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**EQUIPMENT AND TECHNIQUE FOR IN-CELL
OFF-NORMAL OVERHEATING OF ROD BUNDLES**

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ABSTRACT

Equipment and techniques were developed to simulate off-normal overheating of irradiated and unirradiated rod bundles (up to 19 uniform or nonuniform rods contained in bundle).

Deformation features of gaseous pressurized cladding of the VVER reactor fuel element on its heating as a part of bundle were presented. Engineering capabilities of developed equipment were demonstrated. There are experimental results to study zirconium cladding deformation under conditions simulating off-normal overheating of the VVER-1000 reactor fuel assembly.

I. INTRODUCTION

A great variety of equipment and techniques for in-reactor off-normal tests of single fuel elements and bundles of thermal and fast reactors has been developed and put in operation at SSC RIAR. The equipment and techniques are being improved continuously. In particular, refabrication technology of fuel elements and their instrumenting with the different sensors is improved along with the system of information accumulation and treatment in the process of in-reactor off-normal tests.

Equipment and techniques for in-reactor off-normal tests of single fuel elements and bundles are an important part of SSC RIAR experimental complex intended to study the reliability of reactor core components when various off-normal events are initiated. Another part of this complex is off-cell and in-cell stands intended to simulate in-reactor off-normal events.

The results of developments and investigations performed at RIAR by the technique of local off-normal overheating of single fuel elements have been reported at the recent Working Group Meeting [1]. The investigations resulted in the following distinctions in the corrosion and thermal effects between unirradiated and irradiated fuel elements:

- Overheating of the SM-2 reactor irradiated fuel elements (cross-like fuel elements containing dispersive $\text{UO}_2 + \text{Cu}$ fuel clad with steel) leads to the local deformation of cladding with its subsequent brittle fracture. It was observed at

temperature close to 700°C and does not relate with metallurgical fuel-cladding interaction. By contrast, the cladding fracture of unirradiated fuel elements is not accompanied by its deformation and occurs at temperature close to 1000°C that was found to result from the metallurgical fuel-cladding interaction.

- Overheating of the VVER-1000 reactor irradiated fuel elements (oxide fuel clad with Zr-1%Nb alloy) was found to have the more high rate of cladding oxidation by overheated water vapour.

- Overheating of the BOR-60 reactor irradiated fuel elements was found to have the more high rate of the metallurgical interaction between steel cladding and U-Pu metal fuel.

The development of equipment and techniques for off-normal tests of fuel element bundles is the logical continuation of work related to the development and techniques for off-normal tests of single fuel elements. The quantitative and qualitative distinctions are expected to exist in the behavior of single fuel elements and bundles. The behavior of bundles incorporated fuel elements is expected to differ from those contained different elements.

Zirconium alloy used for manufacture of the PWR, BWR and VVER reactor fuel element claddings is of high plasticity at temperature exceeding that of normal operation, especially. In particular, deformation of zirconium alloys including postirradiated "zircaloy" type can reach and exceeds as high as 100% at temperature above 700°C /2,3,4/. These temperatures are found to be lower those at which the intensive cladding oxidation by water vapour, vapour-zirconium reaction and cladding-oxide fuel interaction take place. Therefore, under the effect of excess internal gas pressure, the large and non-uniform deformation of fuel element claddings can have a great influence on the kinetics of interaction between the fuel assembly components and water vapour at temperature above 700°C.

2. EXPERIMENTS ON MECHANICALLY RESTRICTED DEFORMATION OF ZIRCONIUM CLADDING

Origin of non-uniform deformation in the uniformly heated zirconium cladding was verified by a series of experiments. The schemes are given in Fig.1. The single unirradiated zirconium claddings under excess internal gas pressure were tested. A quartz tube and ceramic rod (mechanical deformation restrictors) confined the average cladding deformation and allowed to realize the non-uniform deformation within the uniformly heated cladding. The flux of vapour-gas mixture was generated within the quartz tube.

The experimental results are presented in Fig.1 and 2. The mechanically restricted deformation and friction forces were shown to be the cause of non-uniform

deformation along the perimeter. The maximum deformation is specific to areas which are geometrically similar to the angles of hexahedron or square available in the triangular or square lattice. The values of deformation on these areas were close to the calculated ones.

