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Using FRAPTRAN 1.5 code for safety evaluation of TVS-2006 fuel rod under transient conditions of VVER-1200 (AES-2006) reactor

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Abstract: This article shows the results of important thermal-mechanical parameters related to the TVS-2006 fuel rod design which were analyzed and evaluated by using FRAPTRAN1.5 code. Based on the data given in the Preliminary Safety Analysis Report (PSAR) of AES-2006 (Novovoronezh NPP-2 Power Unit No.1)and FRAP-T (Fuel Rod Analysis Program-Transient), FRAPTRAN1.5 calculations of TVS-2006 fuel rod behaviors in Loss of Coolant Accidents and Reactivity-Initiated Accidents conditions have been made. The calculated results related to safety criteria of fuel element and cladding temperatures, cladding stress and strain, fuel enthalpy, local oxide thickness, gap gas pressure, and elongation of fuel rod have been compared with the ones given in AES-2006 PSAR. A good agreement has been observed between AES-2006 PSAR andFRAPTRAN1.5 calculations.

Key words: *Thermal-mechanical, fuel rod, LOCA, RIA, Safety evaluation, VVER, AES-2006.*

I. INTRODUCTION

For a nuclear power project, fuel rod safety evaluation of a reactor is one of the important requirements which shall be carried out by the governmental agency and/or a technical support organization (TSO). In Vietnam, many related governmental agencies (including Vietnam Atomic Energy Agency (VAEA)) and institutes have conducted research projects (including ministerial and national projects) on characteristics and behaviors of fuel elements and fuel rods in steady-state and accident conditions. In those projects, computer codes are necessary tools. One of them is FRAPTRAN1.5code which is the useful tool developed to calculate and assess thermal-mechanical behaviors of fuel rods in the accident conditions at high burn-up [1,2]. Physical models related to the thermal-mechanical aspects of the fuel rods have been described in details by K.J. Geelhood *et al* [1, 2].

Characteristics of TVS-2006 fuel rod have been determined by FRAPTRAN1.5 code based on the data given in the AES-2006 PSAR [3]. The thermal-mechanical characteristics and behaviors of the TVS-2006 fuel rods have been calculated and assessed for the following accident scenarios:

- Large break loss of coolant (LB-LOCA) in the result of breaking primary inlet pipeline with equivalent diameter exceeding 100 mm;

- Control Rods (CPS CRs) Ejection at Power in case of Drive Housing Rupture (Reactivity Initiated Accident - RIA).

The obtained results have been analyzed to assess the safety of the fuel rods based on design criteria of AES-2006 [3].

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II. CALCULATION METHOD AND FRAPTRAN1.5 MODELING FOR TVS-2006 FUEL ROD

A. Calculation method

With regard to the design criteria on thermal-mechanical behaviors of fuel rods, the following methods have been adopted:

- Modeling of the fuel rod TVS-2006 has been determined on the basis of data given in AES-2006 PSAR;

- Fuel rod parameters which were calculated by FRAPTRAN1.5 code have been verified, in comparison with the design criteria, taking account to the fabrication tolerance, uncertainties associated to modeling and experimental feedback data. Design safety criteria used in the verification were as follows:

• Internal pressure: During normal operating condition, internal pressure must be lower than 16.2 MPa which would lead to loss of the integrity of fuel rods, dimensional instability or degradation of the thermal transfer. In addition, during the fuel rod irradiation in reactor core, the internal pressure induced to the cladding shall remain lower than the coolant pressure, otherwise it will lead to re-opening of the pellet-cladding gap under single action - the so-called “lift-off” of fuel rods;

• Cladding oxide thickness must be lower than 60-70 µm;

• Temperature of fuel pellet must be lower than 2540°C;

• The instantaneous deformation of the cladding (strain rate) due to the local variation of linear power must be lower than 1%;

• Fuel rod axial growth must be lower than 50 mm to prevent bowing phenomenon which could affect the DNB (Departure from Nucleate Boiling) criteria.

B. Modeling TVS-2006 fuel rod

- Nodalization of fuel rod model:

FRAPTRAN1.5 code uses the solution of the nodalization in building the fuel rod model. The TVS-2006 fuel rod model has been developed from stack fuel model, which was divided into 12 axial nodes and 15 radial nodes, as shown in Figure 1. The axial nodalization data specify elevations at which the radial distributions of the fuel rod variables are calculated. Each of these elevations is defined as an axial node. The first axial node is at the bottom of the fuel rod.

Similarly, the radial nodes lie in planes that pass through the axial nodes and are perpendicular to fuel rod axis which is the centerline of fuel rod. The first radial node is at the center of the fuel rod. Other radial nodes are placed at the fuel pellet surface and at the cladding inside and outside surfaces. In addition, an arbitrary number of radial nodes can be placed within the fuel and cladding. Unequal spacing of the radial nodes in the fuel is permitted, and the default situation is a spacing that results in equal-area rings of fuel. Finally, plenum model has been made based on the defined upper volume of fuel rod, and the spring model has been determined by defined outer-diameter, turns and length of fuel-rod spring.

