INTERNATIONAL CONFERENCE ON NUCLEAR POWER AND ITS FUEL CYCLE
SALZBURG, AUSTRIA • 2–13 MAY 1977
INTERNATIONAL ATOMIC ENERGY AGENCY

IAEA-CN- 36 / 549

Study on the Safety of Light Water Reactor Fuel

A.Morishima, M.Ishikawa and S.Suguri

Japan Atomic Energy Research Institute

Tokai-mura Ibaraki-ken Japan

Extensive inpile experiments on fuel behavior during transient overpower and long term irradiation have been carried out by JAERI for the purpose of developing and verifying the computer codes to predict fuel behaviors in power reactors. Those are the experiments in NSRR, JMTR and HBWR, and many fruitful results have been obtained.

1. NSRR experiment

The NSRR is a pulsed reactor in which RIA conditions in LWR and FBR can be simulated. The reactor is provided with a large central experimental cavity, where test fuel elements, one to several, contained in a capsule or a loop are installed and exposed to a high pulse power. The maximum pulse made by quick withdrawal of the poison rods produces enough heat in test fuels to lead to failure. The pulsing characteristics of the NSRR reactor are described in Table I.[1]

More than 70 tests have been carried out in the NSRR since Oct. 1975. The test fuel elements are pelletized fresh UO_2 clad with zircalloy tube, and they are fitted into the capsule with water of ordinary condition. So far as tests of non-pressurized fuel elements and waterlogged fuel elements are concerned, the results are in good agreement with the past SPERT data in general.[2]

Namely, no changes are observed in the tests at heat depositions[1] of up to 140 cal/gram of UO_2. The cladding tube is oxidized in the test at 175 cal/gram of UO_2. In this case, pellets expand and get into contact with the cladding tube, and the cladding surface temperature rises considerably with film boiling initiation as shown in Fig.1. A fuel rod fails at the test at 265 cal/gram of UO_2 with circumferential cracks penetrating the cladding tube. The crack occurs at the location of pellet stack boundary in general. At 335 cal/gram of UO_2, fuel rod is broken into five pieces(Fig.2), but no mechanical energy generation is observed in this test. In the heat deposition range over 380 cal/gram of UO_2, UO_2 pellets are melted and dispersed into water becoming into fine particles(Fig.3), and this results in mechanical energy generation.

Based on the detailed observation of test fuel, it is believed that the fuel failure at more than 335 cal/gram of UO_2 is caused by melting in cladding tube, and that the failure may occur while or after a fuel rod is

1. Heat deposition appeared in this report includes 20~30 cal/gram of UO_2 heat generation which follows after power burst.

being quenched. This may be the reason why mechanical energy is not generated in these tests. In addition, it is also believed that the failure at higher heat deposition is caused by combined effects of reduction in mechanical strength of cladding tube and increase in internal pressure owing to melting of UO_2, and the dispersal of melted fuel into water makes a rapid steam formation that generates mechanical energy release.[3]

Failure of waterlogged fuel elements is caused by the internal pressure increase. In defective fuel elements, so much water is trapped inside the cladding tube that rapid heat generation in pellets increases internal pressure through pellet expansion and steam formation, and this results in the burst of the cladding tube(Fig.4). The waterlogged fuel elements generally fail at the heat deposition of approximately 110 cal/gram of UO_2, when internal pressure goes up as high as 1,200 kgf/cm^2(g) but cladding temperature rise is limited to less than 300°C. In case of such a failure, most of fuel pellets are released into water becoming into very fine particles, and this results in generation of some amount of mechanical energy.[4]

Failure of the pressurized fuel elements is also caused by internal pressure rise, but conditions for and aspects of fuel failure are significantly different from those of waterlogged fuel(Fig.5). In case of the fuel element whose initial internal pressure is 30 kgf/cm^2(g), failure threshold is estimated to be 150 cal/gram of UO_2. At this heat deposition, cladding temperature goes up to 800°C but internal pressure does not exceed 100 kgf/cm^2(g). No fuel meat is dispersed into water and thus no mechanical energy is induced.