3. POSSIBLE EFFECTS IN ROD BUNDLES

The possible specific effects happened in the PWR, BWR, VVER reactor fuel assemblies are determined by capability to large deformation and shape change of zirconium claddings at temperature above 700°C. In particular, when emergency is accompanied by simultaneous overheating of fuel element claddings and origin of excess gas pressure, the following effects may be observed in the overheated fuel assemblies:

- deformation and shape change of fuel element claddings which accompanied by mechanical contact of claddings, decreasing of coolant passage in fuel assembly and worsening of fuel element cooling in the centre of fuel assembly;

- increasing of fuel element temperature in the centre of fuel assembly which is accompanied by thermomechanical stresses occurred in fuel assembly, their relaxation and change of fuel assembly shape;

- oxidation decreasing of cladding external surface on areas of mechanical contact.

The thermomechanical stress of opposite sign was observed in the overheated fuel assembly at the stage of its cooling. The value and gradient of these stresses in the cross-section of bundle depend on the rate of cooling and parameters of preceding overheating. The parameters of bundle overheating and cooling specify deformation type and intensity, change of shape, damage of fuel elements contained in bundle and the bundle itself.

The above thermal, corrosion and mechanical effects are not able to find in off-normal overheating of single fuel elements.

4. EQUIPMENT AND METHODS OF BUNDLE TESTING

In-reactor off-normal tests of rod bundles (integral tests) provide the most valuable and reliable information. However, it is impossible to perform a great number of in-reactor tests to estimate the influence of every parameter and their different combinations on occurrence and growth of damage in the rod bundle. It is reasonable to resolve these problems with in-cell off-normal tests of irradiated rods. The equipment used for such tests was developed at RIAR.

The scheme of experimental stand, type and parameters of possible tests are presented in Fig.3 and Table 1. The stand is considered to be used for testing of single rods and bundles which can incorporate up to 19 rods of 9.15 mm diameter. The lengths of claddings and fuel elements may be contained in the bundle involving those sealed under the excess gas pressure. The lengths of steel channels designed for control rods may be incorporated in the bundle also. The lengths of control rods involving those sealed under the excess pressure may be also contained within lengths of steel channels. The tests may be conducted in inert and vapour-gas medium under various conditions of heating and cooling.

The vertical arrangement of rod bundle in the high frequency inductor, the regulable water level of rod bundle allow to simulate three zones with fuel assembly height of the VVER, BWR and PWR reactor types during in-reactor overheatings. They involve the zone of water boiling and evaporating, the zone of rods and water vapour overheating and the zone of overheated vapour condensation and opposite movement of condensate.

The method of induction heat allows to cool rods without interruption of heating. It allows to simulate the residual energy release from the cooled rods involving the damaged ones. It is impossible to perform a like simulation at stands where rods are heated electrically.

The videorecording of processes occurred in testing of rod bundle may be performed by the viewing window.

5. CONCLUSION

The high plasticity of zirconium alloys at temperature above 700°C can be the cause of the specific corrosion, thermal and mechanical effects appeared in the bundles of overheated rods contained fuel elements available internal excess gas pressure. These effects involve change of shape and non-uniform deformation of fuel element claddings, decreasing of bundle (fuel assembly) coolant passage, thermomechanical interaction of rods on heating and cooling. The similar effects can not be found and studies on overheating and cooling of single fuel elements.

