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Verification of TVS-2006 fuel rod design of VVER-AES2006 reactor under steady-state operating condition Using FRAPCON-3.5 code

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Abstract: The purpose of this paper is to discuss the independent verification of TVS-2006 fuel rod design used in VVER-AES2006 reactor (Novovoronezh NPP-2 Power, Unit 1), based on the acceptance criteria and the reference data given in the Preliminary Safety Analysis Report of the State Research, Design, Construction and Survey Institute “Atomenergoproekt” (PSAR) and the operation of VVER-1000 reactor. The calculations were performed using FRAPCON-3.5 code, including fuel temperature, cladding temperature, fission gas release, internal gas pressure, cladding stress and strain, fuel extension, fuel rod elongation, cladding creep rate, fuel swelling rate, cladding oxide thickness and hydrogen concentration. The results are compared with the calculated data using START-3 code in PSAR and the acceptance criteria required by Russian nuclear regulatory body. Despite some discrepancies, the results showed conformance with the calculated data given in the PSAR and meet the acceptance criteria.

Keywords: *Nuclear fuel rod design, fuel behaviour, design verification, acceptance criteria.*

I. INTRODUCTION

From the 80s of the 20th century to the present, the fuel rod design has been continuously improved to optimize the fuel rod behaviour and meet the higher operating conditions of reactors, such as the high-power level (1000-1600 MWe), power uprate up to 110%, increased burn-up (60-70 GWd/tU) and extended fuel cycles (from 12 to 18 months). Thus, more realistic predictions of fuel performance is needed to allow operating Nuclear Power Plant effectively and safely, as well as improving operating margins and efficiency and higher flexibility in fuel management. So, a reliable prediction of fuel rod behaviours is important for fuel rod design and safety evaluation in nuclear power reactors [1]. While the fuel rod design is performed by the vendors using their own codes, the utilities and the safety authorities also need to perform independent design verification using licensing fuel rod codes such as FRAPCON-3.5, COPERNIC, TREQ, PAD codes.

FRAPCON-3.5 code [2, 3], one of fuel performance codes verified and licensed by United States Nuclear Regulatory Commission (US.NRC) to review fuel design of Light Water Reactor (LWR), is designed to perform the thermal-mechanical calculations of LWR fuel rod such as the temperature, pressure, and deformation as functions of time-dependent fuel rod power and coolant boundary conditions under steady-state

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condition, and to generate initial conditions for transient fuel rod analysis using the FRAPTRAN-1.5 code [4, 5]. The FRAPCON-3.5 code uses data of material properties documented in the updated version of the MATPRO material properties package for high burn-up conditions and advanced cladding alloy such as Zircaloy-2, Zircaloy-4, ZIRLO™, M5,... The main models of FRAPCON-3.5 code used in the calculations include the FRACAS-I thermal-mechanical model, Forsberg-Massih fission gas release model and Cladding oxidation and hydrogen content models.

Until now many features of FRAPCON-3.5 code have been improved to be used in the independent review and safety analysis of fuel rod design, as well as in the operational and licensing supports by some authorities as US.NRC, AREVA NP, Inc. (USA), IRSN (France), ALVEL, NRIR (Czech Republic), CRCD (Ukraine), KEPCO (Korea), NRA (Japan), Tractebel Engineering S.A. (Belgium)…Although some calculations of fuel rod behaviour of VVER-440/VVER-1000 reactors have been performed by FRAPCON-3.5 code but not yet applied for fuel rod of VVER-1200 reactor (VVER-AES006). Therefore, the independent verification of fuel rod design of VVER- AES006 reactor (Novovoronezh NPP-2 Power Unit 1), TVS-2006 fuel rod [6], has been chosen as the main objective of this research. The obtained results, using the FRAPCON-3.5 code, are also compared with the calculated data by START-3 code in PSAR and the acceptance criteria. In which, the acceptance criteria using in evaluation are usually established by the fuel vendors based on experimental data and theoretical considerations, with adequate margins that are accepted by the Russian Safety Authority.

II. CALCULATION MODEL FOR TVS-2006 FUEL ROD

A. Description of TVS-2006 fuel rod design

The TVS-2006 fuel rod design of VVER-AES006 reactor is developed on the basis of TVS-2 fuel rod design, which has been developed by EDB “Gidropress” FSUE for the commercial VVER-1000 reactor, using the design solutions, calculations as well as the experimental justification. A TVS-2006 fuel rod comprises the following parts: Upper plug, cladding, lower plug, fuel pellets and a spring (Fig. 1, Table I) [6].