- Modeling options:

• FRAPCON restart file used for initialization of FRAPTRAN model;

• Cladding type: E110;

• Fission gas release model: Massih model (default);

• Fuel-clad deformation model: FRACAS-I rigid pellet model (default);

• Clad ballooning/burst model: BALON2 failure model (default);

• High temperature oxidation model: Cathcart-Pawel model (C-P);

• Heat option: to specify a central void in the fuel pellets.

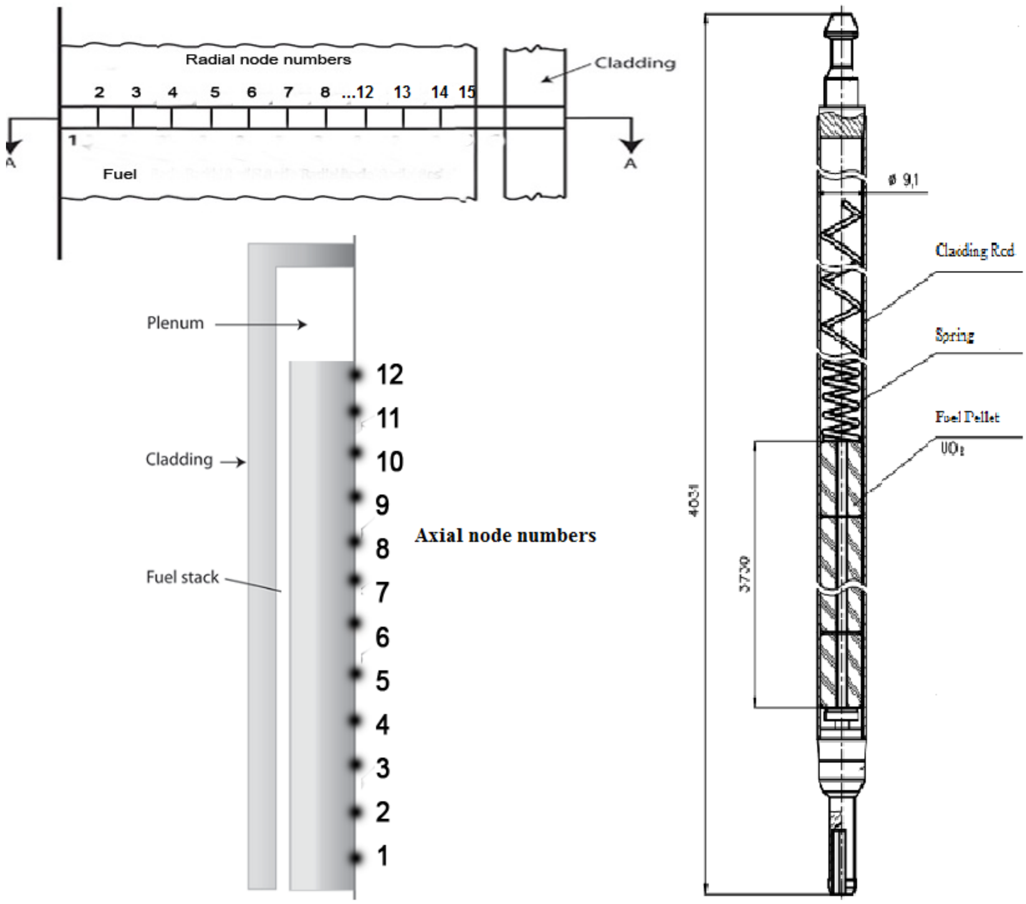

Fig.1. FRAPTRAN1.5 modeling of TVS-2006 fuel rod

III. SAFETY EVALUATION OF TVS-2006 FUEL ROD

A. Initial conditions

The plant nominal initial operating conditions have been used as follows:

- Maximum linear heat generation rate corresponding to nominal power: 420W/cm;

- Coolant pressure: 16.2 MPa;

- Coolant inlet temperature: 298.2^oC;

- Coolant inlet flow rate: 86,000 m^3/h;

- Maximum hydrogen concentration, maximum fuel enthalpy and maximum cladding oxide thickness were 70.9 ppm, 400 J/g (~ 96cal/g) and 15.2 µm, respectively;

- The axial power profile has been given in the PSAR;

- The radial power profile was value obtained fromFRAPCON3.5 output as in the initialization file.

B. Analysis of the results

1. Large break loss of coolant in the result of breaking primary inlet pipeline with equivalent diameter exceeding 100 mm

The necessary digitalized data, including the linear heat generation rate, coolant pressure, coolant inlet temperature, coolant inlet flow rate, power profile, etc, in AES-2006 PSAR [3] are imposed in the FRAPTRAN1.5 model in order to calculate the fuel thermal-mechanical behaviors during this accident. The important characteristics were calculated with 12 axial-node-stack fuel rod model, in which maximum values are defined on the basis of the parameters of axial nodes, and for a period of 500 seconds.