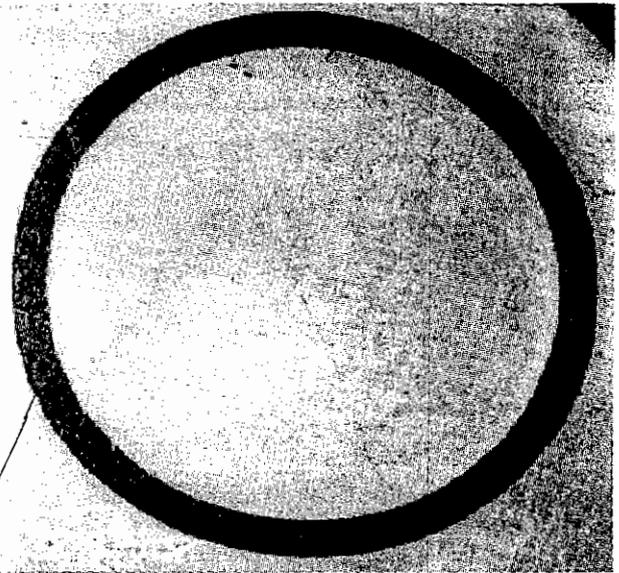
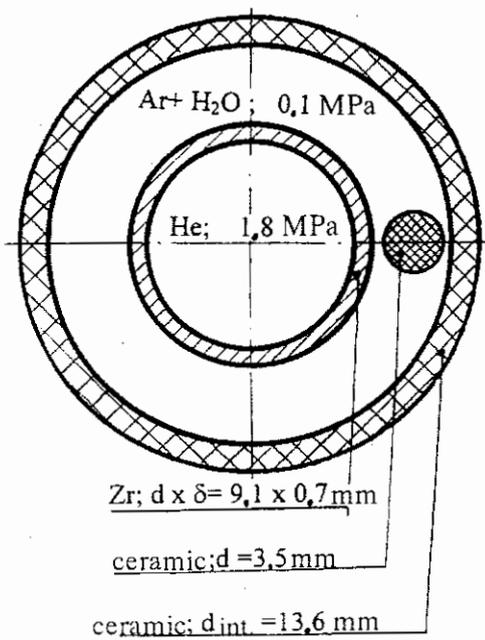
There is equipment developed to simulate the in-reactor rod bundle overheating and cooling in hot cells. These bundles may contain up to 19 uniform or different rods (fuel elements of 9.15 mm diameter, fuel elements and control rod channel).

The techniques and equipment designed for performance of integral in-reactor off-normal tests and in-cell overheatings present the RIAR unique experimental complex used to evaluate the reliability of fuel elements, the VVER-type reactor control rods and rod bundles (fuel assembly) under off-normal events. The complex can be used in the scope of international cooperation to study and to compare the

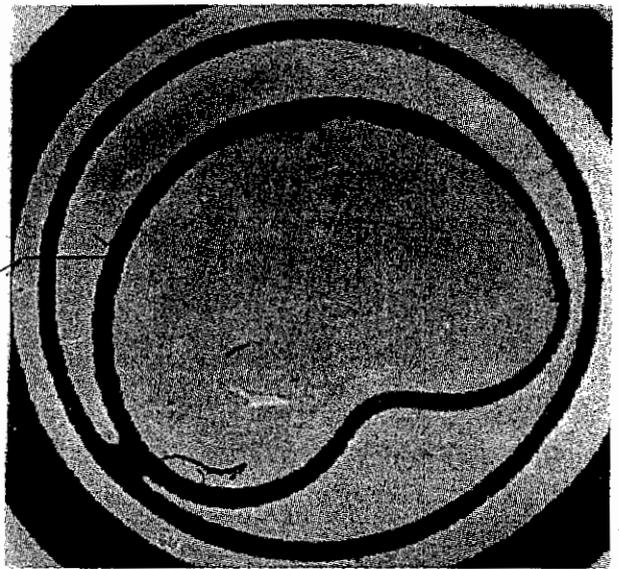
reliability of the different rods and their behaviour under off-normal events which may occur in the VVER, PWR and BWR reactor types.

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885°C x 300 s; P;
 $\bar{\epsilon} = \psi^{max} = 7\%$



