COMPARATIVE PREDICTION OF IRRADIATION TEST OF CNFT AND CISE PROTOTYPES OF CIRENE FUEL PINS, A PREDICTION BY TRANSURANUS M1V1J12 CODE

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ABSTRACT

COMPARATIVE PREDICTION OF IRRADIATION TEST OF CNFT AND CISE PROTOTYPES OF CIRENE FUEL PINS, A PREDICTION BY TRANSURANUS M1V1J12 CODE. A prototype of fuel pin design for HWR by CIRENE has been realized by Center for Nuclear Fuel Technology CNFT-BATAN. The prototype will be irradiated in PRTF Power Ramp Test (PRTF). The facility has been installed inside RSG-GA Siwabessy at Serpong. The present paper reports the preparation of experimentation and prediction of irradiation test. One previous PCI test report is found in, written by Lysell G and Valli G in 1973. The CNFT fuel irradiation test parameter is adapted to both PRTF and power loop design for RSG-GAS reactor in Serpong mainly the maxima of: rod length, neutrons flux, total power of rod, and power ramp rate. The CNFT CIRENE prototype design has been reported by Futichah et al 2007 and 2010. The AEC-India HWR fuel pin is of 19/22 fuel bundle design has also been evaluated as comparison. The first PCI test prediction has experiment comparison for Cise pin. The second prediction will be used for optimizing the design of ramp test for CNFT CIRENE fuel pin prototype.

Keywords: ramp power, irradiation test, HWR fuel pins.

ABSTRAK

***PERBANDINGAN PREDIKSI UJI IRADIASI PROTOTIPE PIN BAHAN BAKAR CIRENE CNFT DAN CISE MENGGUNAKAN KODE TRANSURANUS* M1V1J12.** *Satu prototipe pin elemen bakar PHWR telah disiapkan dengan menggunakan fasilitas elemen bakar eksperimental siap untuk Uji PRT di PRTF dalam RSG, bila diisi pin UO_2 alam menghasilkan LHR 12.9 kW/m. Aksial daya bentuk kosinus. Analisis pra-eksperimen telah dilakukan dengan dengan kode Transuranus, memilih daya dasar sebesar 9 kW/m, kenaikan daya setelah tercapai on-set PCMI, sampai daya maksimal 12,9 kW/m dengan temperature pendingin dan tekanan kondisi PHWR. Model sifat material adalah standar pelet PHWR dan standar kelongsong Zry-4. Diperoleh riwayat temperature pusat pellet irisan ke-3, dari 5 iris sama panjang, mencapai temperatur pellet tertinggi < 600 ^oC. Sementara untuk burnup, Gas fisi terbentuk dan terlepaskan ke plenum tertinggi < 1%. On-set PCMI akibat swelling pellet dan creep down kelongsong, terjadi pada saat iradiasi mencapai 17000 jam. Tegangan hook pada kelongsong mula-mula negative oleh tekanan air pendingin mencapai -90 MPa, setelah PCMI tegangan negative mengecil, mencapai +10 MPa.*

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Temperatur, gas fisi dan tegangan regangan kelongsong setelah PCMI dan kenaikan daya tidak menjadi ancaman integritas pelet. Kecilnya parameter mekanik, konsentrasi gas fisi pada kelongsong terkait dengan kecilnya daya. Untuk menyingkat waktu iradiasi deban dasar minimum 17000 jam disarankan alternative eksperimen dengan mendesain dan membuat pin simulasi kondisi PCMI: geometri gelembung gas fisi pada pellet dan konsentrasi.

***Kata kunci**: daya naik, pin bahan bakar, interaksi mekanik, stress-corrosion cracking.*

INTRODUCTION

CIRENE is the name of a line of heavy-water nuclear reactor boiling[1] of Italian design and implementation, and is an acronym of the words CISE 'Reactor Nebuckle' (means moist reactor), as it developed, at least for a time, from the Centre Information Studies and Experiments (CISE), first research center funded by various private companies, then checked by ENEL. In 1972, ANSALDO -in a joint venture with G.E.(General Electric)- completed the Fabricazioni Nucleari (Bosco Marengo) to manufacture the fuel elements for the future BWR's. The plant can produce 100 tons of fuel annually. It entered in operation in 1976 and has produced more than 500 tons of fuel for the Italian nuclear power stations and Leibstadt nuclear power station in Switzerland[2]. The study of supply chain CIRENE was abandoned as a result of the referendums of 1987, and the prototype, low power with only 130 MW of thermal power[3] and about 40 MWe[1] , in an advanced state of construction, virtually completed only the machine for loading and unloading of the fuel[2], a Latin remains unused and is not yet decided the fate. The choice of heavy water, which is a moderator of less effective water light, but with a lower coefficient of absorption parasite, thereby improving the neutron economy, allows the use of uranium so-called natural. This placed much less problems in Indonesia and other countries signatories of the Nuclear Non Proliferation Treaty and therefore unable in the possibility of self-enrichment of uranium. The reactor Swedish Marviken, a chain similar to CIRENE was completely built, but it was never started[1].

