INVESTIGATION OF THE PELLET CLADDING INTERACTION RELATED ISSUES INCLUDING FUEL ROD FAILURE BY METHODS FOR IDENTIFICATION SYSTEM WITH DISTRIBUTED PARAMETERS

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Abstract

Application of a new method for investigation and control of pellet-cladding interaction (PCI) features in fuel rods of the WWER reactor is described. The method is based on use of the methods for identification of the systems with distributed parameters. A thermal contact resistance (TCR), the value inverse to effective gap thermal conductance, was chosen as the controlled parameter. The TCR value is indicative of potential fuel rod failure from PCI and of such relatively thin effects as pellet jump and relocation. Some experimental and calculating results are described to demonstrate the foregoing.

Introduction

Various interacting mechanical and thermal processes operate in a fuel rod during irradiation. Fuel rod modelling codes are written to predict the outcome of these interactions. The predictions of interest are fuel and cladding temperatures, and stresses and strains relative to probable cladding failure thresholds. The complexity of current codes ranges over a broad spectrum. One way to test the fuel rod modelling codes is to compare their prediction with in-reactor fuel temperature data.

Enhanced requirements for modelling codes resulting from accident analysis do not allow us to confine one to such comparison, since steady-state centreline temperature alone is not definitive regarding details of heat transfer within a fuel rod. Favourable comparison between steady-state temperature data and code prediction prove only that code's models and assumption work together to produce reasonable temperatures. The validity of individual models is not confirmed. Of particular concern are the models for effective fuel thermal conductivity and effective conductance across the fuel-cladding gap. Another way to test the fuel rod modelling codes is to use methods for identification system with distributed parameters.

A method for investigation of the features of the contact heat exchange in the fuel-cladding gaps of the commercial reactor has been developed at the Institute of Reactor Technology and Materials, together with Moscow State Aviation Institute [1,2]. This method is based on use of the methods for identification of the system with distributed parameters using algorithms based on the extreme methods for solving the inverse heat conduction problems. The method has been successfully used for WWER-440 and WWER-1000 fuel rod investigations on the multiloop reactor MR. The investigations presented are shown in Figure 1 as data flow diagram.

An important advantage of this method is that of thermal contact resistance; *R* between the pellet and the cladding that is one of the nuclear reactor safety criteria is determined by the results of the non-stationary measurements carried out in the operating reactor without affecting its operation conditions. The method permits the level of the pellet-cladding mechanical interaction to be investigated. The method has been patented [3]. The first application of the method has been described in the pre-print Kurchatov Institute [4]. A number of investigations have been done on the MR reactor by that method including features of the TCR between pellet and cladding, fuel rod failure due to PCI and TCR behaviour under RAMP conditions. The results for initial period of irradiation obtained in these experiments are discussed below.

Fuel relocation and pellet jump during first power rise

The experiments were performed in three series. The main characteristics of the instrumented fuel rods studied in the first series are listed in Table 1.

The results obtained for the initial start-up with fresh fuel are presented in Figure 2. As the power of the reactor increases within the range of linear heat generation rates (LHGR) from 150 W/cm to 240 W/cm, the thermal contact resistance between the pellet and the cladding decreases with increase in the linear heat generating rate.