Tests with highly enriched fuels, tests with bundled fuel elements with flow shroud, test with a water-filled capsule which simulates PWR reactor condition, test with various subcooling of water are being performed. Tests with highly enriched fuels indicate that the heat deposition in the outer region of the pellets significantly influences the cladding temperature and thus controls the fuel failure in an RIA condition. Tests with high pressure and high temperature capsules which simulate reactor hot stand-by conditions, and tests with loops which simulate reactor operating conditions are also planned. Approximately 500 experiments are scheduled untill 1981, using a total of 1,000 fuel elements, as the first phase of the test program in the NSRR.

2. Tests in JMTR and HBWR

Experiments to study the effects of long term irradiation on fuels have been carried out in the JMTR and the HBWR since 1967.

In the JMTR experiments, two kinds of tests were performed for the purpose of studying the densification phenomena. The first tests were performed in capsules to study fundamental phenomena of densification by varying influential parameters on them. Those are pellet density, grain size and pore size distribution, and a total of seventy six UO_2 pellets were irradiated in a capsule with the condition of the maximum fuel center temperature of less than 1,000°C. The second tests were performed in a loop, where the maximum pellet temperature reached as high as 2,000°C, in order to simulate rather practical irradiating conditions of fuel in power reactors.

Post irradiation examinations are being progressing on these experiments, and some of the results on 95%TD UO_2 pellets have been obtained as shown in Figs.6 and 7. Fig.6 shows porosity change with irradiation. Porosities are measured by the immersion method with meta-xylene. The porosity significantly changes with the grain size as far as the grain size is smaller than 10 μm. This result is in good agreement with Chubb's experimental results.[5] From this result, it can be said that densification hardly occurs when the grain size exceeds 10 μm.

Effects of the pore size distribution on the porosity change are also examined by using an optical micrograph and image analyser, as shown in Fig.7.

The figure also shows that distribution of pore size is changed with irradiation and that pellet with grain size over 2 μm is lost by densification.

In the HBWR tests, a total of twenty three Japanese fuel assemblies have been irradiated since 1967, producing a considerable amount of information. From these tests, IFA-224 fuel assembly test has been applied to the verification work of the FREG-3 code, because fuel center temperature were attained succesfully and fuel pellets of different densities were assembled in this test.

FREG-3[6] is a code to calculate precise temperature distribution in a fuel element in accordance with irradiation histories so that it can accurately evaluate the stored energy in fuel. FREG-3 is programmed to allow all existing functions, corelations, empirical equations, models and so forth concerning material properties and physical states in order to make comparison of the calculated results among them. This is one of the way to have the best evaluation model.

Calculated results on IFA-224 experiment by the FREG-3 is shown in Fig.8. Good agreements are obtained on fuel center temperature between the tests and the calculations in the range of irradiation rate of up to 10,000 Mwd/t of UO_2 under the heat rate up to 600 w/cm when RESAR-41[7] model is used for gap conductance calculation, Rolstad's model in MATPRO[8] for pellet densification calculation and GAPCON-THERMAL-2[9] for relocation model.

REFERENCES

[1] ISHIKAWA,M.,et al.,"Some experiences and plans on reactor safety research by NSRR",1st US/Japan seminar on fast pulse reactors, Tokai,Japan(1976)

[2] ISHIKAWA,M.,et al.,Quarterly progress report on the NSRR experiments(1) (Oct.1975~Mar.1976),Japan Atomic Energy Research Institute Report JAERI-M 6635(1976)

[3] ISHIKAWA,M.,"First progress report of the nuclear safety research reactor (NSRR) experiments",4th LWR safety information meeting, Washington(1976)

[4] ISHIKAWA,M.,et al.,Quarterly progress report on the NSRR experiments(2) (April 1976~June 1976),Japan Atomic Energy Research Institute Report JAERI-M 6790(1976)

[5] CHUBB,W.,HOTT,A.C.,ARGALL,B.M.,KILP,G.R.,Nuclear Technology 26(1975)486

[6] HARAYAMA,Y.,et al.,"User's guide for FREG-3 : A computer program to analyze pellet-cladding gap conductance in accordance with fuel rod irradiation history",Japan Atomic Energy Research Institute Report JAERI-M 6742(1976)

[7] Westinghouse Nuclear Energy System.,Reference Safety Analysis Report (RESAR-41) Vol.II,Docket-STN-50480-2(1973)