The obtained results are given in Figures 2-8.The evaluation was carried out by comparing the parameters' maximum values obtained from computational analysis with the safety criteria (in Table I).

For the fuel pellets, failure will occur if fuel centerline melting takes place. The analysis was performed for the maximum linear heat generation rate anywhere in the core, especially in hot channel factors. If the fuel centerline melting occurred, it should be assured that axial or radial relocation of the molten fuel would be neither allowed to contact the cladding nor produce local hot spots. The fuel centerline temperature transient was calculated and shown in Figure 2. The obtained results showed that local melting does not occur due to the fact that during the entire emergency process, the maximum temperature of fuel element (FE) was 1877ºC (2150 K, as shown in Figure 2), which is much lower than the fuel melting temperature (2540°C for burn-out fuel and below 2840°C for fresh fuel), i.e. the acceptance criterion was met.

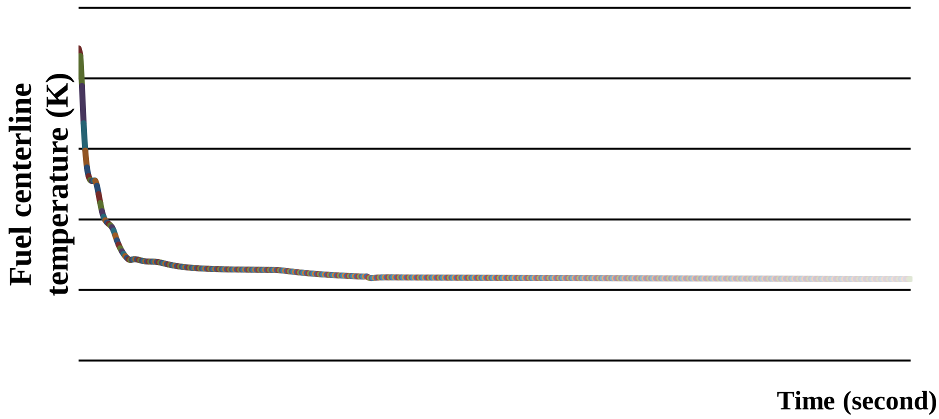

Fig.2. Fuel centerline temperature

For the cladding of the fuel rod, at the beginning of the accident, cladding temperature rapidly increased due to the apparition of the DNB which induced the decrease in coefficient of the heat exchange between cladding and fluid. During this period of time, the temperature at centerline of fuel pellets continued to decrease due to the stopping of nuclear reactions. Consequently, the fuel pellets retracted while the cladding dilated. Thus, there was an augmentation of the gap between cladding and fuel pellets which led to an increase of the corresponding thermal resistance.

The obtained results showed that the cladding temperature was firstly increased and then decreased in accordant with the coolant pressure. This phenomenon is reasonable due to the reactor shutdown and the cooling by Emergency Core Cooling System. Moreover, the increase of the cladding temperature was accelerated from 820°C - 850°C(~1100 K) at which the zirconium-water reaction became consequence. This increase was slowdown when water reached the bottom of fuel rods, and then the emulsion ameliorated sensibly with coefficient of the heat exchange between cladding and coolant. The cladding temperature increased slower and then decreased slowly until the rewetting reached to the considered level. The cladding maximum average-temperature of the most heat-stressed fuel element in the accident was approximately 727°C (~ 1000K, as shown in Figure 3), which is not exceeding the acceptance criterion of 1200°C.