Figure 1. Data flow diagram

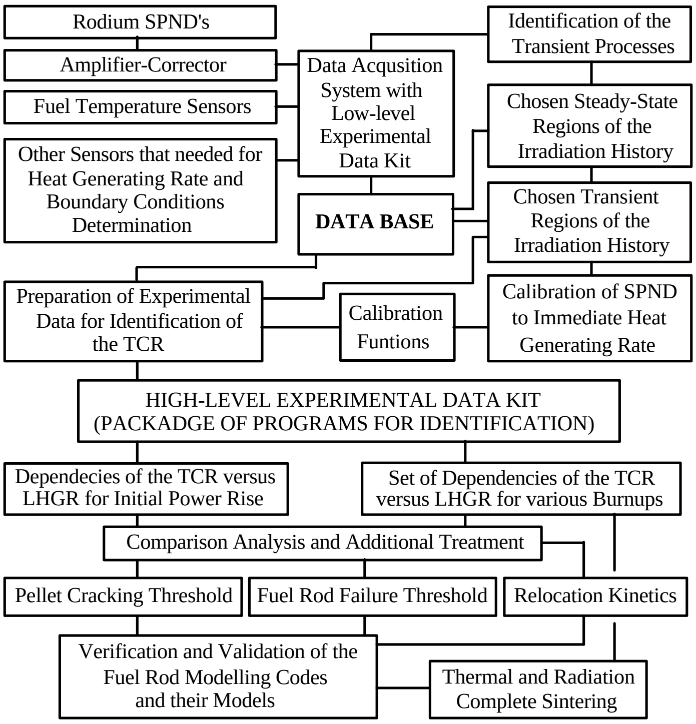
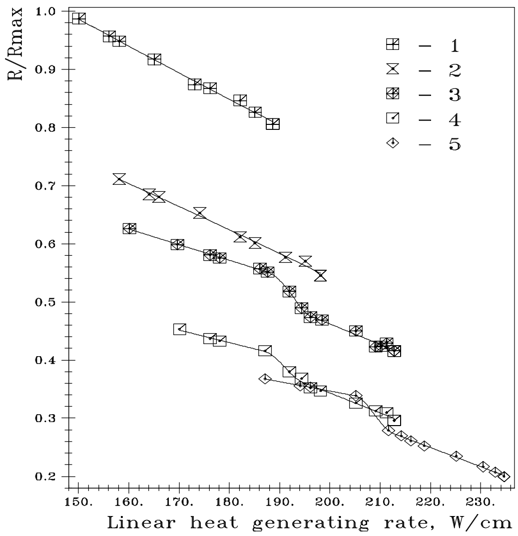


Table 1. SOFIT-1.2. List of the instrumented fuel rods

Fuel Rod Parameter	R1	R2	R3	R4	R5	R6
Pellet Density, g/cm^3	10.60	10.50	10.55	10.45	10.50	10.45
Gas Pressure, MPa	0.1	0.1	0.1	0.1	0.1	0.1
Gas Filler	He	He	Xe	He	Xe	He
Gap, mm	0.15	0.20	0.20	0.27	0.15	0.20

Figure 2. Experiment SOFIT-1.2. Initial increase in the reactor power

1 – R3; 2 – R5; 3 – R2; 4 – R6; 5 – R1



However, as seen from Figure 2, at a linear heat generation rate of 190 W/cm in fuel rods R2 and R3 and at 210 W/cm in fuel rod R1 a “pellet jump” takes place. This is the relocation effect that is a joint manifestation of the effects of fuel pellet cracking and fragmentation that results in effective increase in the pellet diameter and in reduction in the fuel-clad gap size.

A similar effect is described by M. Oguma for a linear heat generating rate of 60 W/cm in the BWR fuel rods with pellet diameter of 13 mm [5]. The result presented agree with those obtained in above mentioned reference since the WWER fuel rods have pellets of 7.8 mm in diameter. At the same time, as seen from Figure 6, the calculation of the thermal gap conductance by the PIN code using the Ross-Staute gap conductance model does not show a pellet jump in the given range of linear heat generating rates. Lack of this jump leads to an error in the determination of the fuel temperature up to 112 degrees and to the conservative estimate of the risk of the steam-zirconium reaction in the loss of coolant accident (LOCA) [6,7]. The foregoing suggests the inadequacy of the relocation model used in above mentioned code. However, nowadays there are some new codes such as CAES-S and CAES-R with relocation models corrected by reported results [8,9]. The predictions of these codes are free from above mentioned errors.

For the inverse problem of the TCR estimation to be solved, some non-measurable parameters determined either by the in-reactor readings or the data of independent experiments and calculations are used as the initial data, together with the parameter measured. These are such parameters as energy generation in the fuel rods, external heat transfer coefficient, thermal and physical characteristics (TPC) of the fuel and of the cladding. The uncertainty in the specification of these parameters resulting from the error in the their determination may affect the precision of the inverse heat conductance problem.

Using the data of non-stationary measurements in bringing the reactor with fresh fuel to power, an analysis was carried out of the effect of uncertainties in specification of non-measurable initial data in solving the inverse heat conductance problem on the error in TCR estimation. The inverse heat conductance problem was solved based on the experimental data but with the changed coefficient of the mathematical model, whose effect on the uncertainty in TCR estimation was analysed. Such parameters as the coefficient of heat transfer from the fuel cladding to the coolant and linear heat generating rate in the fuel rods was considered as non-measurable parameters that were specified during the uncertainty analysis when investigating the greatest effect on the error in TCR estimation.