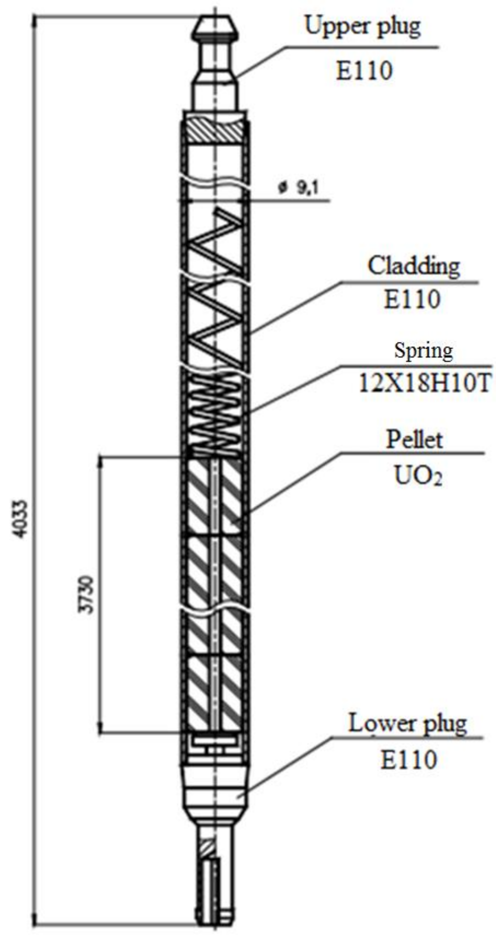

Fig. 1. Configuration of TVS-2006 fuel rod

Table I. Main parameters of TVS-2006 fuel rod

Parameter	Value
Number of fuel rods in fuel assembly	312
Fuel rod lattice	Evenly triangular
Fuel rods pitch, mm	12.75
Fuel	UO_2
Fuel density, kg/m^3	(10.4-10.7).10^3
Mass fraction of uranium isotopes mixture in fuel, %	≥87.9
Cladding material	E110 (Zr-1%Nb)
Total fuel rod length, mm	4033
Outer diameter of fuel cladding, mm	9.10 ± 0.04
Inner diameter of fuel cladding, mm	7.73 ± 0.06
Outer diameter of fuel pellet, mm	7.60 ± 0.03
Diameter of centreline hole in fuel pellet of a fuel rod, mm	1.2 ± 0.2
Grain size in fuel pellet, µm	10-20
Fuel pellet height, mm	9.0-12.0
Fuel column height (cold state), mm	3730
Fuel mass in fuel rod, kg	1.712
Average linear power, W/cm	167.8
Peak linear power, W/cm	420
Maximum cladding temperature, ^{o}C	355
Enrichment U^{235} (maximum value), %	4.95 ± 0.05

B. Modelling method

The TVS-2006 fuel rod has been modelled using the FRAPCON-3.5 code based on the design parameters, reference data in the operation of VVER-1000 reactor [3, 5, 7, 8, 9, 10, 11] and the given data of PSAR [6]:

- The dimensions for TVS-2006 fuel rod were taken from design data. The fuel rod was divided into 50, 17 and 45 for number of equal-length axial nodes, radial boundaries in the pellet and equal-volume radial rings, respectively;

- The initial fill pressure of fuel rod, coolant pressure, coolant inlet temperature, and mass flux of coolant were all taken from design data (Table II) [6].

- The cladding of TVS-2006 fuel rod is made of E110 alloy, however, properties of E110 alloy is not modelled in FRAPCON-3.5 code. Therefore, M5™ alloy is selected instead of E110 because it has the similar chemical composition as E110 alloy (Zr-1%Nb) [7];

- Calculations have been performed for 4 fuel cycles, the length of each cycle is 343.2 Effective Full Power Days (EFPD), using the power history taken from given data of PSAR and reference data in the operation at VVER-1000 reactor (Fig. 2) [6, 7, 10, 11]. Conservative axial power distribution in the hottest fuel rod was used for evaluating maximum temperature of fuel and cladding (Table III) [6].

