IRRADIATION TEST OF SURROGATED PWR FUEL PIN DESIGN MANUFACTURED WITH nUO_2

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ABSTRACT

PREDICTION OF DIAMETRICAL BEHAVIOR ON PRELIMINARY IRRADIATION TEST OF SURROGATED PWR FUEL PIN DESIGN MANUFACTURED WITH NATURAL UO_2. A prototype of fuel pin design for Pressurized Water Reactor (PWR) has been manufactured in the Experimental Fuel Element Installation (EFEI) at Serpong. Power ramp test facility (PRTF) has been revitalized for testing the prototype performance during power ramp. Base-load irradiation of prototype-2 has been done by simulated power in PRTF. The pertinent parameter of pre-ramp irradiation is pellet and cladding dimensional change up to the point of pellet cladding mechanical interaction (PCMI). Slow ramp power mode has been chosen because the evaluated pin consists of natural uranium pellets. The result shows that dimensional change of pellet and cladding resulting from thermal, mechanical and chemical effect of irradiation occurs until depletion of ^{235}U by burnup below depletion in commercial U enrichment. The maximum fuel temperature is < 60^oC, while minimum gap attained 17% of its initial value. It is concluded that the available power inside the PRTF capsule is still below the requirement for fuel pin test. With reference of measured power distribution of RSG-G.A.S. core 83 near PRTF, the author recommends re-evaluating the available flux data inside PRTF capsule to determine if a second simulation is needed.

Keywords: modeling, PWR fuel, geometry, mechanical interaction.

ABSTRAK

PREDIKSI PERILAKU DIAMETER BATANG PADA UJI IRADIASI PENDAHULUAN DESAIN PIN BAHAN BAKAR PWR DIBUAT DENGAN DENGAN BAHAN UO_2 ALAM*. Sebuah prototipe desain pin bahan bakar untuk Reaktor Air Tekan (PWR) telah diproduksi di Instalasi Elemen Bakar Eksperimental (EFEI) di Serpong. Fasilitas uji daya naik (PRTF) telah direvitalisasi untuk menguji kinerja prototipe selama daya naik. Iradiasi beban dasar dari prototipe-2 telah disimulasikan dengan menggunakan daya sesuai data PRTF. Parameter penting pada iradiasi daya naik adalah perubahan dimensi pelet dan cladding hingga terjadi interaksi mekanik pelet kelongsong (PCMI). Kenaikan lambat daya telah dipilih karena pin dievaluasi terdiri dari pelet uranium alam. Hasil menunjukkan perubahan dimensi pelet dan cladding yang dihasilkan dari efek termal, mekanik dan kimia iradiasi sampai menipisnya ^{235}U dengan fraksi bakar di bawah deplesi pada pengayaan U komersial. Temperatur bahan bakar maksimal <600 ^oC, sedangkan gap minimum mencapai 17% dari nilai awalnya. Dapat disimpulkan bahwa daya yang tersedia di*

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dalam kapsul PRTF masih dibawah keperluan untuk uji pin bahan bakar . Dengan referensi distribusi daya RSG-GAS terukur pada teras 83 dekat PRTF, penulis merekomendasikan untuk mengevaluasi kembali data fluks tersedia di dalam kapsul PRTF, kwmudian menentukan apakah simulasi kedua diperlukan.

Kata kunci: *pemodelan, bahan bakar PWR, geometri, interaksi mekanik.*

INTRODUCTION

To accelerate comprehending and mastering the nuclear fuel fabrication state of the art, in 1980 era, Indonesia has started cooperation with ANSALDO to build a facility called EFEI (Experimental Fuel Element Installation) [1-3]. The first NPP deployment in Indonesia was Soviet technology. Site has been decided, located at Serpong in 1960 and civil construction has been started. The construction has been canceled due to change of national regime. Then the R&D related to nuclear electricity started on HWR (Heavy Water Reactor), which allow natural uranium use for fuel. Expansion of pellet production facility has been accomplished for PWR fuel pellet production and related short PWR pin manufacturing. The expansion has been done [3] in order to adapt the new trend of choosing PWR as first NPP to be build in Indonesia

Fuel cycle: Uranium exploration, laboratory scale exploitation mining, yellow cake production, U and UO_2 production, material and instrumentation, mechanic etc since 1975 in Yogyakarta Nuclear center.