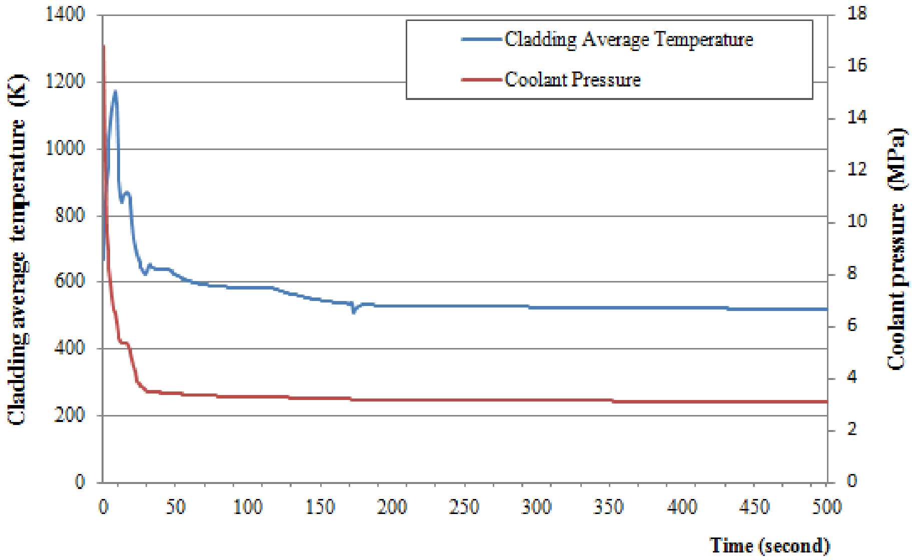

Fig.3. Cladding average temperature

The strain criterion is applied to the long-term strain that occurs after the closure of the pellet-cladding gap was induced by outer overpressure due to the creep-down phenomenon. This process includes the thermal expansion of the fuel pellets, and is governed mostly by the swelling, the creep and relaxation processes of the cladding and fuel pellets. The calculated results showed that the cladding maximum Hoop strain at the seventh node (about 2200 mm from the fuel stack bottom to top as shown in Figure 4), where it had the maximum temperatures of FE and cladding, was calculated as 0.62 %.This value is less than the general safety criterion of the USA and European countries (1%) but a little higher than the PSAR's safety criterion[3]. Besides, the cladding maximum stress was about 170 MPa (see Figure 5) which is smaller than the PSAR's safety criterion (230 MPa) [3]. The changes of these parameters were similar to the changes of the cladding temperature. This phenomenon is reasonable due to the temperature-strain-stress relationships.

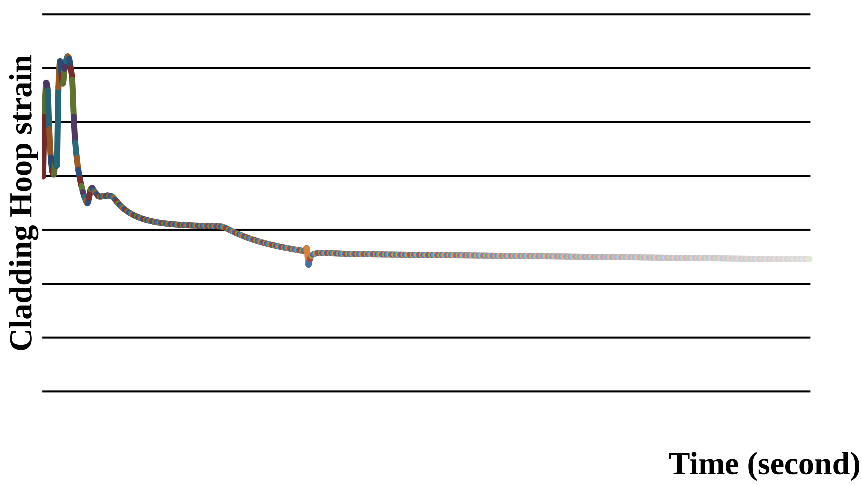

Fig.4. Cladding Hoop strain

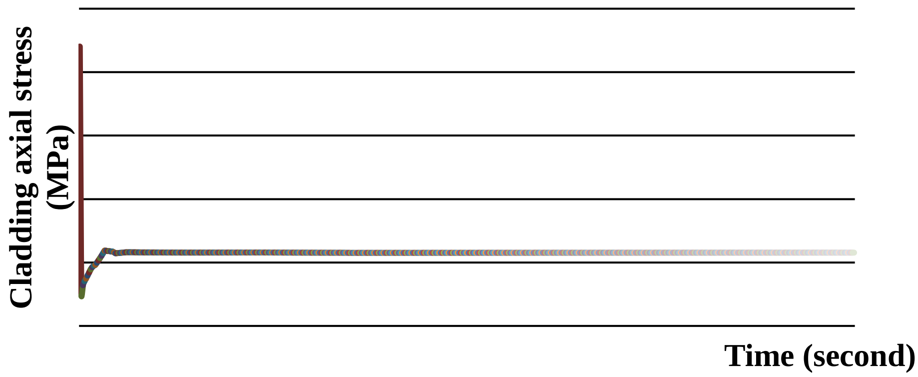

Fig.5. Cladding axial stress

In similarity with the cladding Hoop strain, the cladding has been elongated with the maximum elongation of about 39.5 mm (see Figure 6), not exceeding safety criteria of 50 mm [3], which obviously indicated that safety of cladding was well satisfied. Thus, the interaction of fuel rods with upper plate of fuel assembly has been avoided.

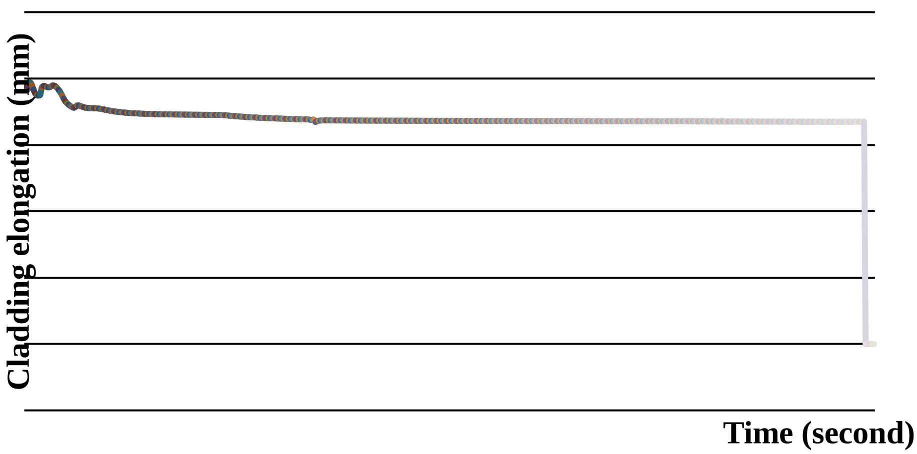

Fig.6. Cladding elongation

The thermal-mechanical characteristics of zirconium alloys (including the E110 alloy) are affected under irradiation. Due to the elongation associated to the creep and growth phenomena of fuel pellets and the cladding during irradiation, the limited elongation of the fuel rod must be assured in order to avoid the axial and radial deformations of the fuel assembly. This phenomenon could induce a "flux tilt" in the reactor core.