945°C x 300 s; P;
 $\bar{\epsilon} = 30\%$; $\psi^{max} = 50\%$

960°C x 300 s; DP;
 $\bar{\epsilon} = 45\%$; $\psi^{max} \approx 100\%$

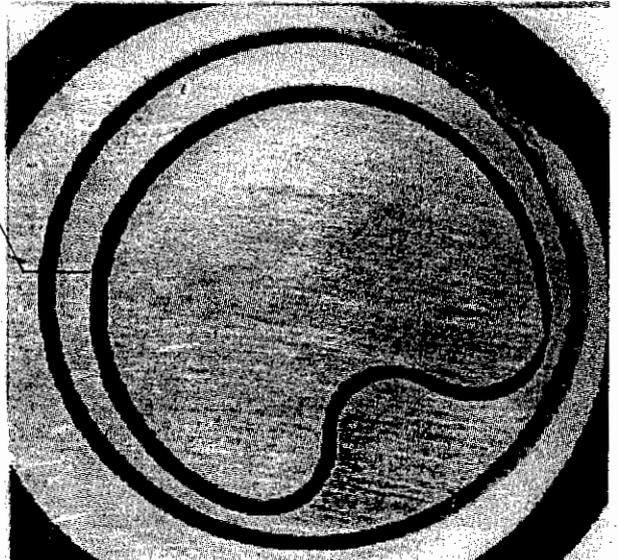