In 1985-1990, some laboratories and pilot scale facilities had been built at Serpong district: RSG-GAS a 30MW MTR (Material Testing Reactor), MTR Fuel Element Production Installation (FEPI), Pilot Conversion Plant (PCP) for yellow cake or UF_6 to UO_2/U_3O_8. Experimental Fuel Element Installation (EFEI) for CIRENE HWR fuel rod. RadioMetalurgi Installation (RMI) for irradiated fuel and material examination, Radioactive Waste Installation (RWI) for handling and management of low and high levels radioactive waste and spent nuclear fuel. There also other facilities of Electro-Spectrometry, Nuclear Mechano-Electrical Installation have been build in this era. The RSG-GAS has been equipped with one loop PRTF (Power Ramp Test Facility) and an equipment of in-pile loop for PWR fuel irradiation testing has been prepared but has not been installed.

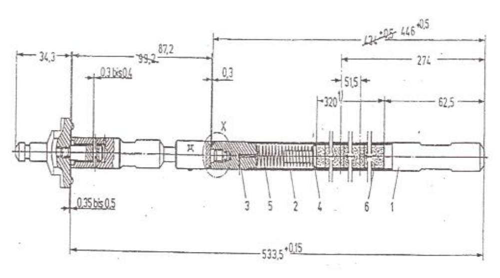

Figure 1. Workshop design of the test pin connected to sample holder of PRTF [2]

Nuclear Fuel is the inner core nuclear reactor which is exposed to the most severe environment that consists of combination of fast neutron, temperature, mechanical and chemical. Out of pile and in-pile test are necessary for qualify the fuel pin. Power ramp test is one of severe in-pile test. Potential failure during ramp power is related to material parameters and irradiation parameters.

The material parameters such as: PCMI, Iodine concentration around inner cladding surface, cladding stress-strain state and cladding mechanical property degra-dation caused by radiation enhanced corrosion and hydriding and pellet property related to microstructure changes such as swelling and cracking. The first four parameters has been considered as prerequisite condition for ramp power failure [4].

The objective of the present study is determined minimum time of irradiation required as pre irradiation state before PCI ramp test and related dependent and independent parameter for the given fuel pin data. The prediction will be used as guide for PIE and as a complement PIE data and in reactor data of irradiation test, as well as scientific support for experiment permit for irradiation test and PIE. Lastly' the numerical study may be used to design optimization of both experiment and fuel rod design.

IRADIATION MODELING

The testing system is shown in Fig. 1 containing (a) drawing of the fuel pin connected to a metal seal, (b) drawing the pin prototype inside irradiation capsule of PRTF capsule which is connected to capsule carrier and coolant system and (c) process diagram of PRTF showing the PRTF capsule, coolant and pressurization system, which system is located inside reactor pool of RSG-GAS.

Cooling water and pressure for irradiation test Fig.1(b) of the PRTF is summarized in [5]: The primary system is a series of stainless steel pipe, 6 mm outer diameter 1.5 mm thickness, 160 bar pressurized water supply to the capsule, for simulating PWR conditions. A pressure of 160 bar is prepared by using helium gas medium. Secondary system has two pumps which work in parallell, pump water into the pond capsule of 1200 l / h. The Reactor water cooler / moderator condition are at atmospheric pressure and temperature at ~ 35 °C. The temperature and pressure inside the PRTF capsule test pin 250 ^{o}C and 150 Atm. To simulate the PWR core condition, inside PRTF neutrons irradiation to the test pin is provided by MTR core, while the temperature and pressure of PRT capsule are provided by PRTF loop system. Neutron fluctuation in the capsule is provided effectively by positioning the capsule close or far to the core.

Table 1. PTBN Pin desaign data extracted From [6]

No.	Table Column Head		
	Fuel Pin Parameter	Value	Unit
0	Rod length	446.3	mm
1	Cladding length	366.5	mm
2	Rod outer diameter	10.75	mm
3	Pellet Length	9.40	mm
4	Pellet outer diameter	8.90	mm
5	Pellet density	10.171	g/cm^3
6	Pellet Material	n. UO_2	-
7	Clad Material	Zry-4	-
8	Clad thickness	0.71	mm
9	Number of Pellet per pin	34	-
10	Fill He gas pressure	1	Bar
11	Plenum volume	2.3764	cm^3
12	Pellet Dish Diameter & Depth	7.7; 0.32	mm