To assure safe operation of the reactor, fuel enthalpy averaged by fuel pellet cross-section shall not exceed 963 J/g for fresh fuel and for fuel with burn-up up to 50 MWd/kgU, and 691 J/g for fuel with burn-up higher than 50 MWd/kgU in any cross-section from bottom to top of fuel element [3]. According to computational data, the obtained maximum radial averaged fuel enthalpy was about 418 J/g with the maximum increase of the fuel enthalpy of approximately 18 J/g, as shown in Figure 7, and the initial maximum fuel enthalpy of 400 J/g given in *Section 1 Initial conditions*. This value is less than the fuel enthalpy limiting value, i.e. the acceptance criterion was satisfactorily met.

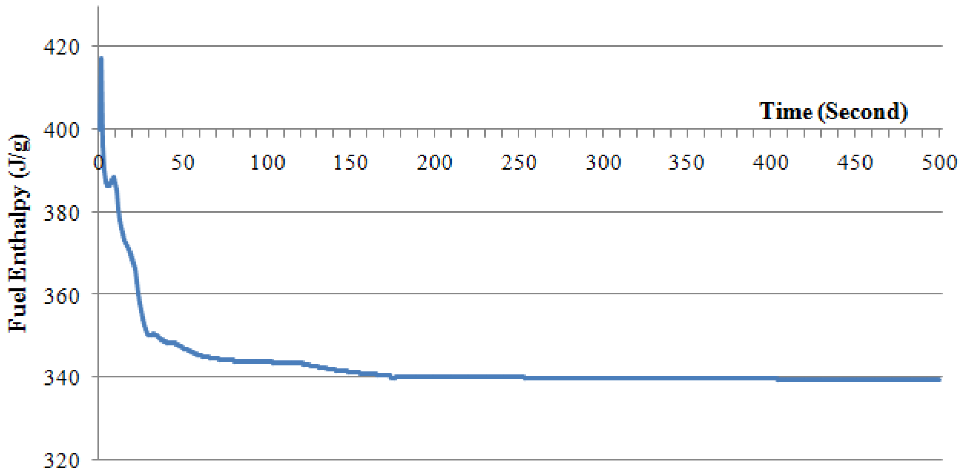

Fig.7. Fuel enthalpy

Table I. Maximum values of several parameters in the FRAPTRAN1.5 results for LB-LOCA

No.	Parameters	Calculated values	Safety criteria
1	Fuel centerline maximum temperature, °C	1877	2540
2	Cladding maximum average temperature, °C	727	1200
3	Cladding Hoop maximum strain,%	~0.59	1
4	Cladding maximum stress, MPa	~170	230
5	Fuel maximum enthalpy, J/g	420	963
6	Cladding maximum elongation, mm	39.5	56

The increase of gap gas pressure due to the fission products associated to cladding corrosion may increase the risk of cladding failure due to the embrittlement during the LOCA reflooding phase. This may also increase the tendency of cladding ballooning and relocation of fuel pellets. Therefore, the gap gas pressure must be smaller than the coolant pressure, which meet the safety criterion [3].The maximum obtained value of the gap gas pressure was about 7.9 MPa (see Figure 8a),not exceeding the coolant pressure. However, in the end of the accident, the gap gas pressure was found to be a little higher than the coolant pressure (3.2 MPa > 3.0 MPa), causing almost no effect on the integrity of the cladding rod. If the difference between the gap gas and the coolant pressure is higher 0.5-1.0 MPa after the refood (about 30s from the beginning of LOCA), it will affect the integrity of the cladding rod [3].

The final considered parameter is the local oxide thickness which has impact on durability of the cladding rod. The obtained results showed that the local oxide thickness reached the maximum value of about 21.45 µm (1.6%) with initial local oxidation depth 15.20 µm, the increases in the outer-diameter and inner-diameter oxide thickness were determined to be approximately 3.25 and 3.00 µm, respectively (see Figure 8b). These values were below values in the safety criteria (the oxide layer on the cladding internal surface shall not exceed 15 µm and the total local oxide thickness shall be under the safety limit of 18 %).