The calculations have shown that an error of 30% in the external heat transfer coefficient; α effects slightly the estimated value of the thermal contact resistance; the error in thermal contact resistance R is as high as 1.5%. The error in the energy generation leads to an error of 5% in the estimation of TCR. This error in energy generation leads to 5.5% errors in R for the helium-filled fuel rods at a linear heat generating rate up to 200 W/cm, and to 2% at higher linear heat generating rates. The letter is due to stepwise change in TCR value at a given value of q_l causes by the reduction of the pellet-cladding gap. The influences of these uncertainties are shown in Figure 3.

The recovered dependence of R versus q_l, obtained by treatment of non-stationary measurements in the third reactor power ascent, is presented in Figure 4. Since in irradiation of the fuel rods gaseous fission products (*Xe* and *Kr*) are collected at the cladding wall, the curves in Figure 4 overlap almost the whole range of possible changes in R for the fuel rods of this type.

Figure 3. Experiment SOFIT-1.2. Initial increase in the reactor power

Fuel rod R1: 1 – recovered value of TCR

2, 3 – recovered value of TCR with allowance for uncertainty in α of ±30%

4 – recovered value of TCR with allowance for uncertainty in λ of ±5%

5 – calculation by PIN code

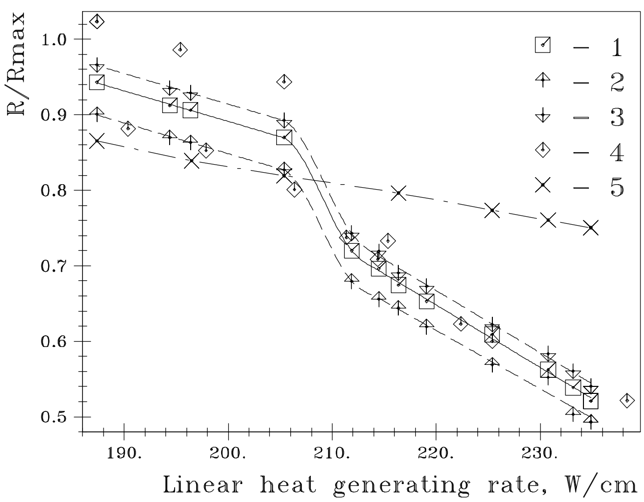
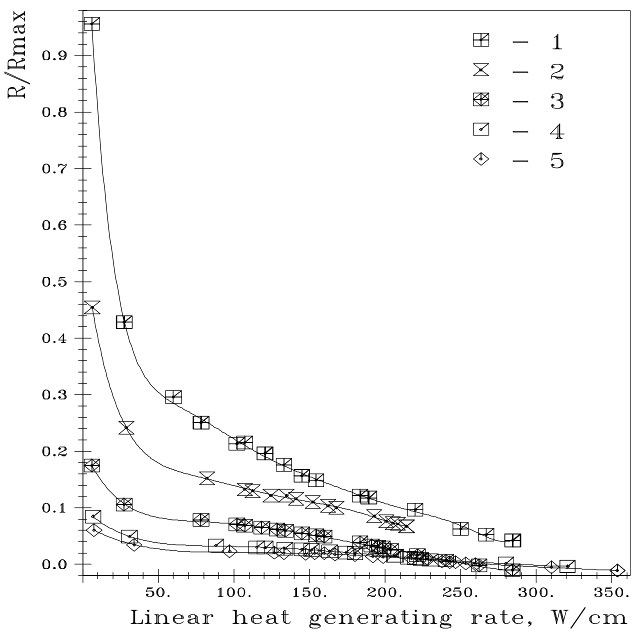


Figure 4. Experiment SOFIT-1.2 Third reactor power rise

1 – R3; 2 – 5; 3 – R4; 4 – R6; 5 – R1



The thermal conductance of the fuel is one of the most unstable thermal and physical characteristics of the fuel rod materials, which vary under irradiation due to change in the structure and density of fuel. The fuel thermal conductance is most reliably known for the initial irradiation period, analysed in the present paper, when the thermal and radiation changes in the fuel structure are still small.