The geometry of fuel pin test is presented in Fig 1. A detailed design not shown allows calculation of pin volume fraction and absolute volume of plenum, gap volume of pellet-to clad at fresh or initial state. This data has been combined to data of Tab.1 [6] represents manufacturing data of test fuel. The fuel properties have been chosen from standard option of Transuranus version M1v1J12 code[7]. Table 2 shows the PRTF and other important data

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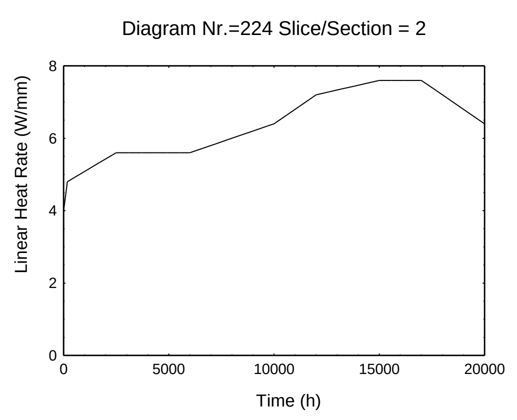

Figure 2. History of loaded average linear heat rate, axial peak location.

The slow ramp power history presented by Fig. 2 has been chosen to give important effect to pellet-cladding gap width. It is related to prerequisites of power ramp test: PCMI, stress-strain state of cladding, Iodine and corrosives chemical products around inner surface of cladding and degraded properties of cladding by corrosion and hydriding [8]. These data and related thermal-mechanical material properties have been input to Transuranus code to obtain radial geometry behavior of pellet and cladding as resulting by thermal, mechanical, metallurgical and chemical change.

The simulation has been conducted by Transuranus fuel code. The input data consisted of material parameter : material specification and properties of pellet, cladding, gap filling gas, irradiation parameters such as power distribution and power history, boundary and initial condition. Different option of solution method and option of properties model are provided by the code which should be chosen for best represent the system to be simulated.

Most important input and calculation output corresponding to on-set time for PCMI are presented for discussion: averaged linear heat rate history, radial distribution of temperature in different irradiation time, accumulated pellet average burnup, radial distribution of burnup and produced and released fission gas, displacement history and change of pin geometry: pellet, cladding and gap size.

RESULT AND DISCUSSION

A. Heat flux history in the cladding

The heat flux rate history at fuel outer diameter surface has been plotted accompanied by cladding heat flux at cladding, axial peak location. The second is an important parameter to corrosion rate and hydriding rate of cladding which degrade its mechanical properties [9]. The curve shape are homolog to linear heat generation rate, because no different of the pellet diameter along the pin. The different heat flux at pellet surface and cladding surface related to its diameter, as the flux is conserved between outer pellet to outer cladding surfaces.

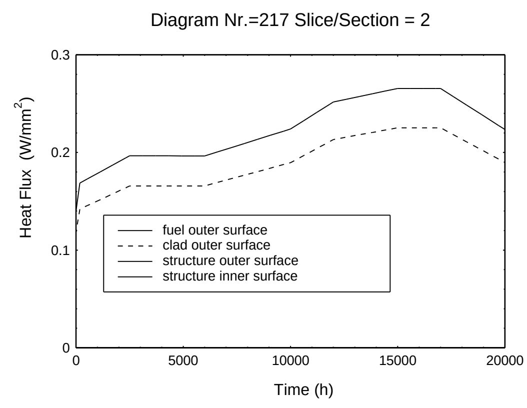

Figure 3. History averaged slice heat flux

Values of LHR at given time are radial-averages. In this case study, the axial distribution of radial-averaged LHR is slice-wise, i.e., invariable along slice of rod. Note that the axial distribution of LHR at given time is not perfectly sinusoidal, the peak is not located in slice 3 or middle rod, but in slice-4. LHR history of slice 5 is lower than slice-4 the highest value. Fig.3 shows history of averaged slice-4 heat flux of 4 different radial position. The distribution curves have higher value at second part of high irradiation time, while in normal operation us higher extreme in the first part of curves. It was preferred for giving higher than power ramp test relative to normal operation.