[8] "MATPRO : A handbook of material properties for use in the analysis of Light Water Reactor fuel rod behavior",(MACDONALD,P.E.,THOMPSON,L.B., edit.),Aerojet Nuclear Company REport, ANCR-1263(1976)

[9] BEYER,C.E.,HANN,C.R.,LANNING,D.D.,PANISKO,F.E.,PARCHEN,L.J.,"GAPCON-THERMAL-2 : A computer program for calculating the thermal behavior of an oxide fuel rod",BNWL-1898(1975)

Table I. Performance of the NSRR reactor

Maximum pulsing operation	
Reactivity insertion	3.4 %Δk (4.7 $)
Peak power	21,000 Mw
Prompt energy release	105 Mw-sec
Reactor period	1.12 msec

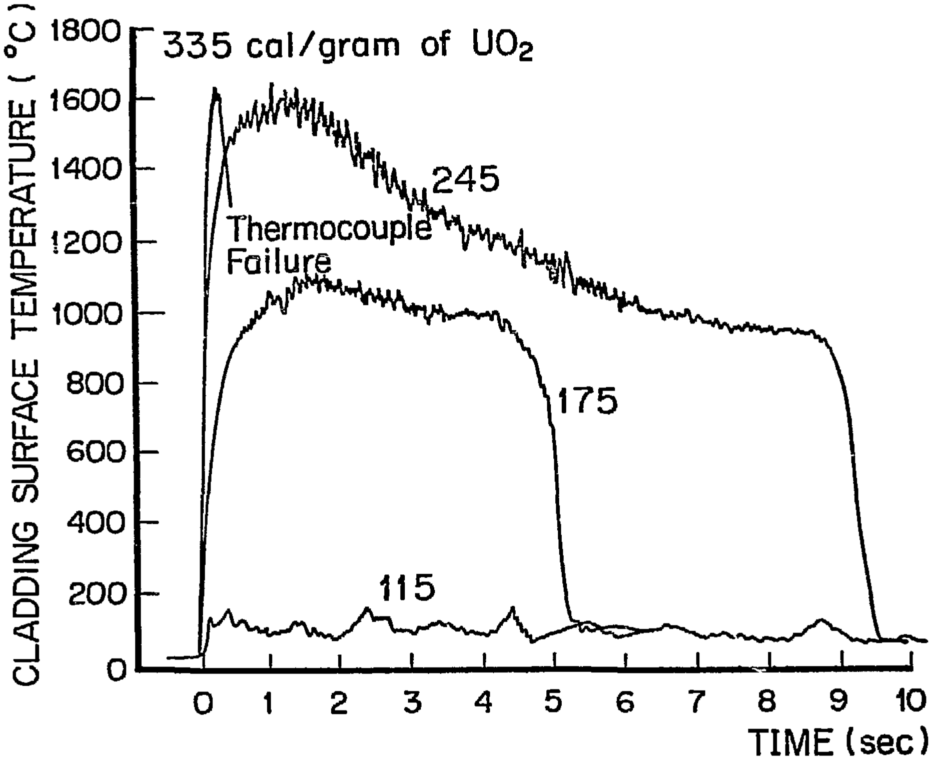

Fig. 1 Cladding surface temperature of non-pressurized fuel elements during scoping tests