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[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 relate to nuclear electricity started on HWR (Heavy Water Reactor), which allow natural uranium use for fuel. D_2O production, 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 period, some laboratories and pilot scale facilities have been built at Serpong district: RSG-GAS a 30 MW MTR (Material Testing Reactor), MTR Fuel Element Production Facility (FEPI), Pilot Conversion Plant (PCP) for yellow cake or UF6 to UO_2/$U3O_8$. Experimental Fuel Element Installation (EFEI) for CIRENE HWR fuel rod. Radio Metalurgi 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 Installation, Nuclear Mecano-Electrical Installation has 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.

Although the CIRENE differ substantially from the Canadian project type pressurized HWR, its fuel technology can be utilized to mastering other HWR, when both manufacturing and design has been mastered.

Recently, an expansion of EFEI capability has performed for PWR pellets fabrication study and PWR fuel rodlet 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. Before irradiation testing its PWR pin prototype, it needs to test the manufactured CIRENE fuel rodlet..

At the beginning of the irradiation, there is a distinct circumferential gap between the external face of the fuel pellets and the inner bore of the cladding tube. On raising power, this gap is reduced through thermal expansion of the pellet. This gap becomes less distinct as the ceramic pellets crack under the influence of the radial temperature gradient. The distance between the pellet surface and the cladding inner bore is further reduced as the pellet fragments relocation radial outwards. The free volume is now shared between the residual gap and the space between pellet fragments.

As irradiation proceeds, the cladding diameter reduces by creep driven by the compressive hoop stress induced by the difference between coolant pressure and the internal pressure of the fuel rod. Eventually, the cladding creeps down onto the fuel finally eliminating the fuel-to-clad gap and moving the pellet fragments closer together. As a rough guide, this occurs during the second cycle of irradiation in the burn-up range 10-20 MWd/kg. Prior to gap closure, the cladding is under a compressive hoop stress. Once the gap is closed and the cladding and pellet fragments are in contact, the hoop stress gradually decreases as the pellet fragments resist the reduction in cladding diameter. The hoop stress in the cladding eventually becomes positive when all the internal free space is exhausted and the pellet fragments are in intimate contact. The irradiation time in base-load to attain this on-set of mechanical contact between pellet and cladding (PCMI Pellet to Cladding Mechanical Interaction) is minimum time of pre-ramp irradiation required before experimenting ramp test.

At any stage irradiation after this on-set time for PCMI, an increase in power can cause the pellets to expand through both thermal expansion and fission product swelling to induce a positive hoop stress in the cladding. The immediate reaction of the cladding is to expand outward by elastic deformation and subsequently by plastic strain and creep, thus reducing the interfacial stress. This interaction between the fuel pellet and the cladding is termed pellet-clad mechanical interaction (PCMI). If, however, the clad hoop stress is sufficiently large, and the pellet temperature is high enough to release corrosive fission products, internal cracks may be initiated at the inner bore by stress corrosion cracking (SCC) which grows under the influence of the maintained hoop stress such that the cladding fails.

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.

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PREDICTION METHOD

The testing system is shown in Appendix-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. The detailed data of Appendix-Fig.1(a) fuel pin test is presented in Appendix-Fig.2 a shop drawing of the pin irradiation test. It allows one calculate some data not found such as plenum volume.

Table 1. Fuel pin design for CIRENE[3]

Fuel Pin Parameter	Value	Unit
Rod length	244	mm
Rod outside diameter	13,2	mm
Pellet Length	12	mm
Pelet outside diameter	12	mm
Pellet density	9,55	g/cm^3
Pellet Material	nat UO_2	
Clad Material	Zry-2	
Clad thickness	1	mm
Number of Pellet per pin	20	

Operational Parameter	Value	Unit
Average linear fuel rating	5 - 7	kW/m
Peak linear fuel rating	8 - 9	kW/m

Table 2. Two data HWR irradiation testing[4, 5]

Parameter*	A-1*	TT-WW**
Outside diameter of clad, (mm)	11,9	13,85
Cladding wall thickness, (mm)	0,55	0,8
Inner Diameter of clad, (mm)	10,8	12,25
Gap , (mm)	0,09	0,905
Fuel pellet diameter (mm)	10,62	10,44
% Enrichment at BOL	0,7	3,35
LHR (kW/m)	N/A	12
BurnUp (pre-ramp MWd/kgU)	N/A	59

*Atucha-1 PHWR fuel
** Instrumented high burnup LWR fuel, T. Tverberg, W. Wiesenack,
N/A not available

Cooling water and pressure for irradiation test App-Fig.1(c)[6] of the PRTF is summarized in Appendix-Tab.1 [10]: 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 oC and 150 Atm.