The temperature of slice-4 at different irradiation time upto 20000 h is plotted in Fig.5. These curves have relatively similar to the history of thermal flux input

Among 4 radial distribution the curve of 2000 h is the highest and the jump temperature is steepest. It correlate to widest gap as pellet densification while cladding creep-down has not yet significant. The temperature increase from 177 h to 2000 h is highest at the center of pellet and decrease continually toward outer surface cladding since the coolant temperature at the point is constant. From 2000 h to 12000 h the pellet temperature distribution decrease nearly homogeneous with slightly higher at pellet surface while in-significant of cladding temperature decrease. These decreases may be related to decrease of LHR. At 20000 h the interior part pellet is hotter than the previous curve but the exterior part of pellet colder than

B. Fissile 235U consumed and ^{236}U + ^{237}Np produced

Fig.4 shows average concentration of fissile ^{235}U (upper curve. Initial concentration is 0.717% and ~50% burnout, generate neutron, energy and other actinides. The average burnup is distributed strongly in radial direction is not presented. The non reacted fissile ^{235}U after 20000 h or ~ 83.33 d or longer than life time of natural uranium oxide pellet in CANDU reactor has been under the ^{235}U concentration of depleted uranium form enrichment plant. So if higher flux is available it will better to evaluate the fuel pin with higher flux.

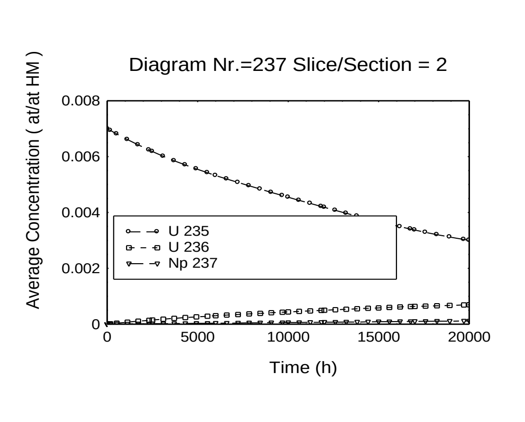

Figure 4. History of average concentration of different initial produced actinide at slice of highest flux

C. History of Pin temperature

Fig. 5 presents average slice temperature change at different point of pin. The temperature at center of pellet change resembles the curve of linear heat rate as heat source. It is slightly damped by diffusion heat transfer to coolant at constant temperature. The other curves of nearer position to the coolant heat sink go flatter, as the temperature of heat sink is constant. It wise to note that the lowest fluctuation of cladding temperature is accompanied by higher fluctuation in heat flux, and both parameters affect corrosion and hydriding rates of cladding material.

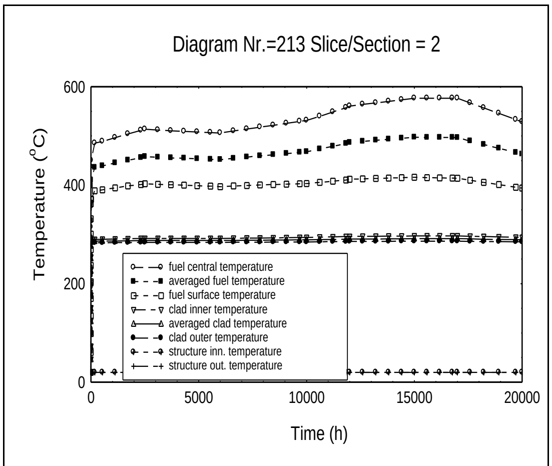

Figure 5. Temperature History of different radial position

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D. Fission gasses produced and released

Produced fission product in solids volatiles and gasses phases cause pellet swelling. Retained fission gas is the most important for pellet swelling as it precipitate as gas bubbles intra and inter grains after saturation of as solid solution. Fig. 6 presents accumulated fission gas in the fuel matrix up to 7.5 □ Mol/mm pellet length, and nearly no fission gas has been released. It is can be related to temperature of 1% threshold released which is depend on burnup, and for 7 MWd/kg the threshold released is 1000 ^{o}C [10]. The maximum pellet temperature is under the threshold, there for no fission gas released is found.

