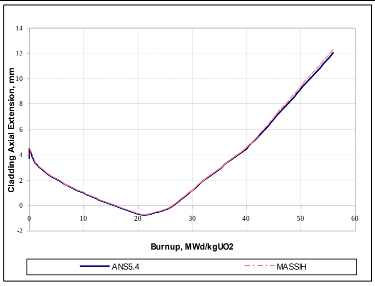

Figure 5 Cladding Axial Extension

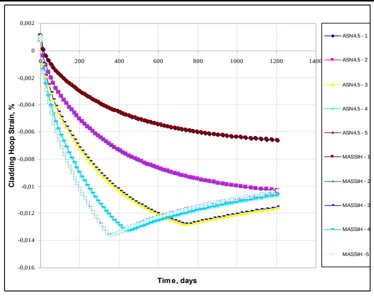

Figure 6 Cladding Hoop Strain

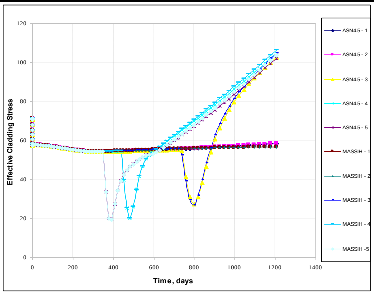

Figure 7 Effective Cladding Stress

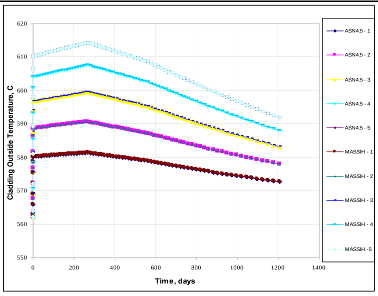

Figure 8. Cladding Outside Temperature

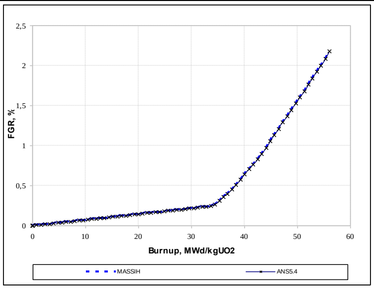

Figure 9.  FGR

## V. Conclusions

According to [1], Successes criteria's for the first safety barrier of the fission products are as follow:

1. Maintaining fuel cladding integrity - in states of category 1 (steady and transient states during normal operation) and category 2 (anticipated operational occurrences, with frequency above $10^{-2}$ events per year)
2. Maximum fuel cladding temperature shall not exceed 1200 degrees centigrade, the local oxidation of the cladding shall not exceed 17% of the initial thickness, and the reacted amount of zirconium shall not exceed 1% of its mass in the reactor core, in cases of loss of coolant accidents of category 3 (accidents of low frequency of occurrence, in the range between $10^{-2}$ and $10^{-4}$ events per year) and category 4 (design basis accident of very low frequency of occurrence, in the range between $10^{-4}$ and $10^{-6}$ events per year).

At present, the deterministic analysis use only successful criteria related to the maximum fuel cladding temperature shall not exceed 1200 degrees centigrade.

The new fuel performance code, FRAPCONE-3 is able to calculate cladding oxidation, i.e. to quantity all successes criteria's.

Fuel centerline temperature is one of the most important parameter to indicate fuel performance (Figure 3). The temperatures at the top and bottom nodes are about 900 C. In all cases, temperatures are considerably below the melting temperatures of fuel material. The results of low fuel temperatures is the observation of relatively low FGR – 2,17 % (Figure 9). The maximum gas pressure observed during the operation is about 8.03 MPa (Figures 1 and 4). Since cladding outside temperature only up to 600 C (Figure 8), it does not promote excessive oxidation. The thickness of oxide layer is very low compared to an acceptable safety limit of 17% of the total cladding thickness.

The results of performed calculation shown that fuel rod failure is not observed during normal operation and all fuel rod parameters investigated in this paper are found to be within the safety limits.

**Authors: M.Sc. Marina Andreeva Kocheva – Technical University of Sofia, PhD student of division „Thermal and Nuclear Power Engineering", Phone: 965 23 61, e-mail: marina.andreeva@mail.bg;**

**Dr. Totju Totev - Argonne National Laboratory, USA, Phone: 1-630-252-5161**

**Assoc. Prof. Dr. Stoyan Stoyanov – Bulgarian Nuclear Regulatory Agency, Phone: 940 68 31, e-mail: S.Stoyanov@bnra.bg;**

## VI. Използвана литература

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