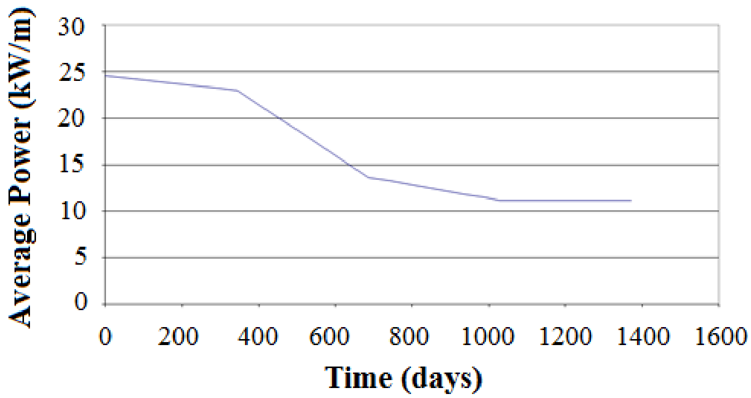

Fig. 2. The linear heat generation rate of the fuel rod during operation

Table II. Main parameters of the boundary conditions

Parameter	Value
The rod initial fill pressure, MPa	2.1
Coolant system pressure, MPa	16.2 ± 0.3
Coolant inlet temperature, °C	298.2 ± 4
Mass flux of coolant, kg/s.m^2	3930

Table III. Conservative axial power distribution (K_z) in the hottest fuel rod

Parameter	Value										
Core height, %	5	15	25	35	45	50	55	65	75	85	95
K_z	0.5	0.83	1.07	1.25	1.35	1.357	1.35	1.25	1.07	0.83	0.5

C. Design verification method

The verification of fuel rod design is performed by the deterministic calculations using FRAPCON-3.5 code and also combines with uncertainty evaluation by the statistical method using the Root Mean Square. The sources of uncertainties include operation parameters (power histories, flow rate, pressure…), manufacturing parameters (cladding thickness, pellet diameter, fuel density…) and key correlations (fuel thermal conductivity, densification, swelling, fission gas release, cladding creep, corrosion, hydrogen pickup). Best estimate calculations of the design parameter with nominal input data and best estimate models are performed by FRAPCON-3.5 code. Then, the assessment of design parameter sensitivities and quantification of design uncertainties via the Root Mean Square method are used to calculate the maximum values of the calculation results, to be compared with the acceptance criteria and the calculated data by START-3 code in PSAR [6, 12].

III. CALCULATION RESULTS

A. Thermal-mechanical calculation results

The results of thermal-mechanical calculations are given in Table 4 and Figs. 3-6 (nominal values), including: Fuel temperature, cladding temperature, fission gas release, and internal gas pressure. The fuel rod temperature distribution depends on design parameters, materials properties and on many phenomena which develop during irradiation. Many properties are exponentially dependent on temperature.

The results of fuel temperature calculations show that the fuel temperature (T_f) reaches its maximum T_{fmax} = 1746.37K at the beginning of the first cycle of operation and is lower than the limit value [T] = T_{melt} = 3113.14K with the safety margin K = 1.78. The maximum of average fuel temperature in four cycles is 1020.25K. For cladding temperature, the maximum cladding outside temperature (T_c) is 625.19K at the beginning and at the end of the first cycle of the operation and does not exceed limit value of 628.15K. These values are close to calculated data by START-3 code in PSAR (maximum fuel and cladding outside temperature are 1860.15K and 627.25K, respectively), also meet acceptance criteria and protect the fuel against excessive degradation of cladding mechanical properties related to hydrogen pickup or accelerated oxidation (high cladding surface temperatures). The calculation results of fuel rod temperature show the guarantee of design in order to protect the fuel against any types of failures resulting from fuel melting or overheating. Therefore, accurate temperature estimates are important for many safety design criteria.

Fission gas release (FGR) and rod internal pressure (P_i) have a major impact on mechanical properties of fuel rod. Fission gas release can cause fuel swelling, pressure build-up (xenon, krypton), pellet-cladding mechanical interaction, stress corrosion cracking… So, the excessive fission gas release can cause the rod pressure to rise beyond system pressure and lead to fuel damage. Thus, rod pressure need to be limited by safety criteria and must be calculated for the design evaluation.

Maximum fission gas release of fuel rod (FGR) is 3.58% at the end of 4th cycle and close to calculated FGR by START-3 code (~3%). Maximum rod internal pressure is 5.69MPa during four cycles of operation with the safety margin K = 2.85. The calculation results of FGR and internal pressure show the guarantee of design in order to protect the fuel against cladding lift-off. These results are lower than the limit values and show that they ensure to prevent the diametral gap between the fuel and the cladding from re-opening during steady-state operation, which causes ballooning and affect the coolant flow or the local overheating of the cladding.