Detailed geometry of fuel pin test is shown in Appendix-Fig 2. It 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 [4] represents manufacturing data of test fuel. The fuel properties have been chosen from standard option of Transuranus version M1V1J12 code[11]. Table 2 shows the PRTF and other important data.

The irradiation parameter has been chosen by comparing 3 pin data from Tab.1 and Tab.2, and corresponding 2 irradiation data of Tab 2, and regarding both the PCI threshold of HWR fuel and the maximum allowable PRTF thermal power cooling. These data and related thermal-mechanical material properties have been inputed to Transuranus code to obtain its behavior, here thermal and mechanical, during power ramp test.

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.

RESULTS AND DISCUSSION

Some cited data of Tab.1 needs correction, because if both outer fuel diameter and outer clad diameter are right then the gap must be negative (gap = - 400 um). When the gap is zero, then the cladding thickness must below 0.60 mm. So the author supposed the clad thickness is result of error calculation between diameter and radius then the analysis assumed that the cladding thickness is 0,50 mm which is a half of cited data, then the nominal gap size is 100 um. This is slightly wider than 90 um of Atucha-1 PHWR, but much tighter than 905 um of TT-WW fuel in Tab.2

When equalized to Atucha-1 fuel pin (A-1) data of fresh fuel gap width (90 um), then the clad thickness will be 510 um still below those Atucha-1. It has been considered the need to perform sensitivity study of fuel diameter which alter the inner diameter of cladding and the thickness of cladding (both or either). Reducing pellet diameter is simpler than increasing/ augmenting the cladding diameter.

Temperature and linear heat generation rate (LHR)

The calculation of temperatures in a fuel rod is one of the primary goals of fuel element modeling. The accuracy of these calculations strongly influences temperature-dependent physical phenomena such as fission, gas diffusion and release, restructuring creep, thermal expansion, etc.

Input 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 id lower than slice-4 the highest value. Fig.1 shows history of radial-average of 5 equal lengths of slices of the test rod.

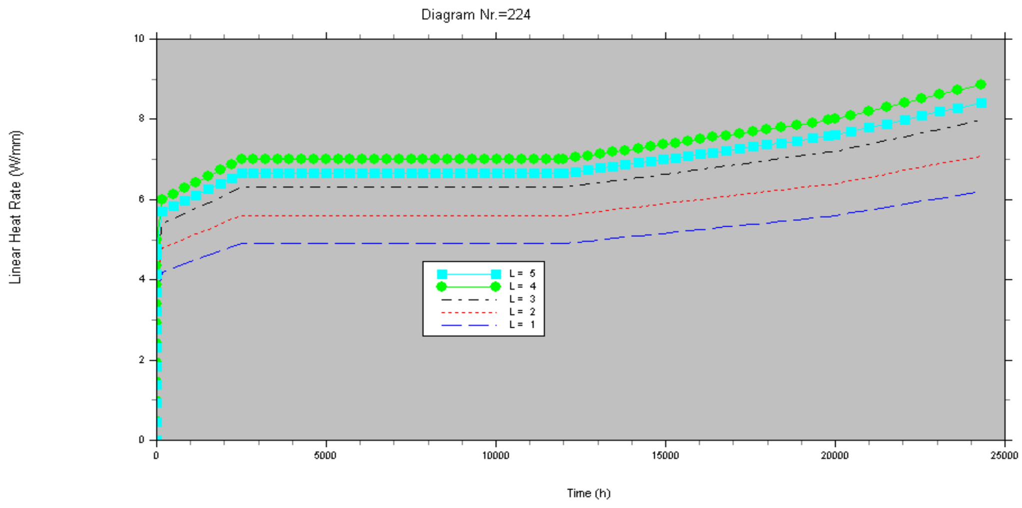

Fig.1 History averaged slice LHR

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The temperature of slice-4 at different irradiation time of hundred to 20000 h is plotted in Fig.2. Among 4 radial distributions 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 about 20 K or 5%, as pellet underwent densification then power density increased proportionally to its shrinkage. At pellet outer surface a slight temperature increased since pellet shrinkage mean pellet to clad gap increase conductance decreased while coolant temperatures invariable. From 2000 h to 12000 h the pellet temperature along pellet radius dropped nearly homogeneous, with only slight higher at pellet center. The temperature drop is connected pellet swelling and clad creep-down means gap closing and temperature jump between pellet and clad decreased. Swelling surpassed densification real LHR decrease related to its pellet density, then temperature drop slightly larger toward pellet center. From both Fig.5 and Fig.6, the on-set of PCMI can be read approximately at about 17000 h of irradiation. At 20000 h the interior part pellet hotter than the previous curve but the exterior part of pellet colder than previously. Higher temperature at pellet center is related to the decreasing pellet thermal-conductivity due to retention of fission gas as bubbles in pellet and a slight increase of LHGR. Colder temperature at exterior of pellet is related to higher contact thermal conductance because pellet swelling increase mechanical contact between pellet and clad causing higher micro contact surface area by deformation of surface roughness of both surfaces.

