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Theoretical and Experimental Heat Conduction in a Cladded Rod with Internal Heat Generation and Non-Uniform Surface Cooling.

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THEORETICAL AND EXPERIMENTAL HEAT CONDUCTION IN A CLADDED ROD WITH INTERNAL HEAT GENERATION AND NON-UNIFORM SURFACE COOLING

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ملخص: يهدف البحث الى اختبار معملى لصلاحية نموذج تحليلى ثنائى البعد وضع لحساب توزيع درجات الحرارة فى أعمدة مغلفة، يتولد فيها حرارة داخلية والتبريد حولها غير منتظم، كما يحدث فى المفاعلات النووية . ولقد اجريت التجربة على عمود يسخن كهربيا ويبرد بالحمل الطبيعى، من جهة فى الهواء ومن جهة فى الماء .

ABSTRACT

The purpose of this work is to examine experimentally the validity of a two-dimensional analytical model proposed for prediction of the temperature distribution in a nuclear fuel rod in steady and transient conditions. Cooling around the fuel rod is considered non-uniform. The reactor fuel rod is replaced by an elecrically heated rod inserted in a cladding tube. The tube is cooled by natural convection in air at one side and by natural convection in water at the other side. Agreement between the theoretical and experimental temperature distribution of the tube surface prove the adequacy of the proposed analytical model.

INTRODUCTION

Reactor safety is concerned with investigation of the thermohydraulic problems arising in transient and/or emergency conditions of nuclear reactors. These conditions are departures from normal operating conditions that are not expected to occur in the lifetime of any particular reactor, but which may be expected to occur a few times within a large number of reactors over 30-40 year period [1]. Examples would be minor or major loss of primary coolant or minor breaks in the secondary steam side [1,2]. In this case, the reactor design should be such that safe shutdown is achieved without any offsite consequences. Decay heat in the reactor core must be removed without overheating of the fuel to avoid formation of hydrogen bubbles as a result of chemical reaction between zircalloy cladding and steam [1]. In other cases, the cladding material may be subjected to high temperatures and pressure difference. The combined effect of such factors may lead to local ballooing or swelling of the clad which means non-uni cooling arround some fuel rods.

In this work, it is assumed that the fuel rod in a pressurized water reactor is subjected to a non-uniform cooling process(hydrogen or steam cooling and water cooling). For this purpose a test device has been designed and constructed to simulate this situation partly. Temperature distribution in the test section is obtained theoretically using the proposed model. Comparison between the obtained theoretical and experimental results is than made.

THEORETICAL ANALYSIS

Soild reactor and radioisctopic fuel elements (and such others as electrical resistance heaters) are devices in which heat is both generated and conducted. On the other hand, there is no heat generation in reactor fuel claddings. The problem of temperature distribution in and heat removal from these elements are important in evaluating reactor performance. In the considered case (cylindrical fuel element), heat is generated and conducted in the fuel material, then conducted in a gas gap between the fuel and cladding, and then transferred by convection to a coolant flowing around the cladding. It is assumed that the cooling process is not uniform around the cladding. To solve this problem analytically, a two dimensional model is developed which is described in details in reference [3]. Briefly, the cross-sectional area of the fuel rod is divided into differential elements in the two polar coordinates r and θ as shown in Fig. 1. Differential equation governing heat conduction with internal heat generation is applied in each element and then integrated with respect to the directions r and θ and the time (t).

$$\rho c(\partial T/\partial t) = (1/r)\frac{\partial}{\partial r}\left(rk\frac{\partial T}{\partial r}\right) + (1/r)\frac{\partial}{\partial \theta}\left(\frac{k}{r}\frac{\partial T}{\partial \theta}\right) + S$$

A system of discretization equations in the following form is then obtained:

$$A_{i,j}\, T_{i,j} = CR_{i,j+1}\, T_{i+1,j} + CR_{i-1,i}\, T_{i-1,i} + CT_{i,j+1}\, T_{i,j+1} + CT_{i-1,j}\, T_{i,j-1} + d_{ij}$$

where

$CR_{i,i+1}$ and $CT_{j,j+1}$ = radial and tangentional heat conductance between the faces i,i+1 and j,j+1 respectively,