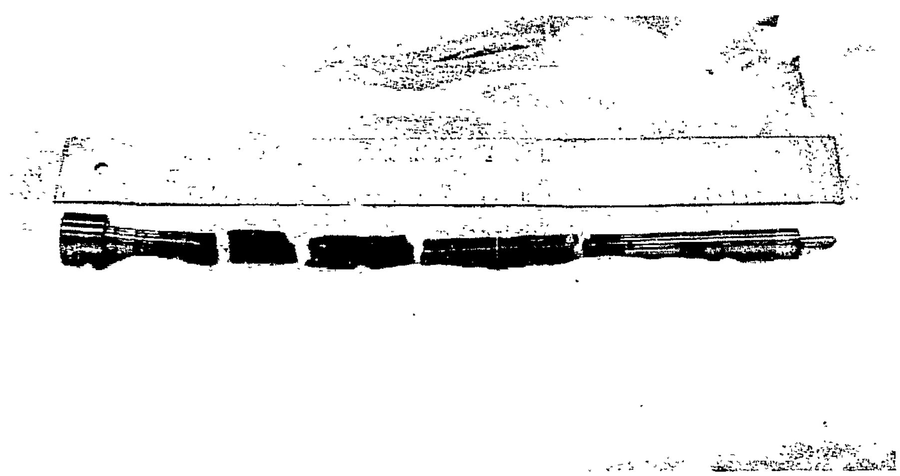

Fig. 2 Failure of a non-pressurized fuel element at 335 cal/gram of UO_2

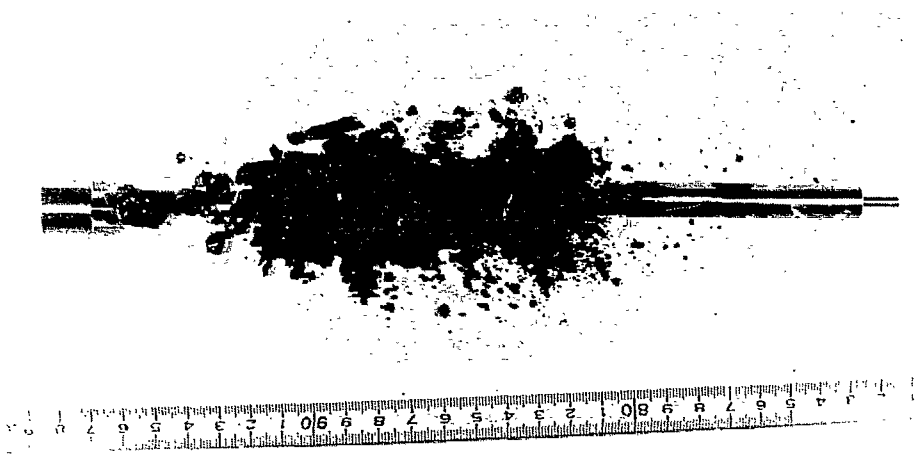

Fig. 3 Failure of a non-pressurized fuel element at 380 cal/gram of UO_2

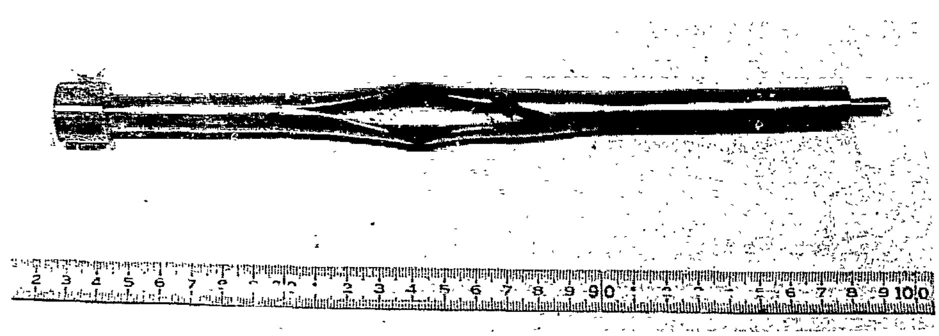

Fig. 4 Failure of a waterlogged fuel element at 110 cal/gram of UO_2