Fig. 1. The scheme of experience and shape changes of Zr-1%Nb alloy cladding after test at deformation restriction

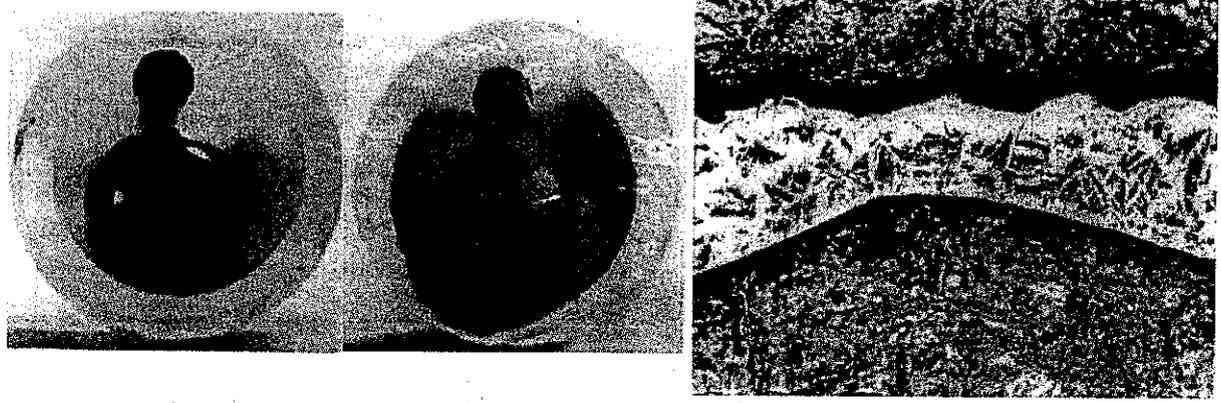
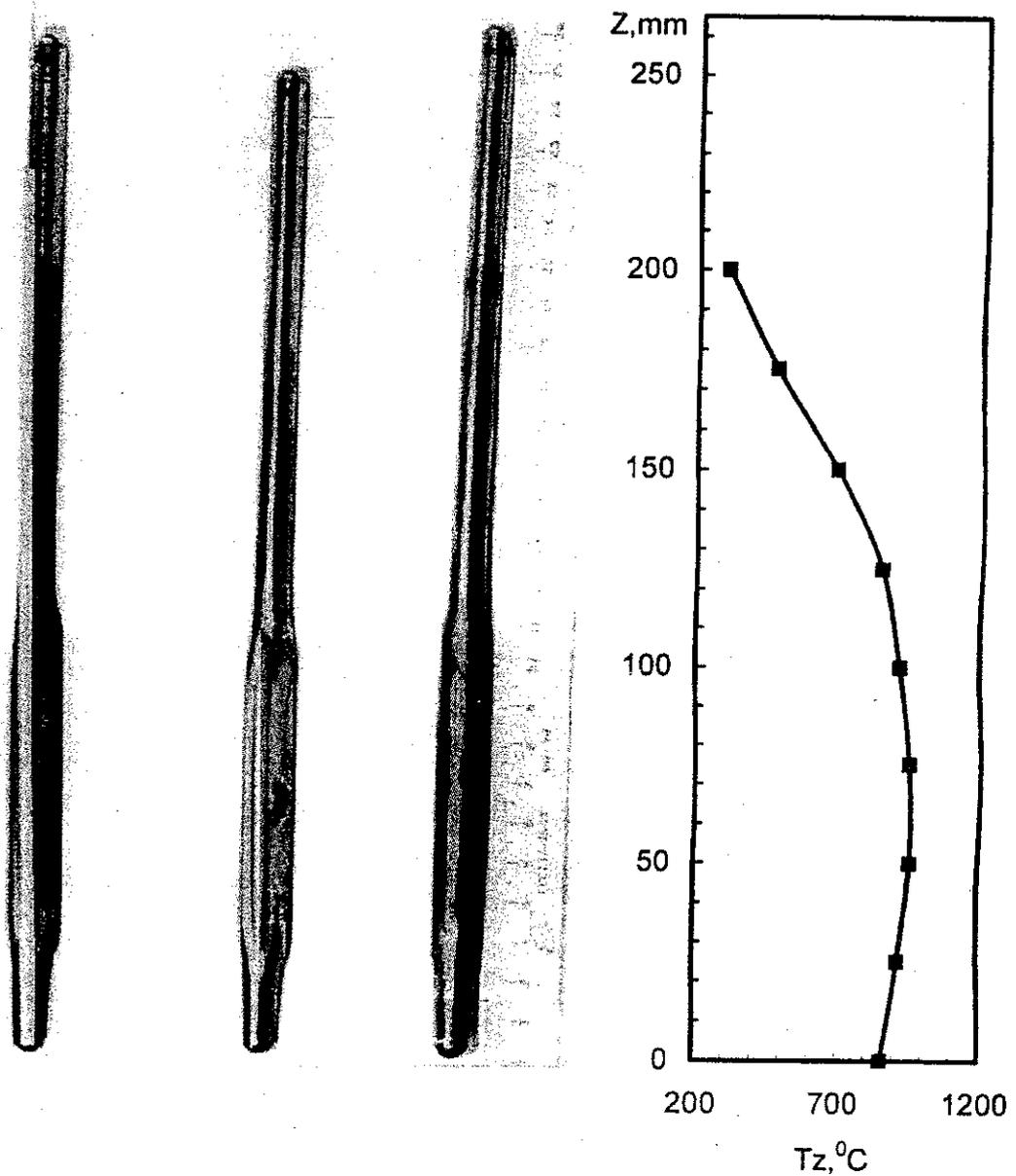