As above analyses, the thermal-mechanical calculations have demonstrated that the results are clove to calculated data by START-3 code in PSAR, satisfy acceptance criteria and also show adequate thermal-mechanical reliability of TVS-2006 fuel rod in operation. However, the rod internal pressure is quite low, which the reason may be due to the insufficient information of the design power histories and the modelling method by START-3 code. It was found that the average variance in rod internal pressure value for a biased similar PWR fuel rod by adjusting the steady-state power by ±10% is 20%. Also, the average value of rod internal pressure varied by approximately 32% when the fuel thermal conductivity model is biased by ±0.5W/m-K [12].

Besides, as shown in Table 4, the internal pressure presented in the PSAR seems to be too high with regard to experience feedback and with longer fuel length in the design, it must be around 10 to 12 MPa after four cycles of irradiation instead of 15.2 MPa (from operational feedback measurements and calculations on irradiated fuel rods) [12,13]. Thus, it should be verified with additional calculations with appropriate power history.

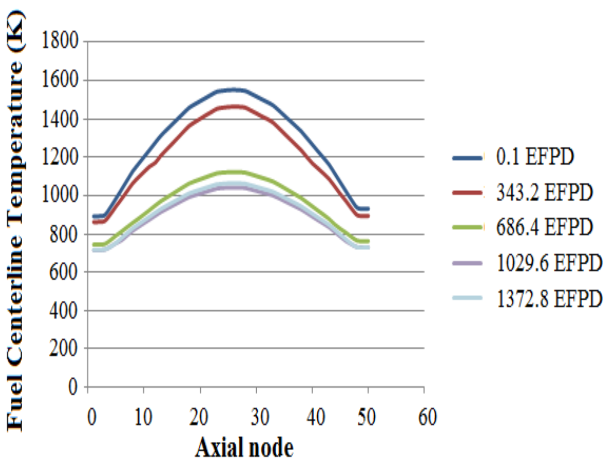

Fig 3. Fuel centreline temperature

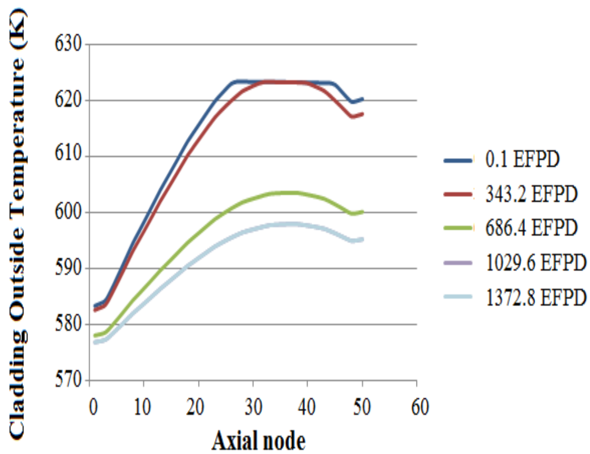

Fig 4. Cladding outside temperature

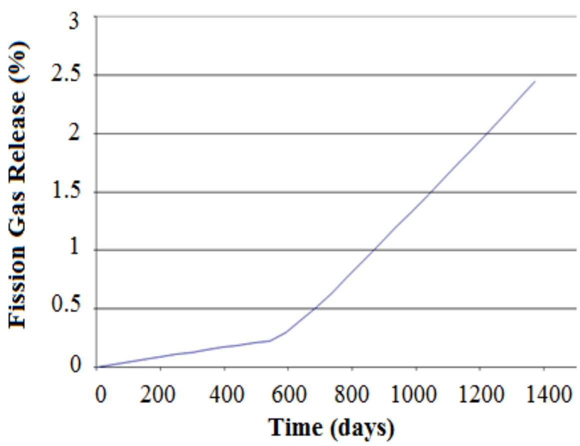

Fig 5. Fission gas release

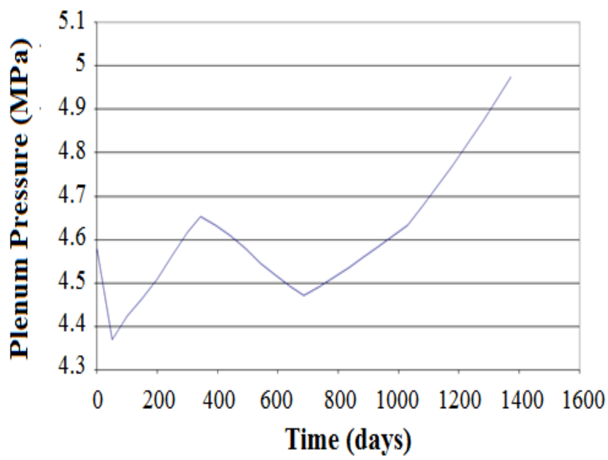

Fig 6. Rod internal pressure

Table IV. Results of thermal-mechanical calculations

Parameter	Nominal Results	Uncertainty	Maximum Results	PSAR data	Deviation, %	Limit value	Safety margin, K*	Standard safety margin, [K]
T_f, K	1551.7	194.67	1746.37	1860.15	-6.12	3113.14	1.78	1.1
T_c, K	623.38	1.81	625.19	627.25	-0.33	628.15	1.005	-
FGR, %	2.44	1.14	3.58	~3	-	-	-	-
P_i, MPa	4.97	0.72	5.69	15.2	-167.14	16.2	2.85	1.1

**Safety margin K = Limit value/Maximum value*

B. Strength calculation results

The strength calculation results of cladding are given in Table 5 and Figs. 7-10 (nominal value). The operating experience of fuel rods as well as calculations and experiments show that hoop stress and strain determine cladding strength in steady-state conditions and during transients, that is why they will be in the focus of further strength analysis.