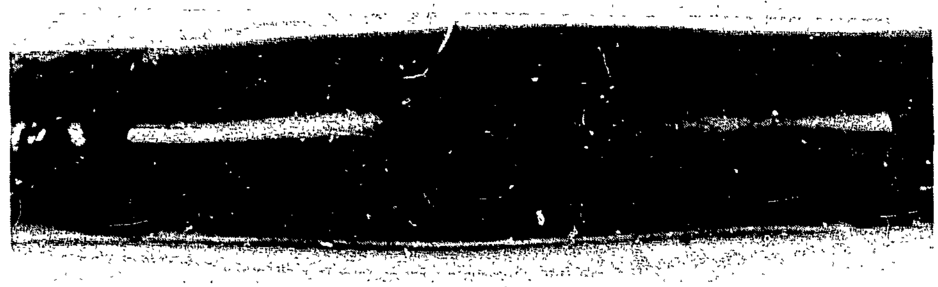

Fig. 5 Failure of a pressurized fuel element at 185 cal/gram of UO_2

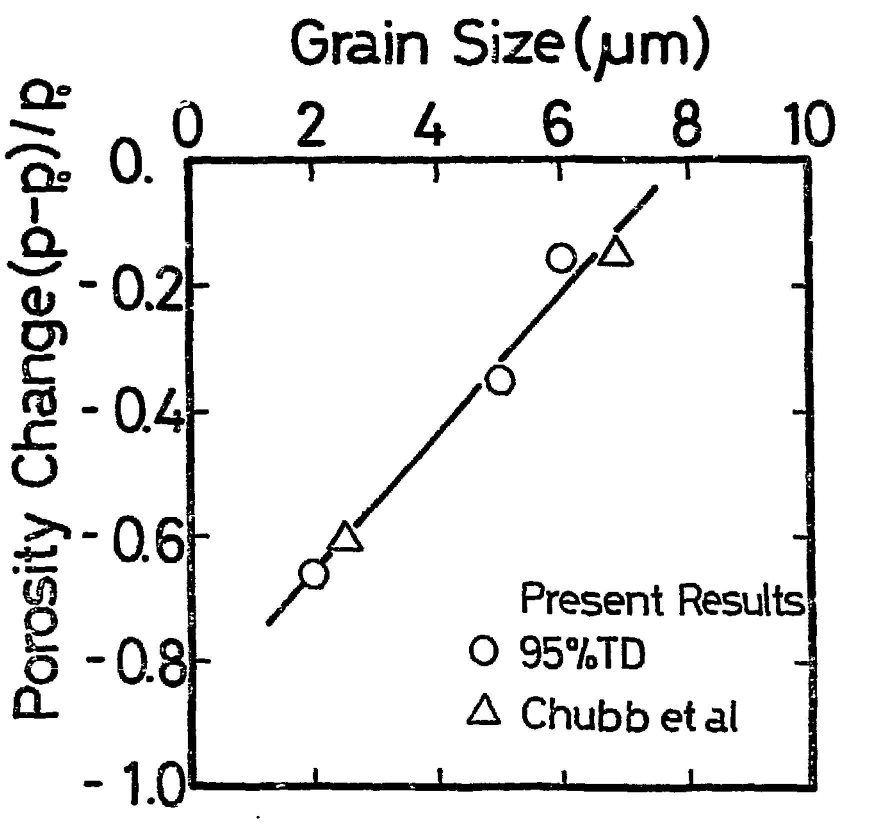

Fig.6 Densification as a function of grain size of UO_2

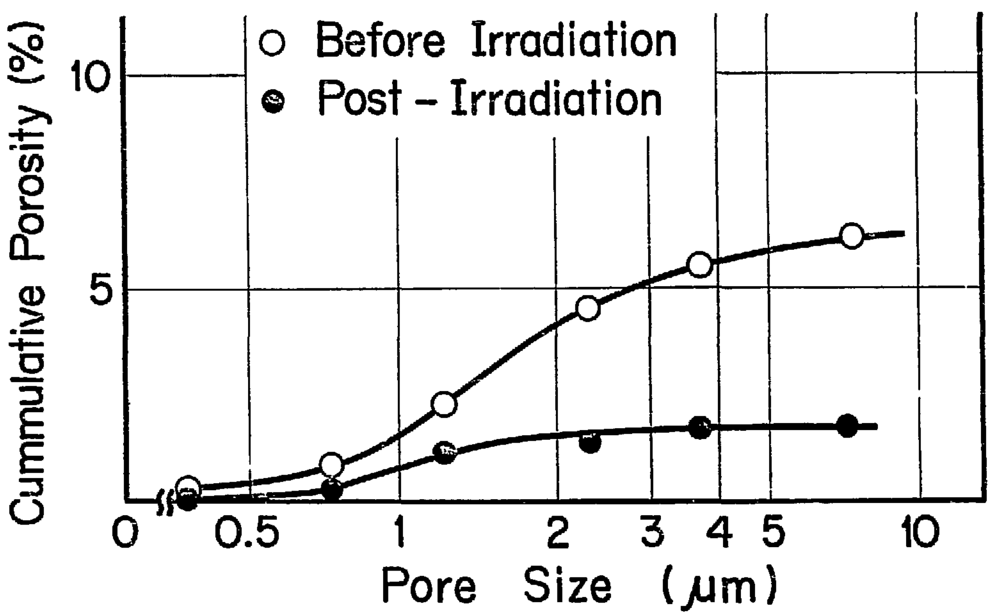

Fig. 7 Cummulative porosity change with pore in a UO_2 pellet as a results of irradiation in JMTR