$A_{ij} = CR_{i,i+1} + CR_{i-1,i} + CT_{j,j+1} + CT_{j-1,j} + (\rho c \Delta v/\Delta t)$,

S = thermal source strength, and

$d_{ij} = S\Delta V + (\rho c \Delta v/\Delta t)\, T^{o}_{i,j}$

Solution of the system of discretization equations is then obtained using line-by-line method. The following boundary conditions are applied:

$(\partial T/\partial r)_{r=0} = 0$ (an assumption which holds exactly for uniform cooling) , and

$$-k(\partial T/\partial r)_{r_o} = \alpha (T_c - T_f)$$

where

T_c = cladding surface temperature

T_f = coolant temperature

α = heat transfer coefficient between coolant and cladding

At the diameter AC - Fig. 1.

$$(\partial T/\partial \theta)_{\theta=0} = (\partial T/\partial \theta)_{\theta=\pi} = 0$$

As an initial condition, grid point temperatures are considered to be the same as the coolant temperature.

EXPERIMENTAL APPARATUS AND MEASUREMENTS

To examine the adequacy of the proposed two-dimensional model some experiments are performed in which a fuel rod is replaced by an electrically heated rod inserted in a stainless steel tube. The tube is cooled from one side by natural convection in air and from the other side by natural convection in water.

A schematic diagram of the experimental appearatus is sketched in Fig. 2. The test section is consisted of a stainless steel rod (3) of 75 cm length and 8.1 mm diameter. The rod is inserted and fixed within a stainless steel tube (4) by two insulating rings (5). The stainless steel tube has length of 50 cm, 11.1 mm outside diameter and 9.4 mm inside diameter. The stainless steel rod acts as an electric heater by high DC current supplied to it by a rectifier unit of the type MERA 900 with maximum current of 900 amperes and a maximum voltage of 65 volts.

The test section is fixed horizontally at the center of a plexiglas pool (2). The pool is kept containing distilled water at a certain level in which it makes the lower half of the test section immersed in the water, while the upper half remains surrouned by the air as shown in Fig. 2. This geometry simulates a reactor fuel rod which may be subjected to non-uniform cooling rates. The test section is connected to the power supply by two copper terminals (8).

The surface temperatures of the stainless steel tube are sensed by five thermocouples (6) distributed on its periphery at different positions as shown in Fig. 2. Two thermocouples out of the five; one is located at the most top (air side), while the other is placed at the most bottom (water pool side). The third one is fixed on the tube surface just at the level of the water surface. The remaining two thermocouples; one is placed in between the top one and the third one, while the other is located between the bottom thermocouple and the third one.

The water pool temperature as well as the ambient temperature are also sensed by two thermocoples (17) as shown in Fig. 2. The used thermocouples are copper-constant made from 30 gauge wires. These thermocouples are connected to a temperature recorder (10) with an accuracy of about 1°C.

Using the measured values of electric current and voltage drop across the stainless steel rod, the electric power can be estimated.

RESULTS AND DISCUSSION

The experiment is carried out under the following conditions:

- Steady-state conditions.
- The electric heating power is 434 W which corresponds to a a volumetric thermal strength of 17 MW/m^3, and
- Ambient temperature and water pool temperature (T_a and T_w) are 23 and 35 C respectively.

To obtain the value of the heat transfer coefficient in the air gap between the heating rod and the cladding tube, air is assumed to be stagnent in the gap, which means heat is assumed to be transferred through the gap by conduction. Thermal conductivity of air at 150°C is about 0.03707 W/m.C, and the air gap is 0.65 mm thick, then one obtaines 57 W/m^2.C as a value for the heat transfer coefficient in the air gap.

On the other hand, the information which are given as input data to get the theorctical temperature distribution in both an actual nuclear fuel rod and in the experimental test section are listed in the table [4, 5].

Using these data and applying the analytical model, one obtaines the peripheral temperature distribution of the test tube. Results are shown in Fig. 3 plotted on the same graph with the experimentally obtained data. According to the figure, there is a good agreement between the theory and experiment. The theoretical temperature distribution is obtained for values of the heat transfer coefficients in air and water to be 6-10 and 100-120 W/m^2.C, where heat is transferred by natural convection [4,6]. From Fig. 3, the theoritical temperature on the top point of the test tube (air side) is found to be 136-147°C, where the experimentally obtained value for this temperature is 146°C. In water side, the theoretical value of the tube surface temperature lies between 128 and 140°C where the experimental value is 128°C. These neglegable discripances can be explained as due to the uncertinty in the values of the heat trnasfer coefficients in air and in water. Fig. 4 illustirates the qualitative picture of natural convection around the test section.