In the beginning of the cycle, hoop stress on the internal cladding surface are mainly determined by thermal gradient and external differential pressure. After closure of the radial gap between fuel and cladding, the fuel first comes into “soft” contact with cladding, and the contact becomes “hard” after crack healing in fuel. As a result, the hoop stress of the internal cladding surface increase first in the central and then in the side cross-sections of fuels rods. During four cycles of operation, the stress reaches a steady level of about 70-80 MPa and the maximum effective cladding stress (σ_{eff}) is 103.28 MPa. This value is lower than the yield stress of cladding material in irradiation conditions (340-350 MPa). The maximum cladding hoop stress (σ_h) is 95.49 MPa. Although the result meets design criteria but it is quite low than calculated data in PSAR (190.4 MPa) because E110 alloy is modelled exactly by FRAPCON-3.5 code and for the calculations, M5™ alloy was assumed since it has the same chemical composition as E110 alloy (Zr-1%Nb) [7].

For the cladding hoop strain (ε_h), the maximum value is 0.20% with the safety margin K = 2.5. For the cladding elastic strain, the maximum values are 0.06%, 0.09% and 0.0629% for hoop strain (ε_{eh}), axial strain (ε_{ea}) and radial strain (ε_{er}), respectively. These results of strain show ability to protect the fuel against pellet-cladding interaction (PCI) failure. The intent of these analyses is to ensure integrity of cladding due to slow rate strain accumulation at which the stress does not reach the stress limit (yield stress). The calculation results of stress and strain show that they satisfy acceptance criteria and steady-state operating conditions. However, it is noticed that the stress is not an adequate criterion for fuel failure since it cannot be measured during irradiation and these calculations should be considered in the Condition II transients of ramp power.