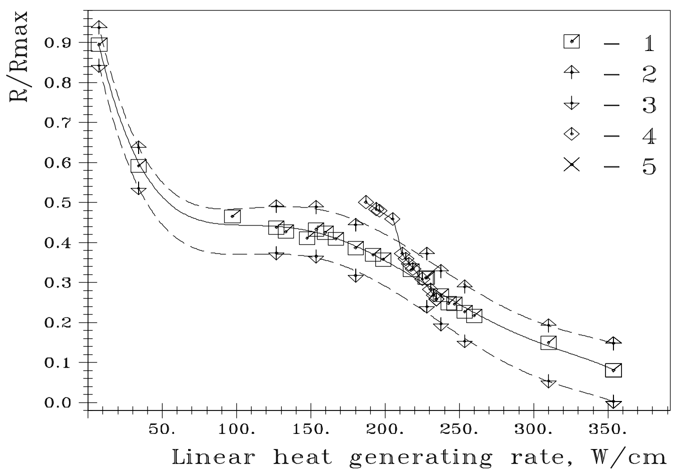
On the basis of the non-stationary measurements during the third occassion that the reactor was brought to power the effect of the uncertainty in the fuel conductance specification on the error in TCR determination has been analysed. The analysis has revealed that the uncertainty in the fuel conductance specification leads to an equidistant shift on the curve approximating the dependence of TCR on the linear heat generating rate down in underestimation of the fuel thermal conductance and up in its overestimation. It is shown in Figure 5 on the example of fuel rod R1. This leads to reduction of the corridor of solution around low heat generating rates where the TCR gradient on the linear heat generating rate is the highest one and to increase in the error in determination of TCR resulting from that in the specification of the temperature dependence of the fuel conductance around high linear heat generating rates since TCR decreases with the rise in the linear heat generating rate.

Figure 5. Experiment SOFIT-1.2. Third the reactor power rise

Fuel rod R1: 1 – recovered value of TCR

2, 3 – recovered value of TCR with allowance for uncertainty in the specification of the fuel thermal conductance in ±10%

4 – recovered value of TCR in bringing the reactor with fresh fuel to power (initial increase of the reactor power)



The error in TCR determination with the uncertainty of 10% in the fuel thermal conductance determination was from 8% to 20% depending on the linear heat generating rate for the helium-filled fuel rods and from 4% to 10% for the fuel rods filled with xenon. The 10% uncertainties for the fuel thermal conductance were specified by the lower and upper boundaries of the 10% corridor and the error in λ_f for the $\lambda_f = \lambda_f(T)$ function [10].

The analysis presented here makes it possible to estimate the error of ±10% in TCR determination. It should be pointed out that allowance for the uncertainties does not lead to any change in the character of the dependence of *R* on the linear heat generating rate. Besides the effect of equidistant shift of the curves, shown in Figure 5, permits a simple and reliable method to be constructed for determination of the thermal and radiation complete sintering of the fuel, based of the $R = R(q_l)$ dependencies obtained for the irradiation range studied. In present case additional densification of the fuel above 1% was observed to be determined by that method.

Therefore investigation of the fuel rods using the TCR identification method presented is an unconventional method for fuel rod investigation, which permits new original results to be obtained.

In the experiment SOFIT-1.3 (the second series of experiments presented) a number of investigations has been conducted through the method presented here. Practically all transient processes have been recorded with high registration speed (1 times per second). Besides that, some power dips for above 50% of nominal reactor power have been done during first month of irradiation of the experimental fuel assembly. This made it possible to provide investigations of the pellet cracking threshold; irradiation induced densification and relocation kinetic versus burnup during initial period of irradiation, when the main part of transformation into fuel takes place. The main characteristics of the experimental fuel rods studied are listed in Table 2.