Data	Actual fuel rod	Test section
Density, Kg/m^3		
fuel rod	10200	7800
cladding	6300	7800
Specific heat, J/Kg.K		
fuel rod	296	480
cladding	319	480
Thermal conductivity, W/m.K		
fuel rod	2.59	55
cladding	15.13	55
Outside radius of the cladding (R_o), mm	4.68	5.55
Fuel radius (R_F), mm	4.0985	4.05
Cladding thickness, mm	0.50	0.85
Volumetric heat generation, W/m^3	7.2×10^8	1.7×10^7
Gap heat transfer coefficient, $W/m^2.C$	4500	57

Fig. 5 shows the theoretically obtained temperature distribution in the heating rod of the test section and in an actual nuclear fuel rod of a 900 MWe pressuried water reactor [3]. The figure indicates that the temperature gradient in an actual fuel rod much higher than in the experimental heating rod. This is because the value of the thermal conductivity of the rod material (stainless steel) is much higher than that of the fuel material. In addition, the volumetric thermal source strength in an actual fuel rod is much higher than in the experimental rod. The same conclusion can be obtained using the simple one dimensional equation for heat conduction with internal heat generation and given in the following form [7]:

$$t_m - t_s = q''' R^2/4 K$$

where t_m is the temperature in the center line of a cylindrical rod and t_s is the surface temperature of the bare rod having a radius (R) and q''' is the volumetric heat generation rate. Substituting for q''', R, and K one obtaines a temperature difference of less than 1.27 °C for the experimental heating rod.

CONCLUSION

From the above discussion, the following conclusions may be drown

1. Values of the heat transfer coefficients in air and in water affects the surface temperature distribution of the cladding tube.
2. Temperature distribution inside the experimental heated rod is more flat than that inside an actual nuclear fuel rod due to the high thermal conductivity of the experimental rod material and because of the low heat source strength in the experiment.
3. Agreement between the experimental and theoretical results prove the validity of the proposed two-dimensional analytical model in steady-state operation.

NOMENCLATURE

A,d	Occasional coefficients
c	Specific heat
CT,CR	Heat conductance in tangentional and radial direcions.
S, q'''	Volumetric thermal source strength
T	Temperature
t	Time
K	Thermal conductivity
v	volume
ρ	density
α	heat transfer coefficient