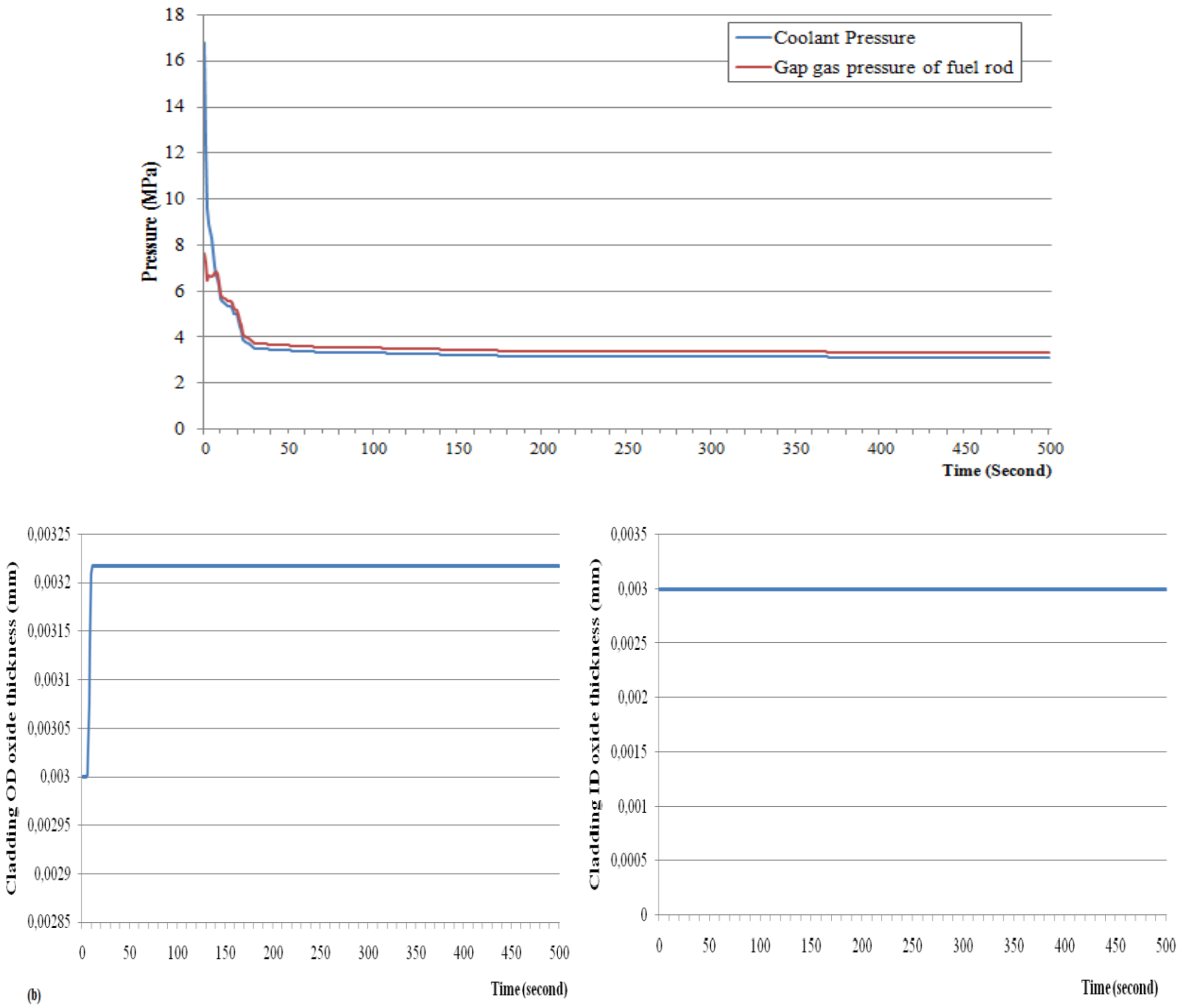

Fig.8. (a) Gap gas pressure of fuel rod; (b) Oxide thickness of fuel rod: Outer-diameter oxide thickness, and Inner-diameter oxide thickness

Thus, it was demonstrated that for the main circulating pipeline break accident, the maximum design limits of FE failure were not exceeded, i.e. the acceptance criteria were satisfactorily met, and the cladding was assured to store FE and prevent the release of fission products. However, the cladding Hoop strain is a little higher than the PSAR's safety criterion [3]. Therefore, reviewing safety criterion of cladding Hoop strain given in PSAR and further calculations are necessary.

2. Control Rods Ejection at Power in case of Drive Housing Rupture

The digitalized boundary condition data in AES-2006 PSAR [3] were imposed in the FRAPTRAN1.5 model in order to calculate the fuel thermal-mechanical behaviors during this accident. The transient was initiated at 0 s, when the control rod was ejected from the position 0 % from the core bottom. The CPS CR ejection inserted positive reactivity, and both reactor power and pressure began to increase. Upon the reactor neutron power above 107 %, at 0.01s of the transient, the signal for reactor shutdown was generated. Reactor parameters became stable owing to negative feedback effect. The maximum values of parameters were taken into consideration in the further evaluation step as shown in Figures 9 - 11 and Table II.