Fig. 2. External appearance and microstructure of Zr-1%Nb alloy cladding after at deformation restriction ($T_z^{\max} = 960^\circ\text{C}$, $\tau = 300\text{s}$, He(1.8 MPa), Ar+H₂O (0.1 MPa)).

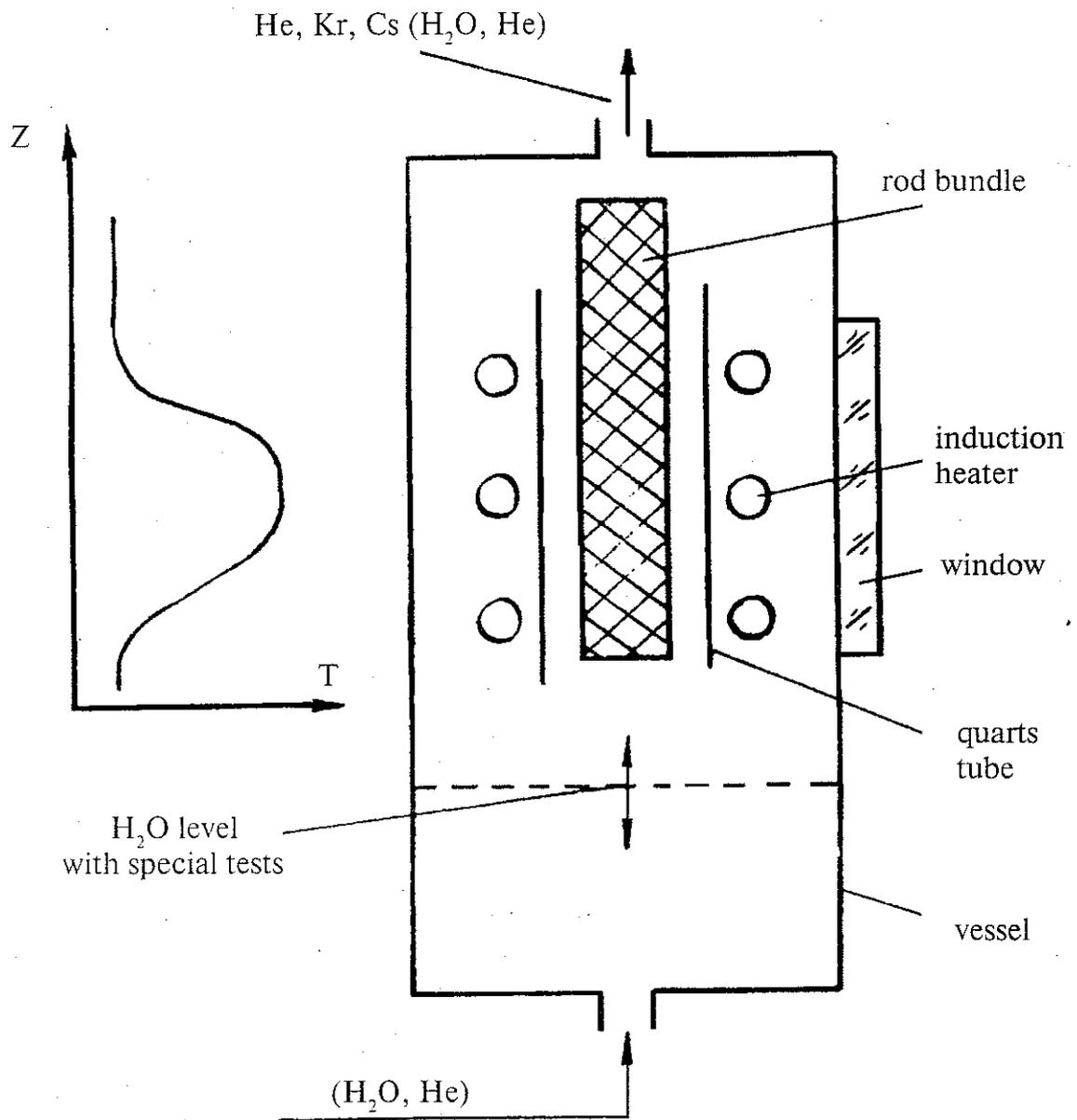


Fig. 3. Schematic diagram of stand for over-heating of rod bundles.

Table 1

TYPE AND PARAMETERS OF TESTS IN INDUCTION FURNACE

| Parameter | Value | Type | Registered parameter | |
|------------------------------------|--|--|--------------------------|--|
| | | | In test | After test |
| Max. temperature | $\geq 1900^{\circ}\text{C}$ | 1. Cladding deformation and depressurization | $\Delta d=f(T, \tau)$ | |
| Heating rate | $1 \div 20^{\circ}\text{C/s}$ | | $DP=f(T, \tau)$ | |
| Length of uniform temperature area | 80 mm | 2. Cladding oxidation (one-way, two-way) | $^{85}\text{Kr}=f(\tau)$ | |
| Test time | $1^1 \div 10^4\text{s}$ | | $\text{H}_2=f(\tau)$ | |
| Medium | He(Ar); H ₂ O; He + H ₂ O | 3. Fuel-cladding interaction | | $\Delta_{\text{corr}}=f(T)$ $\sigma, \delta=f(T)$ |
| Flow rate: | He(Ar) H ₂ O | 4. Flooding | | - "- |
| | | 5. Fuel oxidation | $DP=f(T)$ | - "- |
| | | 6. Cladding melting and runing off | $^{85}\text{Kr}=f(\tau)$ | $\Delta_{\text{corr}}=f(T)$ |
| | | 7. Eutectic formation and runing off | Kinetics | |
| | | 8. Interact. of melts with steel, concrete and water interaction | Kinetics | $\Delta_{\text{corr}}=f(T)$ |