Subscripts

c	Cladding
f	Fuel, fluid
a	air
w	water

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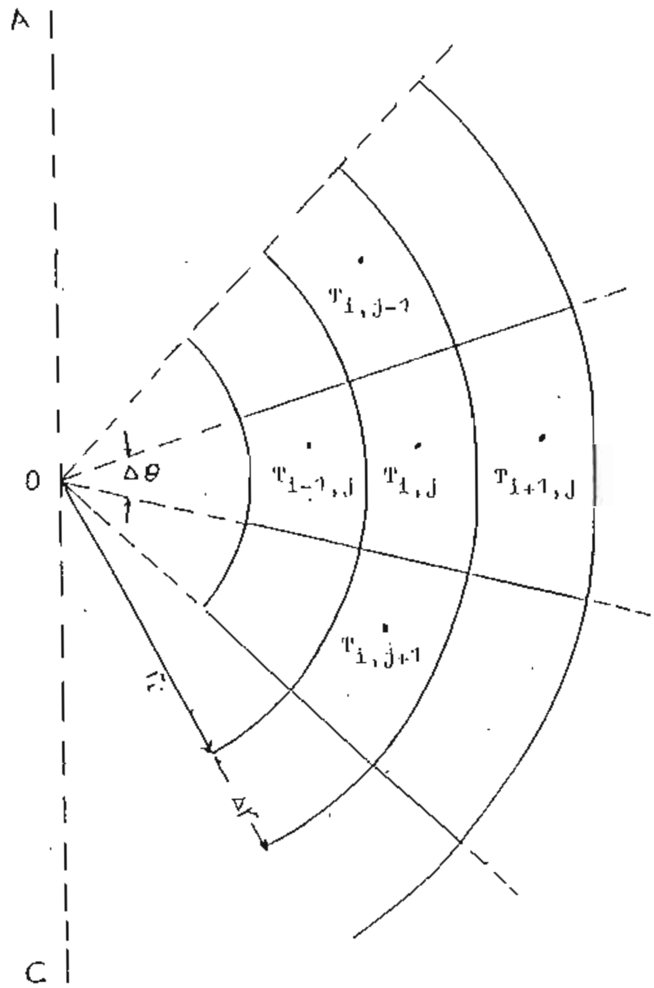
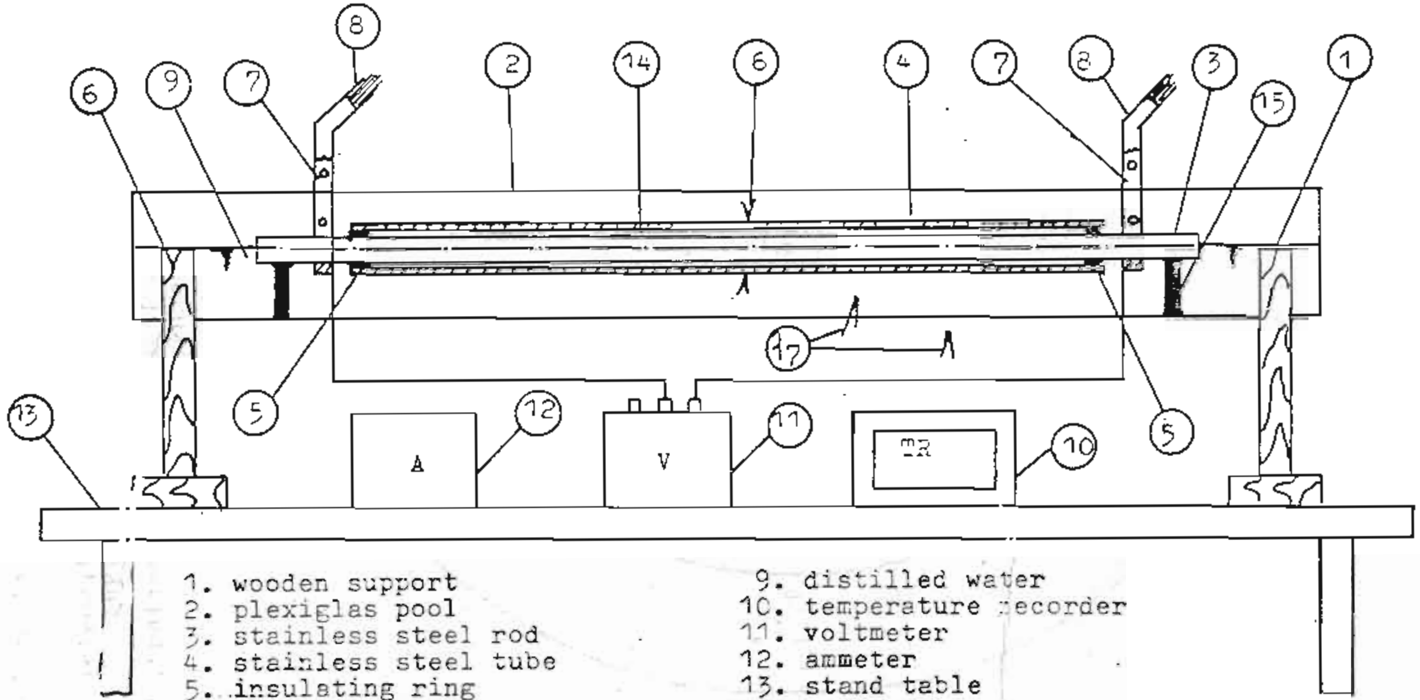


Fig.1 Grid design



1. wooden support
2. plexiglas pool
3. stainless steel rod
4. stainless steel tube
5. insulating ring
6. thermocouples
7. cable connection
8. electric cable
9. distilled water
10. temperature recorder
11. voltmeter
12. ammeter
13. stand table
14. air gap
15. insulating support
16. convection stream lines