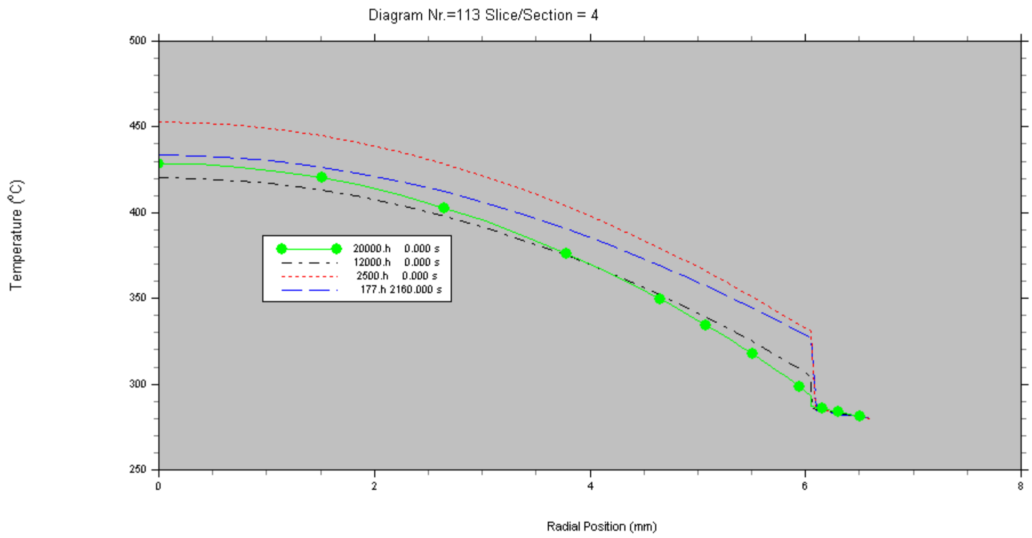

Fig. 2. Evolution in time of radial distribution of pellet temperature.

Burnup distribution in time and in space

Fig.3 shows the average burnup in each pin slice with a slight acceleration of burnup accumulation along time. It can be related to the slowly increase of LH(G)R (Linear Heat (Generation) Rate) with time. The curve slopes or burnup rate are proportional to its LHGR. The PCI test parameter and the CISE CIRENE fuel design are obtained in "Overpower ramp tests on CIRENE prototype" reported by Lysell G and Valli G, 1973, [8,9] is considered to severe to be used as comparison to present PRTF, or to be experimented with existing Batan facility. The cladding wall thickness CISE test rod is 10% thicker than BATAN rod, fresh or initial gap size of CISE rod is 10% lower than BATAN one.