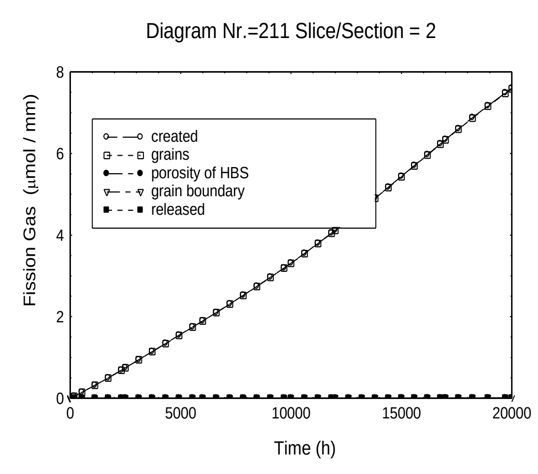

Figure 6. Fission gas produced, with nearly zero released

E. Radial geometry change

Radial geometry change is very important of related to the cause and the impact on fuel integrity. The history of radial geometry during irradiation is shown in Fig. 7. It shown that at the beginning of irradiation, cladding and pellet radius displaced with pellet got the higher displacement. It is related to thermal expansion, where the ceramic pellet has much lower coefficient of thermal expansion than metal alloy and specially the radial thickness of pellet is >15 times of radial clad thickness. The clad displacement then goes negatively nearly the whole of irradiation time which is related to thermal and irradiation creep as the outer pressure is about 15 MPa, while inner pressure of pin only 0.1 MPa and its temperature is about 250-300 ^{o}C.

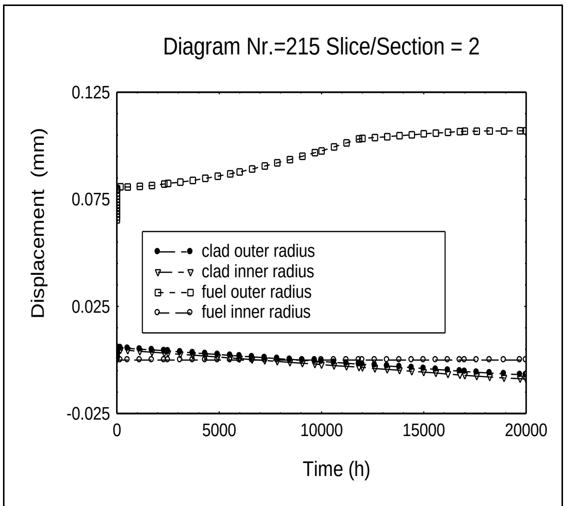

Figure 7. Diameter change of outer, inner clad and pellet during irradiation

The pellet outer radius after displaced by thermal expansion goes constant on the curve for short time, before important positive displacement related to fission solid, volatile and gas swelling. Densification phenomenon is not shown by the step point of 500 h at just after thermal expansion. Slower swelling after about 13000 h appears which can be related to decrease in power rating of Fig 2 and 3.

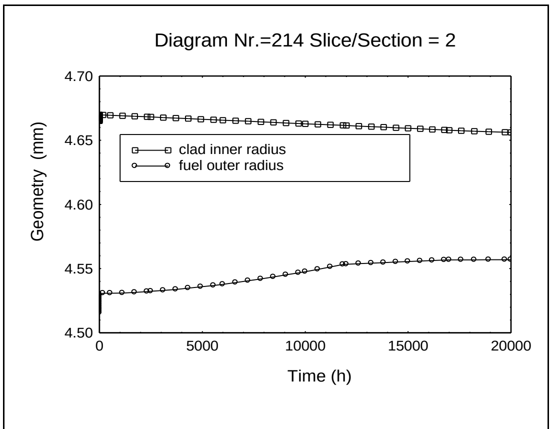

Figure 8. History of pellet and inner cladding radius

The result of displacement of pellet surface to the pellet radial behavior is presented by lower curve of Fig.8. The history of radius of inner radius of pellet is shown as upper curve of the same figure.

The combination of relative lower pellet temperature, low absolute burnup and relative wider pellet to cladding gap has resulting the decrease of gap width but never arrive to pellet to cladding mechanical interaction. It has happened in commercial CANDU and HWR pins [11], for the natural UO_2 prototype pin for PWR and it is irradiated with low power linear rating of 4 – 8 kW/m during 833,3 full days.

The author recommend to evaluate an irradiation test of about envelop power rating history of HWR natural uranium oxide fuel for this test fuel pin.