Fig.2 Schematic diagram of the experimental apparatus

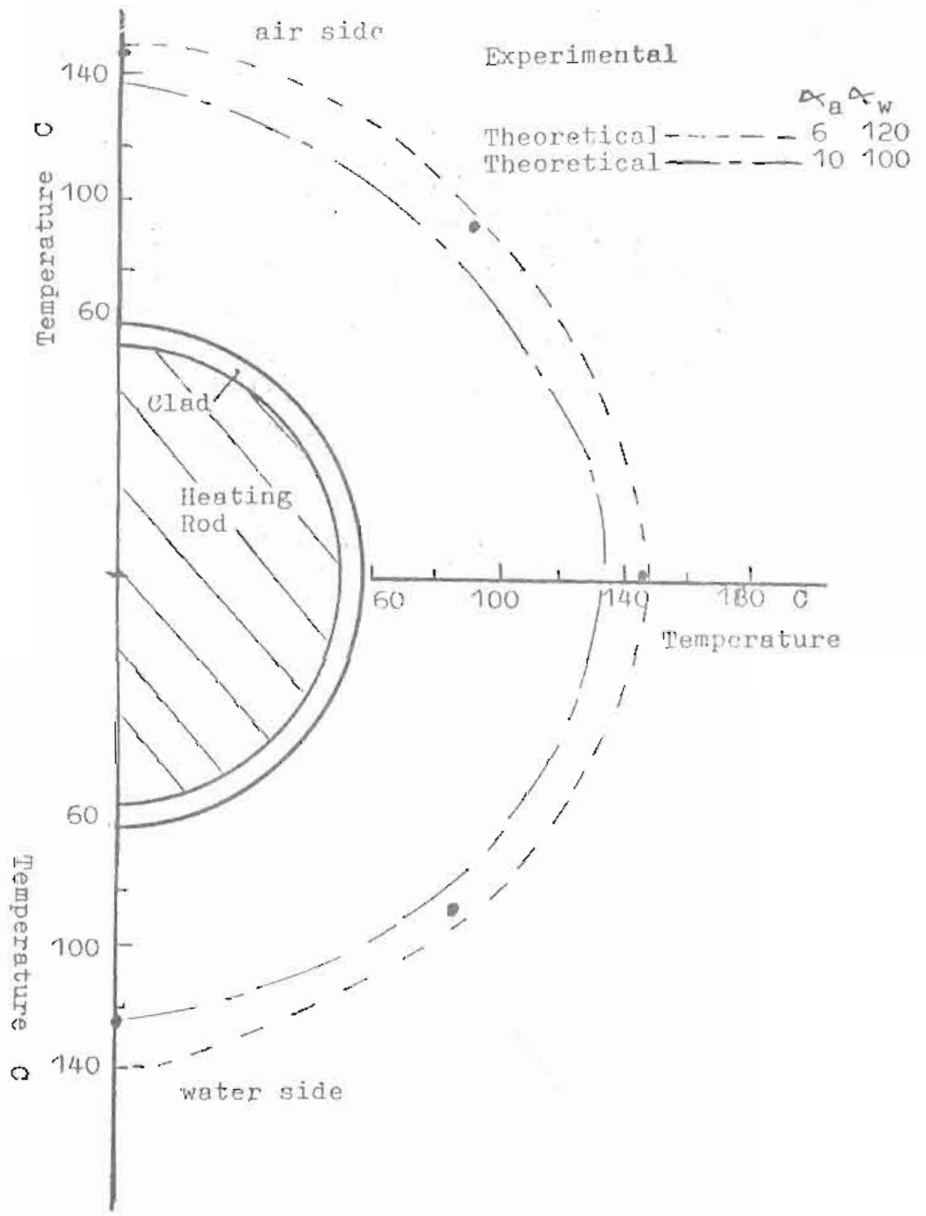


Fig.3 Experimental and theoretical temperat... distribution on the surface of the clad tube