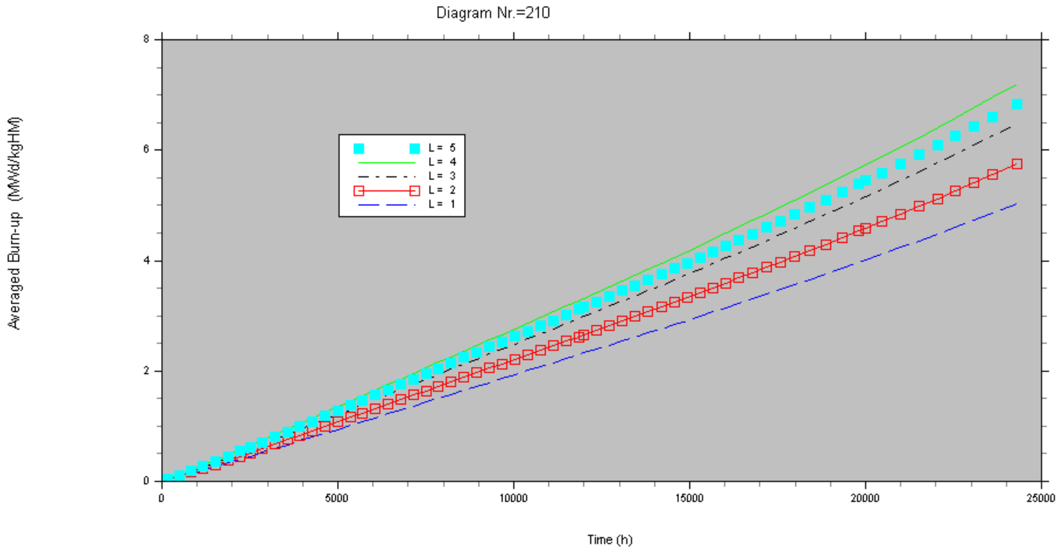

Fig.3 History of slice averaged burnup

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. When pellet temperature exceeds release threshold some fission gas released to open pore, gap, and plenum. Since the pellet temperature under release threshold, the released gasses are very small compared to total produced, as shown in Fig.4. Retained bubbles gasses swell the pellet. At 20000 h fission gases produced mostly at 10 % exterior part of pellet, at surface accumulated fission gas produced about 2,5 average. This radial distribution curve is proportional to power distribution.

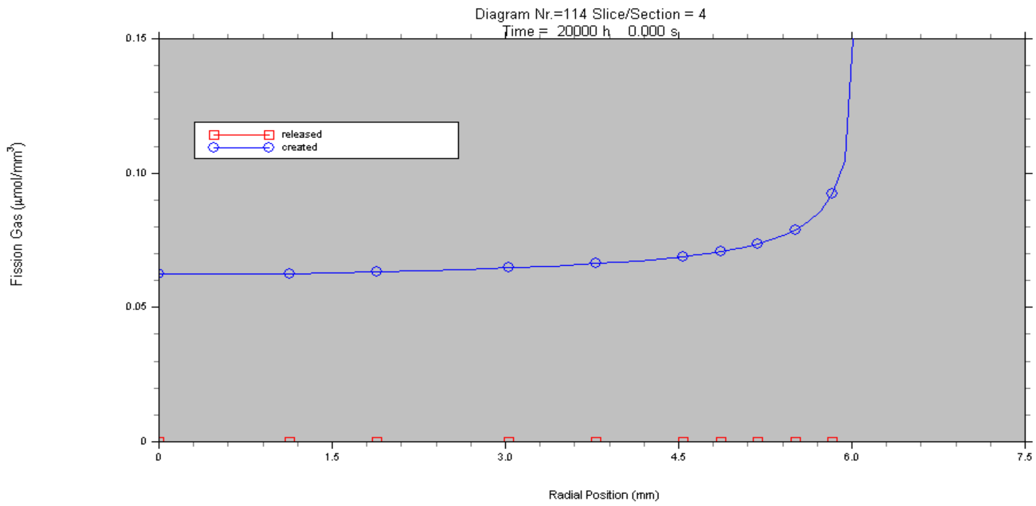

Fig. 4 Radial distribution of produced and released fission gas along position at 20000 h

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Geometry change

Geometry change as shown in Fig.5 and 6 are interaction of thermal-mechanical and irradiation swelling, influenced by thermal, mechanical, chemical and its combination. It imply directly to fuel reliability. Gap with change caused by fuel swelling, cracking, clad creep etc. PCMI is pre-request for PCI failure, then the base irradiation need to be done at least until it start or simplified by the gap closed, to get efficiently power ramp test.