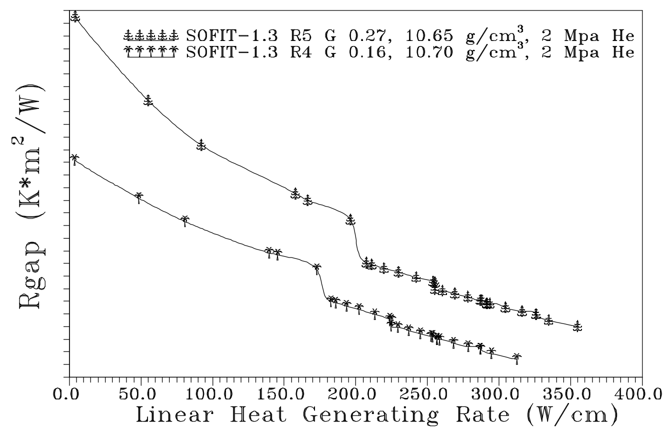
Table 2. SOFIT-1.3. List of the instrumented fuel rods

Fuel Rod Parameter	R1	R2	R3	R4	R5	R6
Pellet Density, g/cm^3	10.65	10.65	10.70	10.70	10.65	10.65
Gas Pressure, MPa	2.0	2.0	2.0	2.0	2.0	2.0
Gas Filler	He	He	He	He	He	He
Gap, mm	0.27	0.27	0.15	0.15	0.27	0.27

The results obtained for the initial start-up with fresh fuel are presented in Figure 6. As seen from Figure 6, the pellet jump took place at a linear heat generating rate of 175 W/cm in the fuel rod with 0.27 mm gap and at 200 W/cm in the fuel rod with 0.15 mm gap. The secondary relocation took place at 225 W/cm and 260 W/cm accordingly. So, pellet jump (or initial relocation) has been observed at the same linear heat generating rate range for helium filled fuel rods in the experiments SOFIT0-1.2 and SOFIT-1.3 both.

Figure 6. Experiment SOFIT-1.3. First the reactor power rise

Fuel rods R4 and R5



Fuel rods failure during power rise

The results obtained for the first (and the last) power rise in experiment SOFIT-1.4 are presented in Figures 7-13. The main characteristics of the experimental fuel rods studied are listed in Table 3.

In this case the experimental fuel assembly with fresh fuel has been irradiating within the shutdown period of the reactor during which it was cooled to a temperature below 100°C for exactly thirty days. As a result, initial gaps between pellet and cladding have been decreased for above 0.03 mm.

The matter is in distortion of the pellets due to radiation damage. In general, when uranium, plutonium or americium which is radioactive and fissionable, or a metallic oxide, a metallic carbide or a metallic nitride containing such element is let to stand at the normal temperature, it sustains a damage due to radioactive rays and a crystal lattice is distorted. That is, there develops a phenomenon in which one uranium element constituting the crystal lattice of uranium has its position shifted by irradiation, whereby the crystal lattice is outspread and the volume of uranium is increased.

Table 3. SOFIT-1.4. List of the instrumented fuel rods

Fuel Rod Parameter	R1	R2	R3	R4	R5	R6
Pellet Density, g/cm^3	10.45	10.65	10.48	10.45	10.47	10.57
Gas Pressure, MPa	0.5	0.5	0.5	0.1	1.0	0.5
Gas Filler	He	He	He	Xe	Xe	He
Average Gap, mm	0.22	0.14	0.27	0.21	0.21	0.22