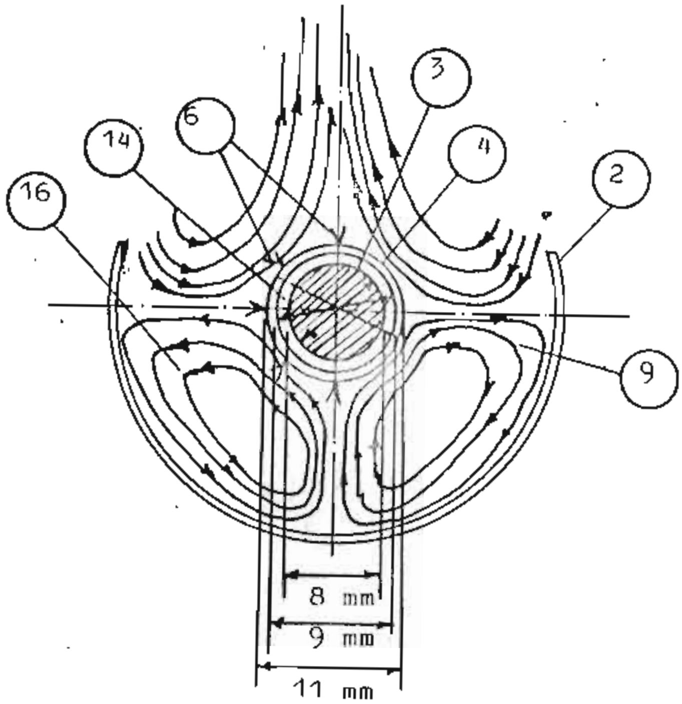


Fig.4 Qualitative picture of natural convection around the test section

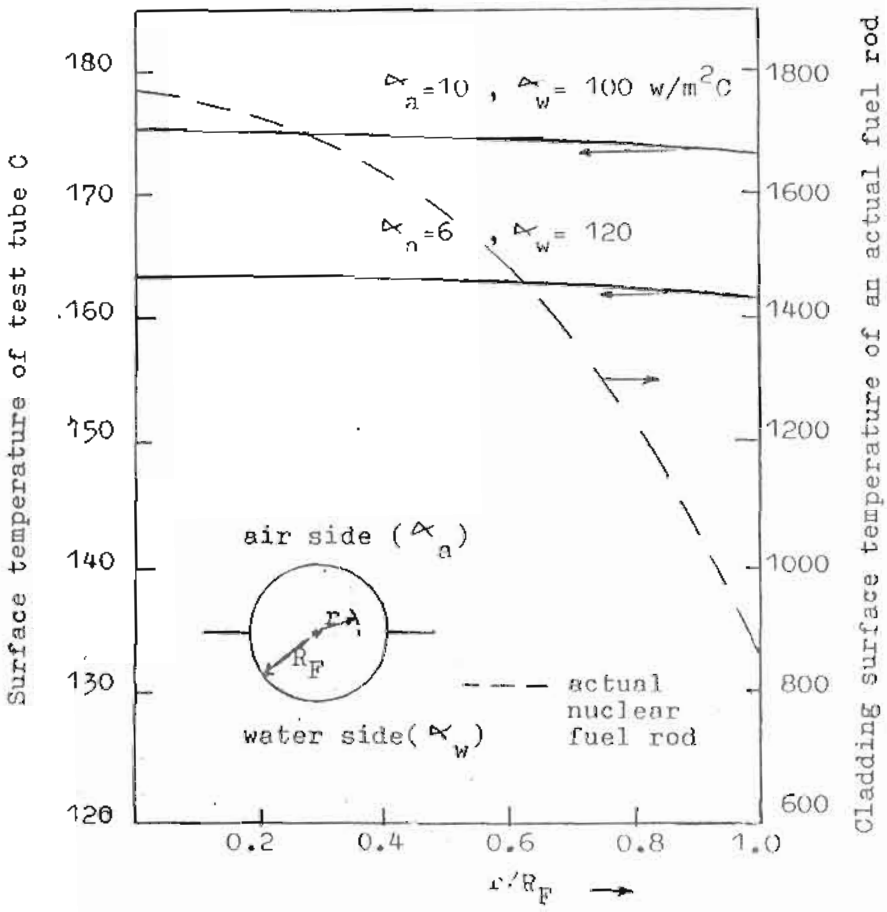


Fig.5 Theoretical distribution in the heating rod and in actual nuclear fuel rod