The fuel centerline temperature was calculated and shown in Figure 9. The results showed that fuel centerline temperature rocketed and reached the maximum value of 2473ºC ± 10ºC (2746 K). However, this value still met the safety criteria for the local melting of fuel pellets (less than 2540ºC for the burn-up fuel and less than 2840˚C for fresh fuel).

Fig.9. Fuel centerline temperature

Unlike the fuel temperature, cladding temperature increased more slowly due to the distance between fuel centerline and cladding, and the heat exchange between cladding and coolant. As shown in Figure 10, the cladding maximum average-temperature of the most heat-stressed FE in the accident process was determined as ~627 °C ± 5ºC (900 K),not exceeding 1200°C-temperature limit of the cladding as required in the acceptance criteria[3].

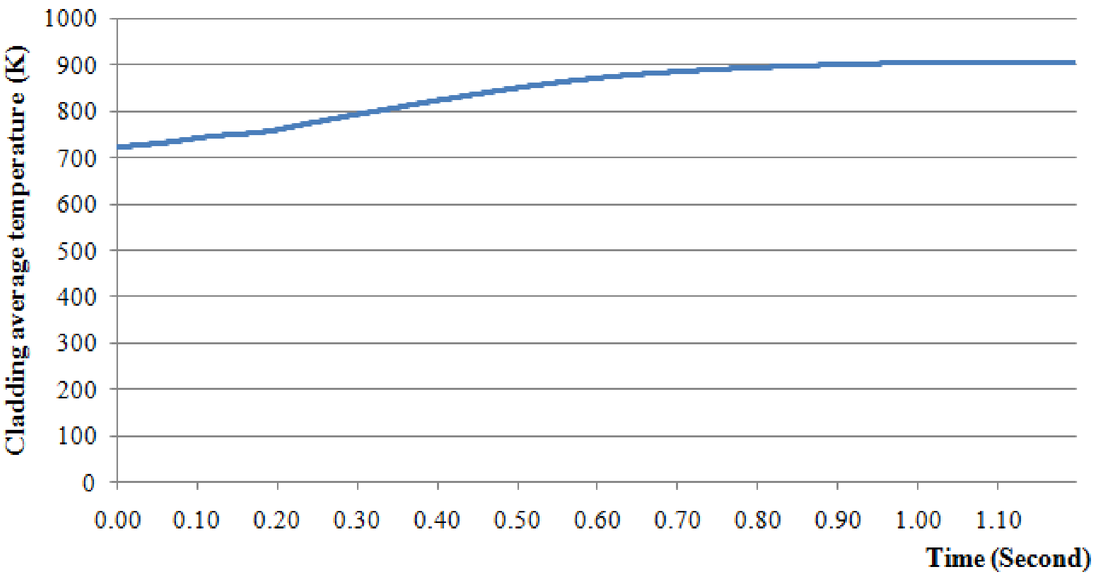

Fig.10. Cladding average temperature

It is required that the channels for the coolant flow inside fuel assembly shall not be blocked to the extent that cooling capability is violated due to swelling, fuel element cladding collapse as well as due to strain of other fuel assembly parts and core internals. The drop time of the control rod assembly is then accessed. Actually, this criterion was successfully assured during the accident. Furthermore, the melting of control rods did not occur as well.

Moreover, for considering the accepted enthalpy criteria, the fuel average enthalpy shall not exceed 830J/g for burn-up fuel and 963J/g for fresh fuel. In fact, the obtained maximum value of the fuel enthalpy was about 540J/g with the maximum increase of the fuel enthalpy of 140 J/g as shown in Figure 11 and the initial fuel enthalpy of 400 J/g given in *Section 1 Initial conditions*.