Figure 7. Experiment SOFIT-1.4. First the reactor power rise

Fuel rod R1

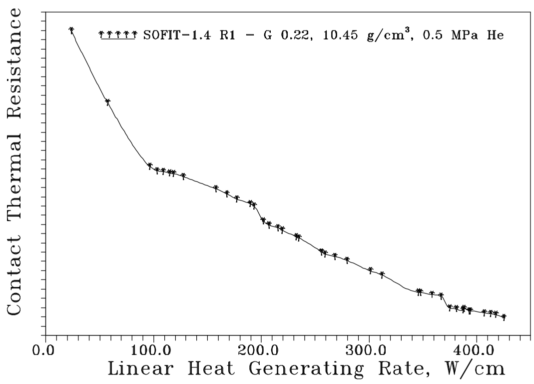


Figure 8. Experiment SOFIT-1.4. First the reactor power rise

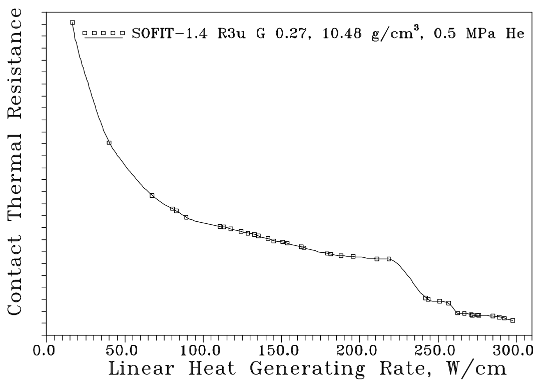


Figure 9. Experiment SOFIT-1.4. First the reactor power rise

Fuel rod R3 at the location of the centre thermocouple

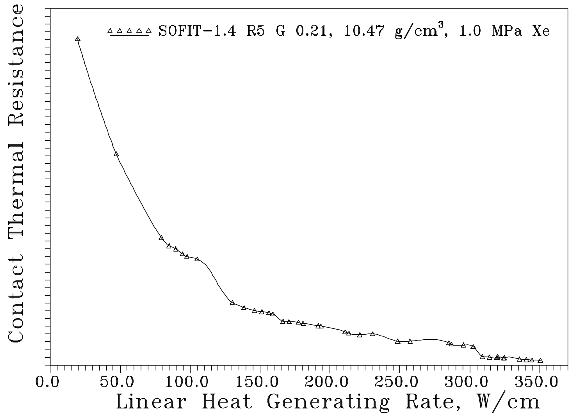


Figure 10. Experiment SOFIT-1.4. First the reactor power rise

Fuel rod R5 at the location of the upper thermocouple

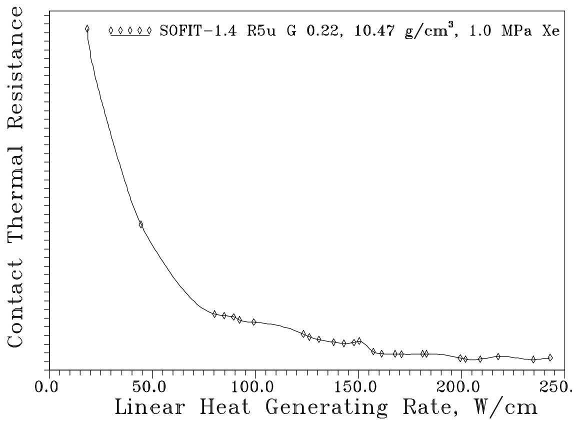


Figure 11. Experiment SOFIT-1.4. First the reactor power rise

Fuel rod R2

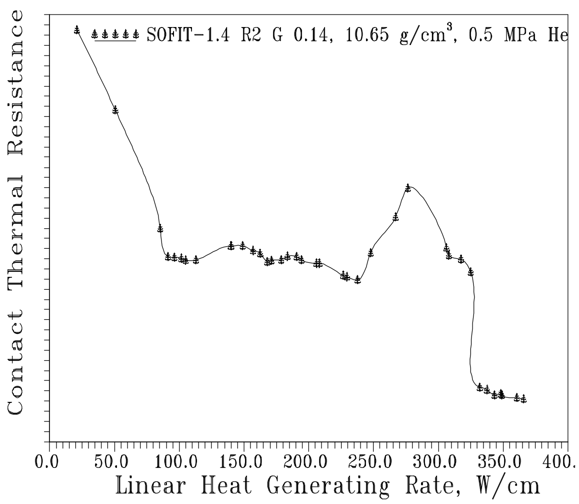


Figure 12. Experiment SOFIT-1.4. First the reactor power rise

Fuel rod R6

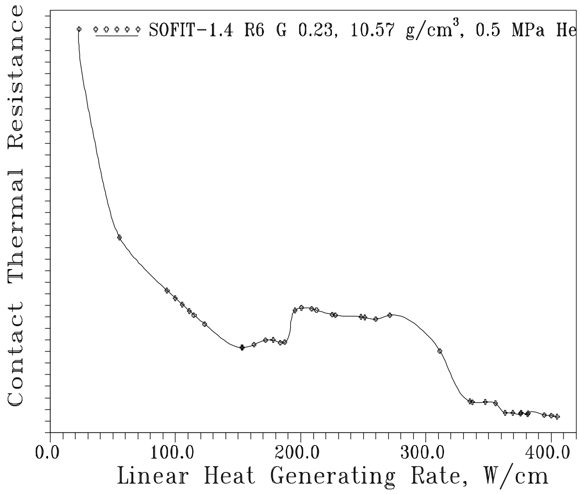
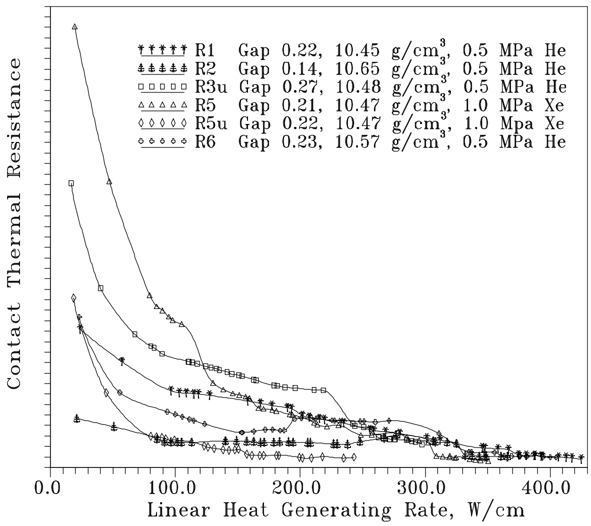


Figure 13. Experiment SOFIT-1.4. First the reactor power rise

Thermal contact resistance between the pellet and cladding versus linear heat generating rate for all instrumented fuel rods



It is known that a similar phenomenon develops also in case of a polycrystalline sintered compact made of the aforementioned substances. The quantity of distortion due to the radiation damage differs to some extent in dependence on kind of the substances.

When a specific substance is considered, the period of time in which the quantity of saturation distortion is reached differs in dependence on the disintegration constant of a radioactive substance contained in the specific substance, the kind and energy level of radioactive rays, etc., but the quantity of saturation distortion is considered to be determined by the substance. For example, when a metallic oxide containing U-238 is let to stand, it sustains a distortion due to radiation damage by approximately 0.2-0.3% in a year or so. However, where such material is fissioned in the nuclear reactor, a large number of fission products emitting radioactive rays at very high energy are accumulated in the material. For this reason, the quantity of saturation distortion is sufficiently reached within the shutdown period of the nuclear reactor during which it is cooled to a temperature below 100°C. Such phenomenon is defined as radiation damage.