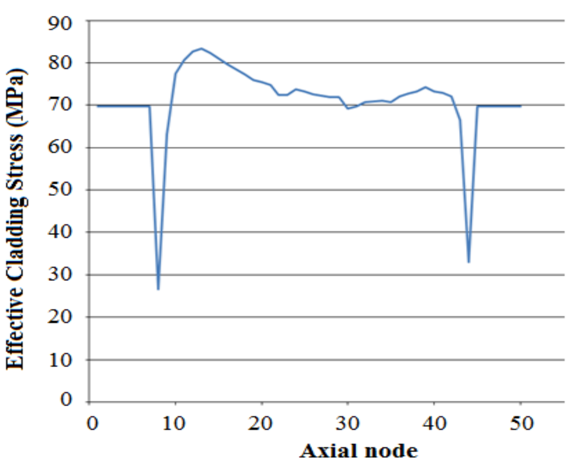

Fig 7. Effective cladding stress

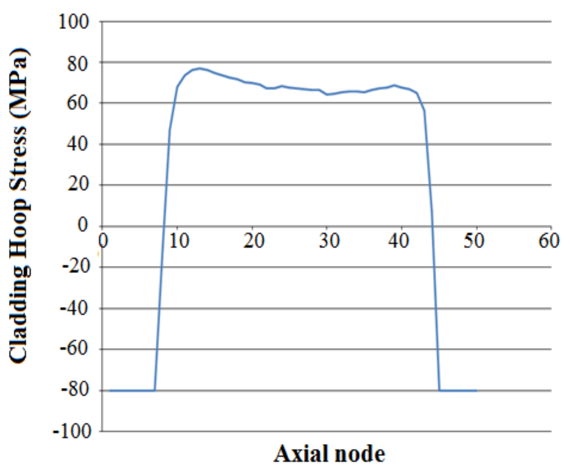

Fig 8. Cladding hoop stress

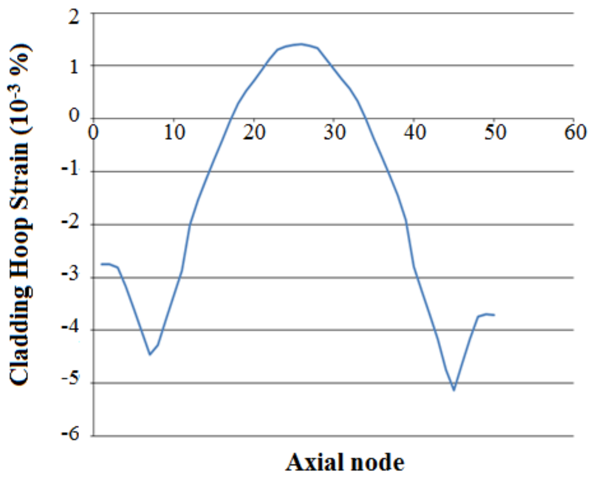

Fig 9. Cladding hoop strain

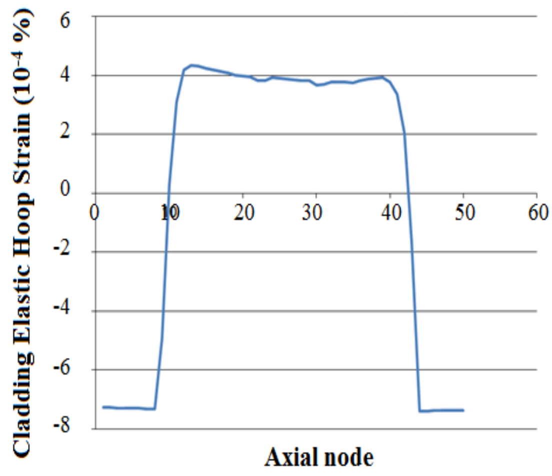

Fig. 10. Cladding elastic hoop strain

Table V. Results of strength calculations

Parameter	Nominal Results	Uncertainty	Maximum Results	PSAR data	Deviation, %	Limit value	Safety margin, K*	Standard safety margin, [K]
σ_{eff}, MPa	83.32	19.96	103.28	-	-	350	3.39	-
σ_h, MPa	76.99	18.50	95.49	190.4	-49.8	230	2.41	1.2
ε_h, %	0.14	0.06	0.20	-	-	0.5	2.5	-
ε_{eh}, %	0.05	0.01	0.06	-	-	-	-	-
ε_{ea}, %	0.07	0.02	0.09	-	-	-	-	-
ε_{er}, %	0.06	0.0029	0.0629	-	-	-	-	-

Table VI. Results of deformation calculations

Parameter	Nominal Results	Uncertainty	Maximum Results	PSAR data	Deviation, %	Limit value	Safety margin, K*	Standard safety margin, [K]
ΔH, mm	3.18	1.84	5.02	-	-	-	-	-
ΔL, mm	18.31	11.92	30.23	47.7	-36.62	61.6	2.04	-
V_{creep}, 10^{-11} m/m/s	5.78	0.70	6.48	-	-	-	-	-
V_{swell}, 10^{-11} m/m/s	7.63	1.01	8.64	-	-	-	-	-

C. Deformation calculation results

The results of deformation of fuel rod are given in Table 6 and Figs. 11-13 (nominal value), including: Fuel stack axial extension, fuel swelling rate, fuel rod elongation, and cladding creep rate.

Fuel rod length changes due to irradiation effects and differential thermal expansion shall cause interference with the fuel assembly structure. This evaluation is a critical design input because it determines the assembly length (fuel assembly mechanical design). The analyses have to show ability to prevent cladding from axial buckling or overstressing of the thimble tubes and/or thimble-to-nozzle connections.