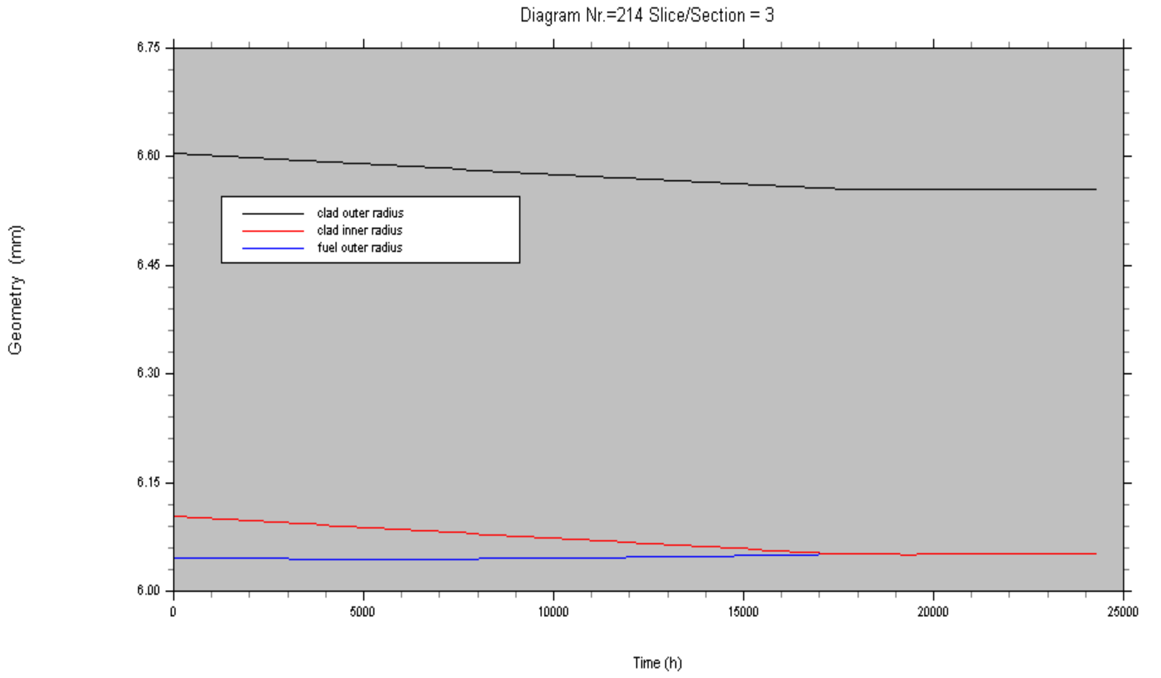

Fig.5. Diameter change of outer, inner clad and pellet during irradiation.

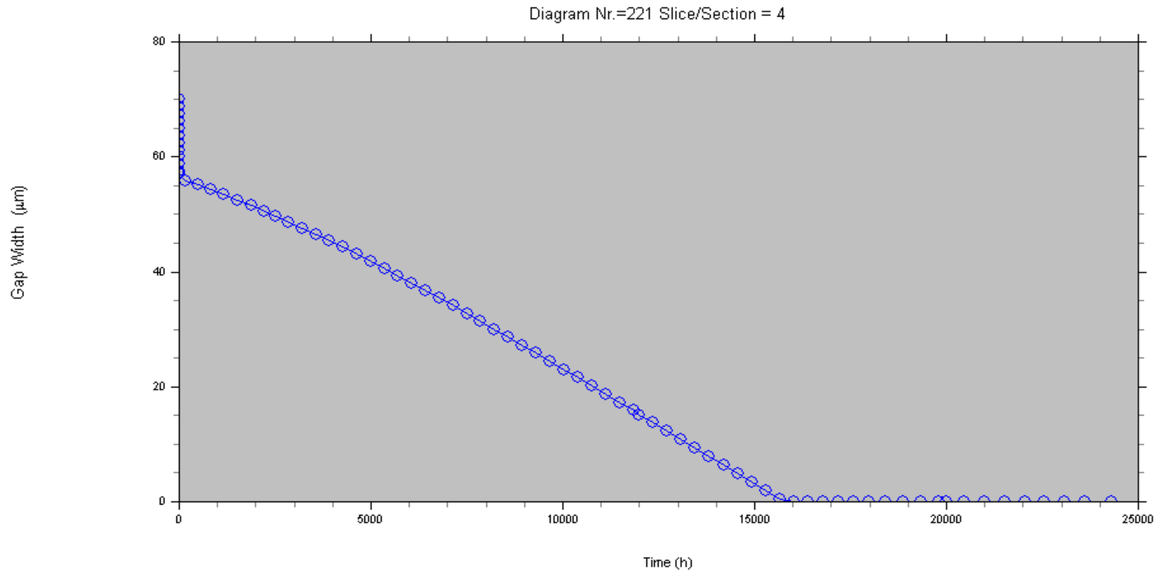

Fig.6. Pellet – Clad gap width corresponding the difference of Dci – Dfo in Fig.5

Fig.6 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 (inner cladding diameter) equal to Dfo (outer fuel diameter). 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.

Stress and strain of pellet and cladding

Different kind of cladding stresses caused by coolant tend decreasing during PCMI and starts 1700 h or 508,3 d of irradiation. Pellet swelling and fission gas released balancing the stress resulted by coolant pressure. It is important parameter related strongly to thermal, mechanical interaction and chemical interaction. When power ramp happen after PCMI occur, gas released increases significantly, which containing strong potential oxidation to Zr and at the presence of PCI, give higher potential to SCC Stress corrosion cracking phenomena which is considered as the mechanism of cladding failure during power ramp. Then the effective power ramp test may be done after the of PCMI event. It is important parameter which is related strongly to thermal, mechanical interaction and chemical interaction.