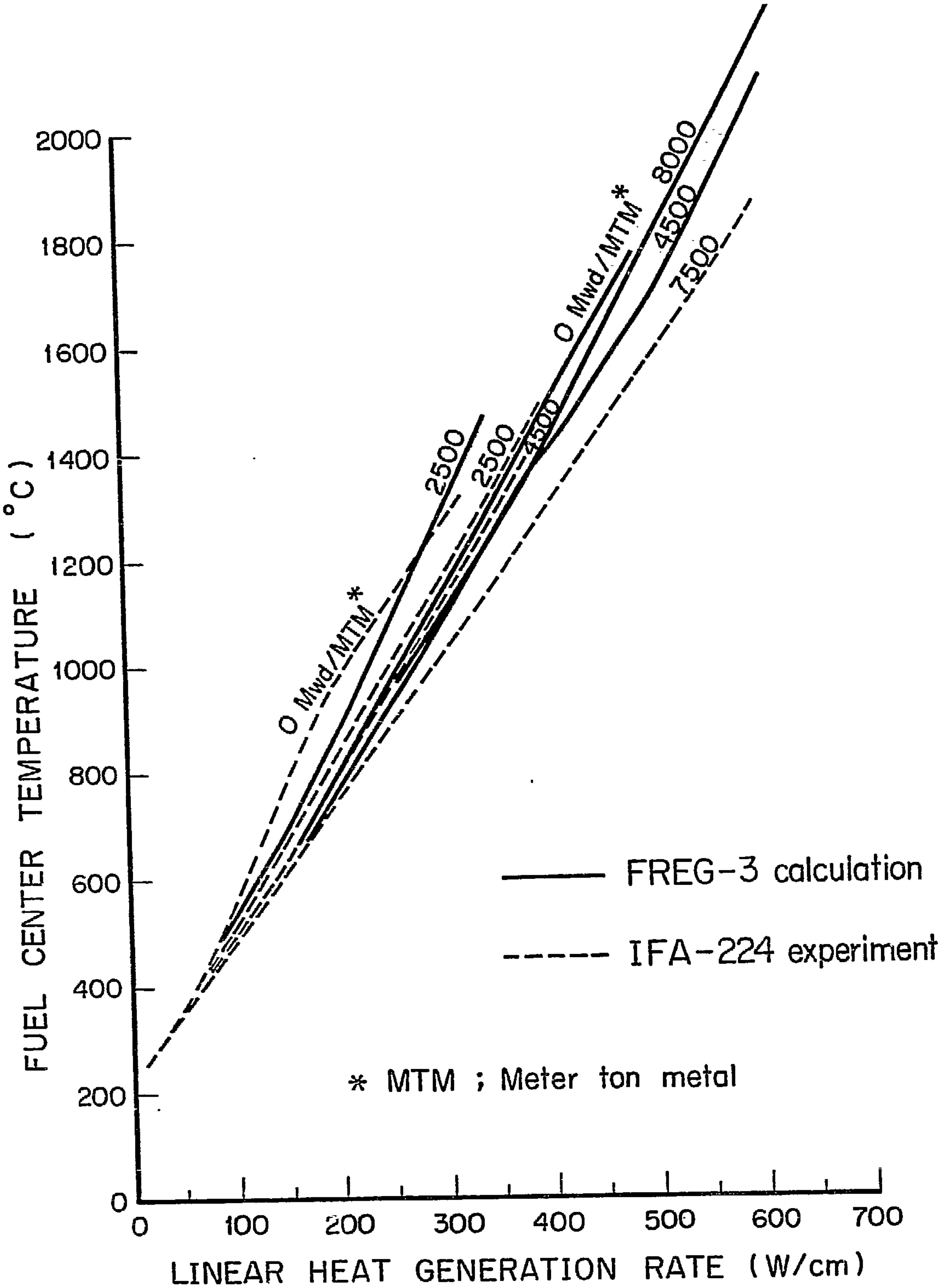

Fig. 8 Calculated and measured fuel center temperature as a function of LHGR