The results show that during four cycles of operation, maximum fuel stack axial extension (ΔH) is 5.02 mm, and maximum fuel rod elongation (ΔL) in operating conditions is 30.23 mm. The result meet design criterion but it is lower than calculated data in PSAR (47.7 mm) because in the calculations, M5™ alloy was assumed since it has the same chemical composition as E110 alloy (Zr-1%Nb) [7]. Taking into consideration the thermal elongation and irradiation growth (about 0.15 %) of fuel assembly skeleton, the clearance between the upper fuel rod plugs and the fuel assembly head in a hot state is about 61.6 mm without fuel rod elongation. This value is considered the limit value. Thus, the calculated safety margin for fuel rod elongation is K = 2.04.

The cladding can be free standing at beginning of life (before densification) and no long-term buckling. However, fuel cladding is prone to instant collapsing when reaching critical pressure for this cladding state as well as to long-term accumulation of creep deformations and fuel swelling. Therefore, fuel cladding has to be ensured against cladding creep collapse (axial slip failures). The results show that the maximum cladding creep rate (v_{creep}) is 6.48 (10^{-11} m/m/s) and maximum fuel swelling rate (v_{swell}) is 8.64 (10^{-11} m/m/s), this ensures no failure of cladding.

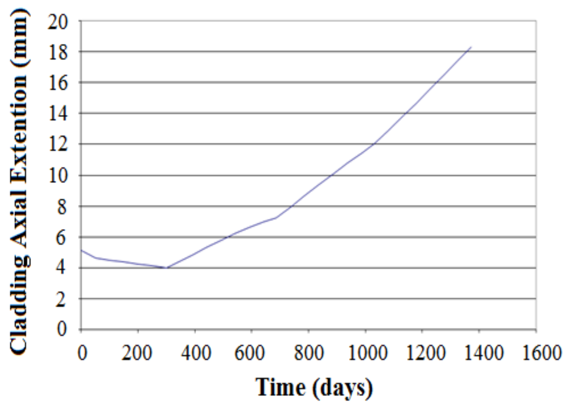

Fig 11. Cladding axial extension

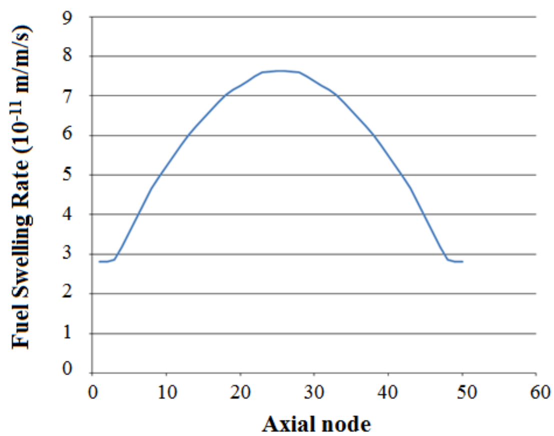

Fig 12. Fuel swelling rate

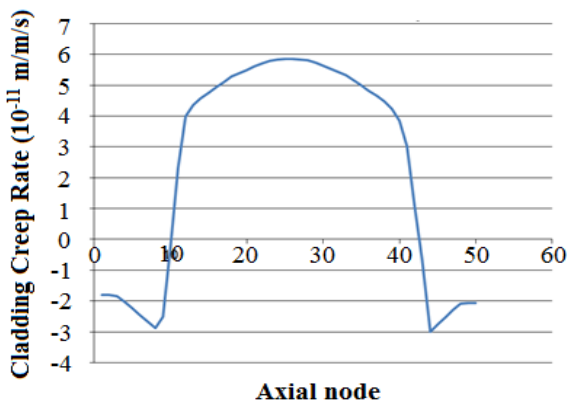

Fig 13. Cladding creep rate

D. Cladding oxidation and hydration calculation results

The results of oxide thickness and hydrogen concentration of cladding are given in Table 7 and Figs. 14-15 (nominal value). Oxidation and hydriding under normal operating conditions of reactor directly impact fuel performance, not only during normal operation, but during transients and accidents as well. Cladding corrosion reduces the effective thickness of the cladding, decreases the effective thermal conductivity of the cladding and thus increases the cladding and fuel temperatures and also reduces effective cladding-to-coolant heat transfer. Hydrogen absorption by the cladding and subsequent formation of hydrides may lead to cladding embrittlement. These phenomena are increasingly important at higher exposures. So, the analyses have to show ability to protect the fuel against any type of cladding corrosion induced failure.