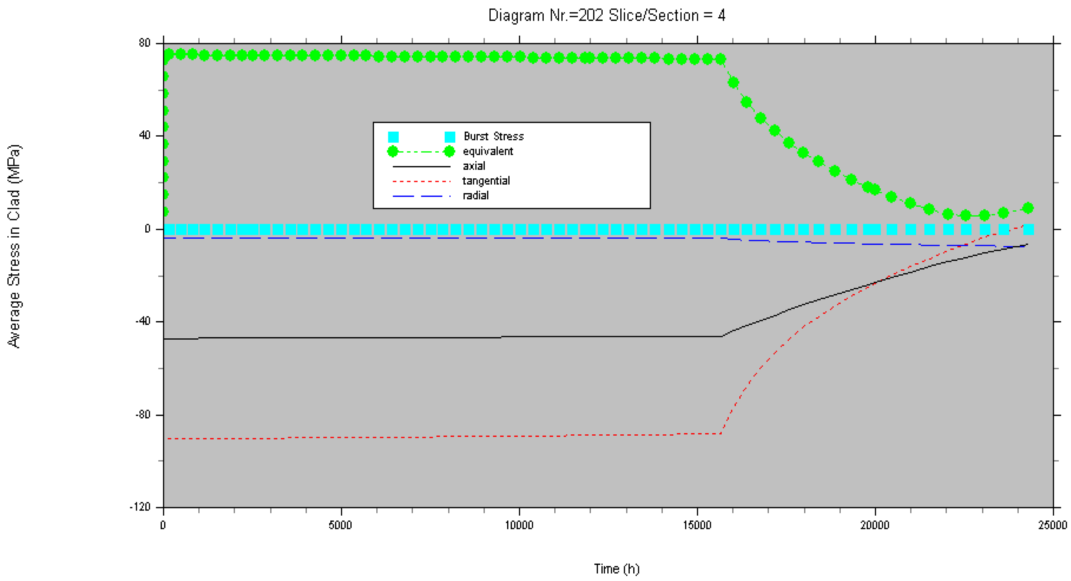

Fig.7. Evolution of different Stresses of clad, volumetric average.

CONCLUSION

The correction of pin geometry data has been done for wall thickness. An attempt to evaluate a modification of clad wall thickness from 1 mm to 0.5 mm and initial pellet-to clad gap of 0.1 mm with model of material properties, parameters choice and a scenario of irradiation test with LHR history which a slight increase from ~ 13000 h by using Transuranus code.

The important results, fuel pin behavior which strongly related to PCI have been presented and discussed. In relation to power ramp test for the fuel pin analyzed with the scenario and model chosen, it is found that the PCMI occur at about 17000 h or 708.333 d. It is about the end of commercial irradiation of PHWR fuel with

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only natural uranium.

Consequently this first analysis recommends an irradiation base at least 700 days or burnup about 7.600 MWd/kg U.

The PCI test parameter and the CISE CIRENE fuel design are obtained in "Overpower ramp tests on CIRENE prototype" reported by Lysell G and Valli G, 1973, is considered to severe to be used as comparison to present PRTF, or to be experimented with existing Batan facility.

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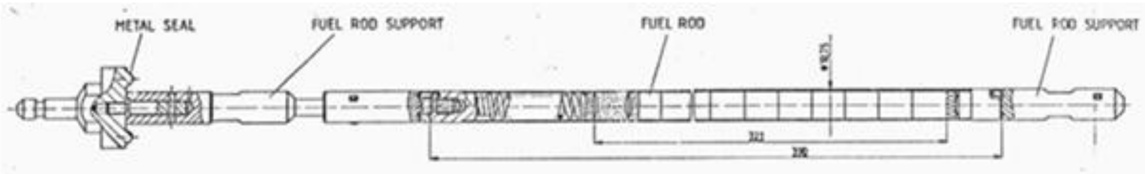
APPENDIX

Appendix-Tab.1. Technical specifications of PRTF cooling system [10]

Primary cooling system	Secondary cooling system
1. Type: Water Cooling 2. Pressure: 160 bar 3. Flow: 3.7 liters / hour ~ 1 cm3 / s	1. Type: Cooling water. 2. Pressure: 2,5 bar 3. Flow: 750 liter / hour



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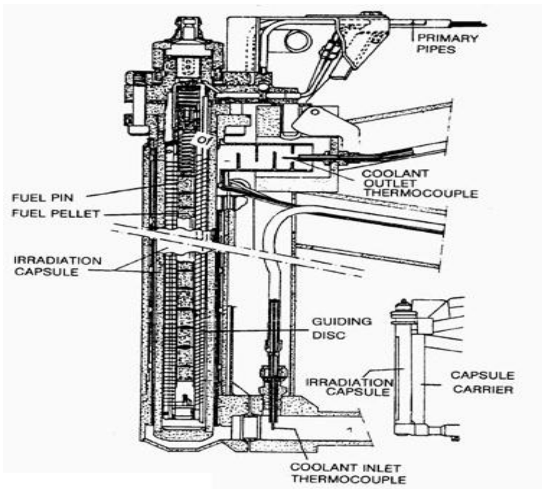