So, experiment SOFIT-1.4 was an experiment with extremely small gaps between the pellet and cladding. It caused the failure of at least two fuel rods: fuel rod R2 and R6, as will be discussed below and demonstrated in Figures11 and 12. The failure of these fuel rods occurred because the gap became zero before the relocation and densification of the fuel had happened. It proves the particular danger of the reactor power rise with fresh fuel.

A linear heat generating rate above 200 W/cm and the secondary relocation at above 370 W/cm was observed in the experiment SOFIT-1.4. It corresponds well with the results obtained in experiments SOFIT-1.2 and SOFIAs seen in Figures 7 and 13, in which the initial relocation of the fuel rod R1 took place at a T-1.3 for helium filled fuel rods. In the fuel rod R3 the pellet jump (initial relocation) took place at a linear heat generating rate above 230 W/cm at the location of the upper thermocouple and the secondary relocation at above 370 W/cm, as shown in Figures 8 and 13. This corresponds well with the results obtained in experiments SOFIT-1.2 and SOFIT-1.3 as well. The thermocouple at the maximum zone of the heat generating rate was out of order during that experiment, as well as the thermocouple in fuel rod R4.

As seen from Figures 9 and 13, in fuel rod R5 at the location of the upper thermocouple the initial relocation took a place at a linear heat generating rate above 150 W/cm and the secondary relocation did not occur before 250 W/cm in the SOFIT-1.4 experiment. In this fuel rod at the maximum zone of the heat generating rate the initial relocation took place at a linear heat generating rate above 120 W/cm and the secondary relocation at above 305 W/cm, as shown in Figures 10 and 13.

As seen from Figures 11 and 13, the failure of the fuel rod R2 took place at a linear heat generating rate above 250 W/cm. It caused increasing of the TCR to more than $5{\cdot}10^{-5}$ $K{\cdot}m^2/W$ and temperature rise to above 250 degrees. The pellet jump took place at 325 W/cm in this fuel rod, when one was filled with water vapour.