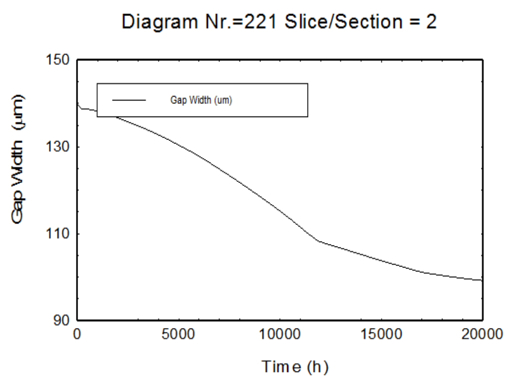

Fig. 9 Gap width change during irradiation of second axial position of highest flux

PCMI is pre-request for PCI failure, then the base irradiation need to be done at least until PCMI start or simplified by pellet – clad gap closed, to get efficiently power ramp test.

Fig.9 shows the continuous decreasing pellet to cladding gap during irradiation caused by pellet swelling and cladding creep down in ~ 550 K and higher pressure of coolant then internal pressure originated by fill gas plus released fission gasses. A deceleration can be seen after PCMI started when Dci equal to Dfi. The stress from swelling pellet tends to balance the stress from coolant. On surface contact of pellet and inner cladding surface does not homogenous because some crack tips on part of pellet surface caused a concentrated stress on contact to cladding surface.

F. Stress and strain of pellet and cladding

Different kind of cladding stresses caused by coolant tend to decrease during PCMI which starts at 1700 h or 508,3 d of irradiation. Pellet swelling and fission gas released balancing the stress resulted by coolant pressure. It is an important parameter which is related strongly to thermal, mechanical and chemical interaction. When power ramp happen after PCMI occur, potential gas released is high because higher stress of pellet during PCMI, and consequently a stronger oxidation to Zr cladding, giving higher potential to SCC (Stress corrosion cracking) phenomenon which is considered as the mechanism of cladding failure during power ramp. Accordingly, the effective power ramp test may be done after the onset PCMI. It is important parameter which is related strongly to thermal, mechanical interaction and chemical interaction.

The maximum fuel temperature is < 600 ^oC, while minimum gap attained 17% of its initial value. It can be considered that the available power inside PRTF capsule [12] is under required the test fuel pin. A recent flux measurement of several location of RSG of core 83 noted that at position on G7 at reactor power of 15 MW, show its flux was 10^{13} n/cm^2/s [13]. it is apparently the measured data of pin power at nearest position of 4 W with natural UO_2 pellet of 40 cm is need to be re-evaluate.

The Transuranus code applied to this modeling-simulation was developed by Lasmann the first author [14] and then further development by Committee European – Joint Research Cooperation hosted by the Institute

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of trans Uranium Element [15]. The commercial code has been verified and validated by different researcher and institution both by code to code benchmarking [16] by validation by analytical solution for simplified problem [17] and by NEA-OECD IFPE (Nuclear Energy Agency-OECD Irradiation Fuel Peformance Experiment) data base [18-20].

CONCLUSSION

Diameter change of PTBN fuel pin prototype-01 for PWR during irradiation is an important parameter for performance evaluation. Mild ramp of available power in PRTF has been chosen to be evaluated an it has been analyzed by a fuel code, validated by several nuclear regulator as licensing code. On start-up irradiation both pellet and cladding diameters increase significantly which related to thermal expansion as temperature increase bruftly. The absolute increase of pellet diameter is higher than the cladding as its thickness is > 15 times than cladding thickness. The increase of inner and outer cladding diameter on stat-up then followed by continous decrease. It is caused by higher pressure of coolant water than inner pressure. For pellet diameter, after bruft increase by thermal expansion it is followed by slow increase for about 5000 h then a slight acceleration until 1200 h where the mild power ramp cheased by contant power before slower power decreasing. At the end or irradiation, 62% of fuel initial 235U content (0.71%) has underwent fission. Remained 0.27% 235U is far under the content depeleted U by u enrichment process., but the gap of fuel pin has never attainned PCMI. The maximum fuel temperature during irradiation is < 600 ^{o}C, while minimum gap attained 17% of its initial value. It is considered that the available power inside PRTF capsule is under required the test fuel pin. With reference of measured power distribution of RSG-G.A.S. core 83 near PRTF, the author recommends to re-evaluate the available flux data inside PRTF capsule, then to determine if a second simulation is needed.

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