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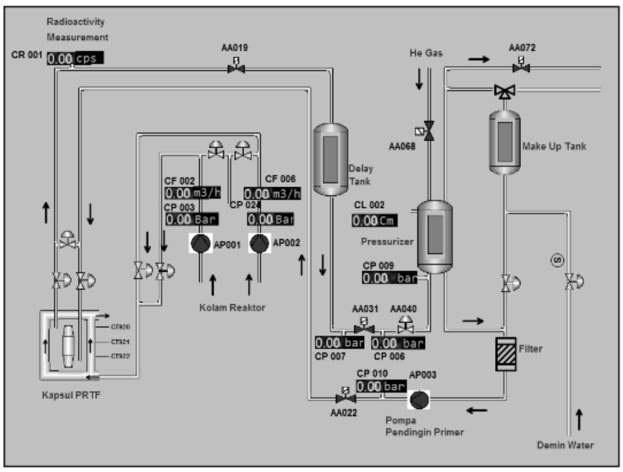

(c)

Appendix-Fig. 1. Technical drawing of (a) fuel pin connected to support and metal seal, (b) Technical drawing of Power Ramp Test Facility capsule containing fuel pinand (c) Schematic drawing of installed PRTF in RSG-GAS 30MW[10].

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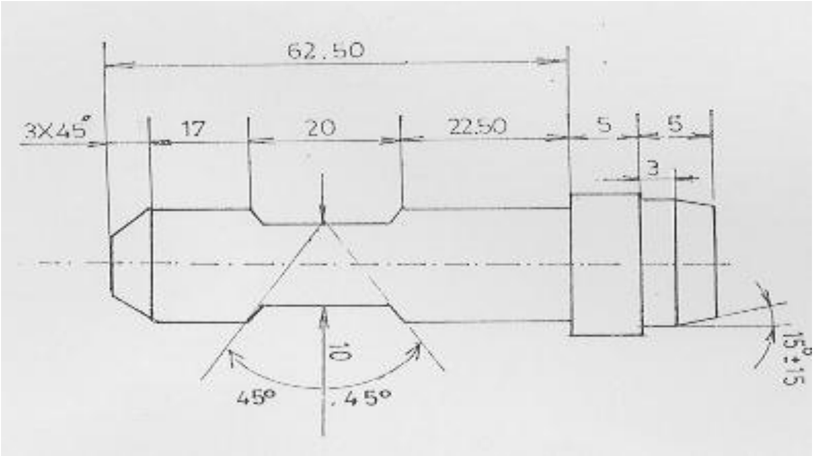

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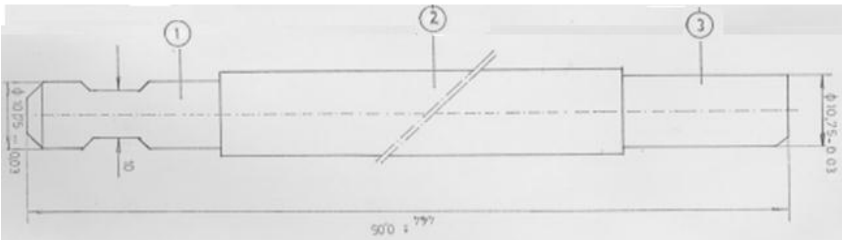

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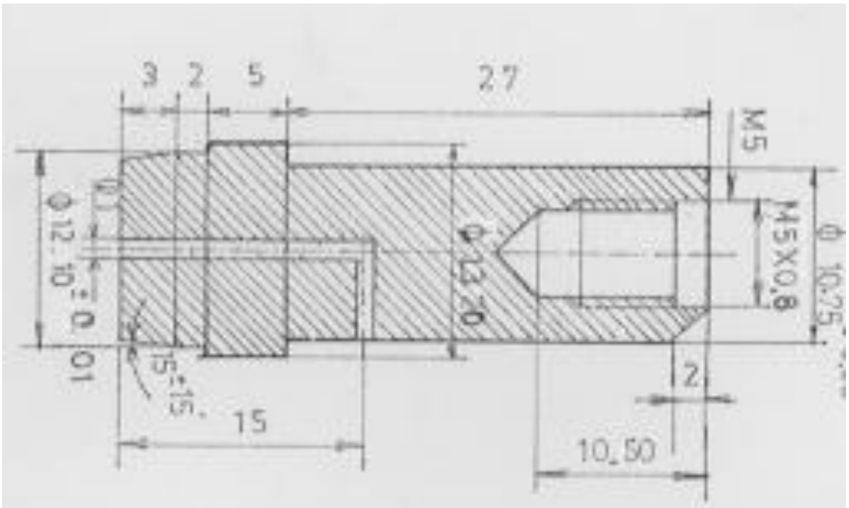

(c)

Appendix-Fig.2. Detailed geometry of fuel pin, (a) end plug support-1
(b) whole pin, (c) end plug support-2 [6]