As seen from Figures 12 and 13, the failure of the fuel rod R6 happened at a linear heat generating rate of above 200 W/cm. It caused increasing of the TCR at above $5{\cdot}10^{-5}$ $K{\cdot}m^2/W$ and temperature rise for above 200 degrees in 1000 seconds. The pellet jump took a place at 320 W/cm in this fuel rod and the secondary relocation at above 360 W/cm, when one was filled with water vapour.

This experiment was interrupted immediately after the first power rise due to an extreme increase in loop radioactivity, which was indirect evidence of the fuel rods' failure. The experimental fuel assembly consisted of 18 fuel rods. The foregoing results for TCR identification made it possible to point out at least two instances where fuel rod failure had occurred, and to form a special program for post-irradiation examination. The result presented can confirm the particular danger associated with WWER's power raise operational condition after shutdown period. Such WWER operating conditions demand a thorough preconditioning of the fuel rods depending upon the length of the shutdown period. On the other hand, the WWER's power increase with fresh fuel does not seem as dangerous as previously thought.

Fuel rod behaviour depending on burnup and ramp condition

The method presented is a powerful tool for fuel rod behaviour investigations under high burnup and ramp conditions. To demonstrate this, some results from the second series of experiments with power ramp are presented in Figures 14 and 15, in comparison with the first power rise. The main characteristics of the experimental fuel rods studied are listed in Table 2. Investigations consist of obtaining dependencies – as shown in Figure 1 – with a wide range of burnups for further analysis,. In particular, it is interesting to observe the signs of the beginning of "soft" mechanical interaction between pellet and cladding after the power ramp shown in Figures 14 and 15. Besides, the method seems very sensitive to the rim effect which occurs at a high level of burnup.

Figure 14. Experiment SOFIT-1.3. First the reactor power rise in comparison with first power ramp

Fuel rod R4

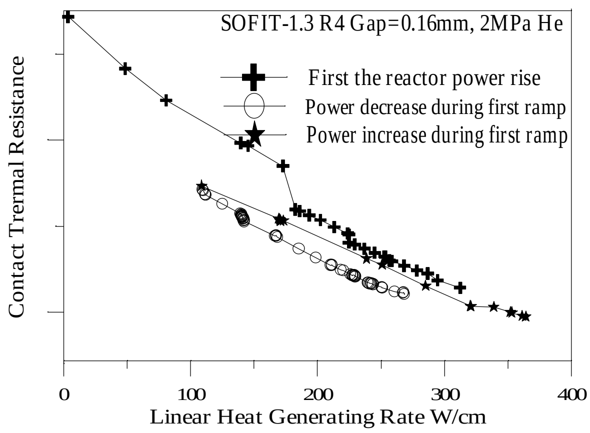
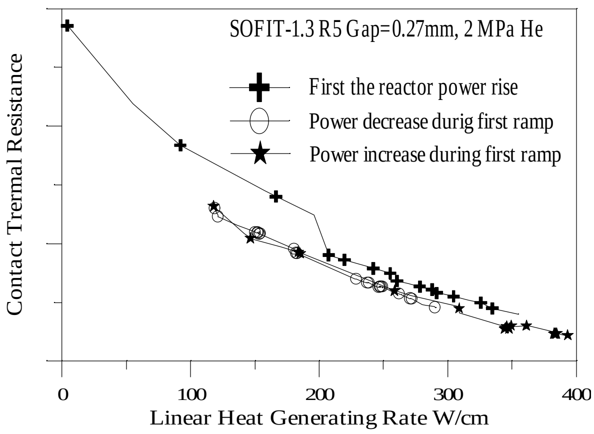


Figure 15. Experiment SOFIT-1.3. First the reactor power rise in comparison with first power ramp

Fuel rod R5



The results presented here were obtained for the local value of TCR, since the thermocouple, located in the centre hole of the fuel rod, has been used as a temperature sensor. However, the method used allowed to recover the integrated value of TCR with the desired grade of integration. In this case an integral temperature sensor such as an expansion thermometer should be used. Use of this type of temperature sensor makes the method independent of the limited lifetime of the thermocouples. The method and specialised software used also permit the thermal conductivity and specific heat of the fuel pellets as function of temperature to be recovered. This requires at least two temperature sensors to be located in the desired cross-section of the fuel rod.

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