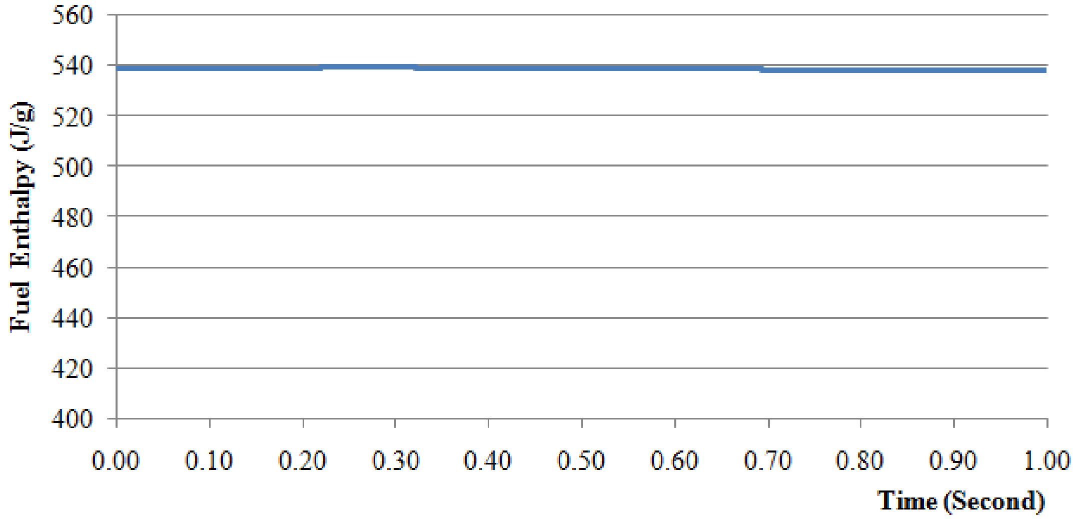

Fig.11. Fuel enthalpy

Table II. Maximum values of some parameters in the FRAPTRAN1.5 result of RIA

No.	**Parameters**	**Calculated values**	**Safety criteria**
	Fuel centerline maximum temperature, °C	2473	2540
	Cladding maximum average temperature, °C	627	1200
	Fuel maximum enthalpy, J/g	580	963

According to obtained results, temperatures of both FE and cladding were rather high (2473°C and 627°C, respectively). However, in general, the AES-2006 fuel rod was assured with its integrity in the RIA condition.

IV. DISCUSSION ON CALCULATED RESULTS

In the consideration of uncertainties of FRAPTRAN1.5 code, it is necessary to take into account many uncertainty parameters related to manufacturing and models, as follows [4, 5]:

- The manufacturing parameters, including the inner – outer diameters, density, fuel-cladding roughnesses, hole diameter, plenum length, rod fill pressure, cladding thickness, etc.;

- The model parameters such as the thermal conductivity, thermal expansion, fission gas release, creep, axial growth, hydrogen concentration, cladding stress/strain, high temperature oxidation, etc.

For FRAPTRAN1.5 code, the uncertainty method for modeling parameters and/or the results can be considered by random sampling of the input parameters in many running times of FRAPCON3.5 and then of FRAPTRAN1.5in order to achieve the upper bound results, which were further compared with the safety criteria (as in PSAR). The sampling of input uncertainty parameters for the consideration is dependent on operating conditions such as fuel burn-up, LOCA conditions, RIA conditions, etc.

However, in the scope of this article, the uncertainties in the input parameters and models were not considered due to the lack of the above information. Therefore, the results of calculations using FRAPTRAN1.5 code were directly compared with the PSAR values as shown in Table III.

Table III. Comparison of FRAPTRAN1.5 results with PSAR values

Operating conditions	Parameters	FRAPTRAN1.5 results	PSAR values	Differences, %
LBLOCA	Fuel maximum temperature, ^{o}C	1877	1800	+4.18
	Fuel minimum temperature, ^{o}C	217	210	+3.27
	Cladding maximum average temperature, ^{o}C	727	690	+5.22
	Fuel maximum average enthalpy, J/g	420	415	+1.19
RIA	Fuel enthalpy	580	546	+6.03
	Fuel maximum temperature, ^{o}C	2473	2450	+1.0
	Cladding maximum average temperature, ^{o}C	377	373	+1.1

The results from FRAPTRAN output showed small deviations of about 1- 5 % for LBLOCA and about 1 - 6% for RIA calculation, compared to the PSAR values. This means that with the rough estimation of boundary conditions from PSAR and some conservative assumptions, the calculations using FRAPTRAN1.5 code, in general, appeared more conservative than the PSAR values.

V. CONCLUSIONS

The calculations using FRAPTRAN1.5 code have been performed on the basis of the rough estimation of the boundary conditions for the fuel rod of AES-2006 required in the PSAR, and have been analyzed for both LBLOCA and RIA scenarios. In the LBLOCA condition, the obtained results showed that temperatures of both fuel element and cladding increased rapidly and reach the relatively high values, which required the evaluation of the FE integrity. However, it was evident that fuel melting did not occur, the cladding rod was not damaged, and the capability of storing fuel elements was ensured thanks to the good integrity of the cladding rod.

For fuel rod under the RIA condition, the fuel maximum temperature was found to be rather high of 2473ºC.However, the obtained results showed that the important parameters related to safety criteria, including fuel and cladding temperatures, and fuel enthalpy did not exceed the limited values. Therefore, the fuel rod failure did not occur, and the integrity of the cladding rod was maintained as well.

In general, the calculated results usingFRAPTRAN1.5code were similar to those given in the AES-2006 PSAR although in the calculating process, only conservative estimation of the boundary conditions and some conservative assumptions were used due to the lack of accurate information related to the power history from the PSAR. However, the results showed that the FRAPTRAN1.5 code can be further used to calculate the characteristics of the TVS-2006 fuel rods effectively and with high reliability.

It is expected that in the future, the FRAPTRAN1.5 code will be applied to evaluate the fuel rods of other nuclear reactors which may be selected for the second and third NPP project of Vietnam with more sufficient information from SAR and/or the results obtained from using several thermal hydraulic and neutronic computer codes (RELAP5, PANTHER, COBRA, SRAC, etc.).The lack of accurate information could be addressed to allow more realistic thermal-mechanical analyses of the fuel rods by FRAPTRAN1.5 code.

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