The results of surface corrosion and cladding hydration calculation show that maximum oxide thickness is 20.21μm and maximum hydrogen concentration is 73.42 ppm with the calculated safety margins are 2.97 and 5.45 for surface corrosion and cladding hydration, respectively. The results meet design criteria but it is lower than calculated data in PSAR (oxide thickness 30 μm). This deviation can be due to assuming M5™ alloy instead of E110 alloy [7]. The calculation results have showed that cladding of TVS-2006 fuel rod can meet operating ability in normal condition of reactor for the cladding oxidation and hydration.

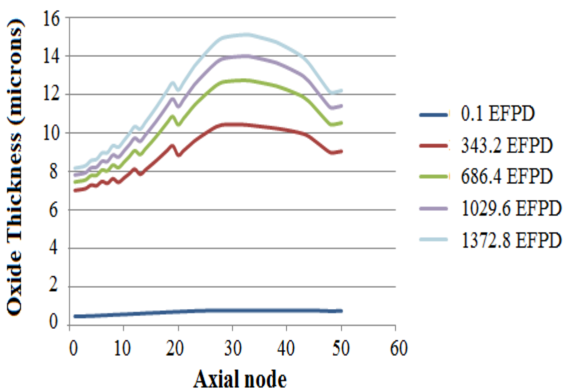

Fig 14. Cladding oxide thickness

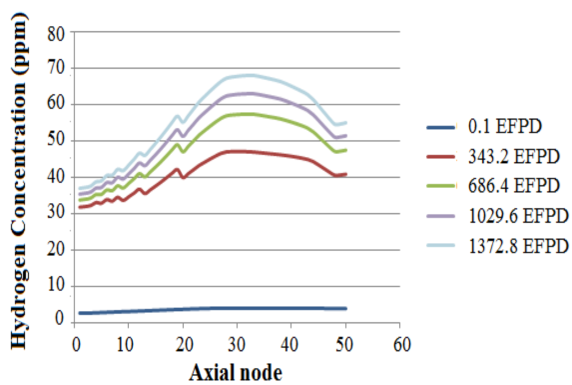

Fig 15. Claddinghydrogen concentration

Table VII. Results of oxide thickness and hydrogen concentration calculations

Parameter	Nominal Results	Uncertainty	Maximum Results	PSAR data	Deviation, %	Limit value	Safety margin, K*	Standard safety margin, [K]
Oxide thickness, μm	15.11	5.10	20.21	30	-32.6	60	2.97	1.5
Hydrogen concentration, ppm	68.05	5.37	73.42	60-80	-	400	5.45	-

IV. CONCLUSIONS

The independent verification of the TVS-2006 fuel rod design of VVER-AES2006 reactor under steady-state operating condition using FRAPCON-3.5 code was performed based on the design parameters and the reference data from the operation of the VVER-1000 reactor. The calculation results are compared with acceptance criteria and the obtained data by START-3 code in the PSAR.

It has been indicated that the calculation results show conformable tendency of the operating behaviors, as well as satisfying the operational ability in normal condition of reactor and are also close to obtained data by START-3 code in the PSAR.

The calculation values by FRAPCON-3.5 code are lower than the limit values of acceptance criteria and safety margins are greater than standard safety margins.The deviations between calculation results of FRAPCON-3.5 code and START-3 code may be due to the insufficient information about the design power histories and the modelling method using START-3 code in the PSAR, as well as assuming M5™ alloy (similar Zr-1%Nb alloy) instead of E110 alloy due to FRAPCON-3.5 code does not model exactly characteristics of E110 alloy.

However, it was found that the average variance in the calculation parameters for a biased similar PWR fuel rod by adjusting the steady-state power by ±10% is 20%.

Additional, when the calculations have been performed on two computing systems of the Vietnam Atomic Energy Agency (VAEA) and Tractebel Engineering (TE, GDF Suez, Belgium) with the same input, it has been indicated that the calculation results are similar. This shows the reliability and the realistic meaning of the used tools at VAEA such as computer server and FRAPCON-3.5 code version. Also, this is the first calculation result using FRAPCON-3.5 code in order to verify the TVS-2006 fuel rod design of VVER-AES2006 reactor.

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