IRRADIATION EXPERIENCE OF IPEN FUEL AT IEA-R1 RESEARCH REACTOR

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ABSTRACT

IPEN/CNEN-SP produces, for its IEA-R1 Research Reactor, MTR fuel assemblies based on U_3O_8-Al dispersion fuel type. Since 1985 a qualification program on these fuel assemblies has been performed. Average ^{235}U burnup of 30 % and peak burnup of 50% was already achieved by these fuel assemblies. This paper presents some results acquire, by these fuel assemblies, under irradiation at IEA-R1 Research Reactor.

INTRODUCTION

The IEA-R1 Reactor of IPEN/CNEN-SP is a pool type reactor in operation since 1957. It operated at 2 MW up to 1997 when its power was increased to 5 MW. Today the reactor operates with a core arrangement of 25 LEU fuel assemblies which are designed and produced by IPEN. The program for autonomous serial fuel assemblies production started at IPEN in the 80's, motivated by the political constraint for buying these fuels abroad and the necessity to maintain the reactor in operation. Due to IPEN previous experience in fuel development, LEU U_3O_8-Al dispersion fuel was chosen to be used. Previous papers of the authors presented the history and strategy of LEU fuel development and utilization in IPEN.[1,2] Up to 1996 IPEN produced fuel assemblies with 1.9 gU/cm^3 in the fuel plate meat and since 1996 has been producing fuels with 2.3 gU/cm^3. This paper presents some relevant aspects observed in the irradiation of these fuels.

FUEL QUALIFICATION PROGRAM

Brazil has no hot cell laboratories for testing irradiated fuels or irradiated materials. Irradiation and post-irradiation analysis of miniplates and full size plates at foreign countries would be very expensive and there were no budget for this. As IPEN fuel specification was conservative for a dispersion fuel and as the power of reactor is low it was decided to take the risk of testing and evaluating , in pile, the fuel performance along the reactor operation. A sequence of testing events and procedures was set to qualify the fuel.[3] It started with irradiation of some miniplates at the border of the reactor core just to identify any abnormal event. A fuel element assembly with only two fuel plates (the external plates) and 16 aluminum plates was placed in the core to start fuel qualification (July 1985). After this, a fuel element assembly with 10 fuel plates and 8 aluminum plates was also placed in the core (November 1985). These two fuel element assemblies were identified as the precursor fuels and a periodic monitoring and evaluation was done upon them. After good experience with the precursor fuels it was decided to start loading standard fuel element assemblies in the reactor core (September 1988). The criteria adopted was that each IPEN fuel element assembly had to start irradiation at peripheral positions of the core, with lower power densities, up to 4 % burnup (almost one year of irradiation) and then it could go to higher power density positions in the core. It was decided that the precursor fuels had to stay in the core up to the time that a complete fuel assembly could equal their burnup. The last precursor fuel assembly was taken out of the reactor core with 18% burnup (average).

The program of evaluation has been consisted of the following items:

i) *monitoring:*

reactor power; time of operation; inlet and outlet water temperature in the core; water pH; water conductivity; chloride content in water; radiochemistry analysis of reactor water.

ii) *calculation*

neutron flux calculation at the position of each fuel assembly; burnup calculation; temperature distribution for each fuel assembly.

ii) *inspection:*

periodically underwater visual inspection ; eventual sipping test of fuel assembly, gamma scanning for burnup determination and profile, control rod drop profile inside control fuel assembly during scram.

Periodically reports are done and sent to regulatory authorities showing the results of this qualification program.[4]

Neutronic Calculations. Neutronic calculations are done in order to determine, for each fuel assembly, the average flux (thermal and fast) and the average burnup at each core configuration and power history. Table 1 presents the actual burnup history for each fuel assembly. Figures 1 and 2 show the evolution of average thermal flux and burnup for the first fuel assembly (IEA-130).

Thermohydraulic Calculation. The reactor water temperatures at core inlet and outlet are measured during the reactor operation. Based on these temperatures thermohydraulic analysis is performed, for each fuel assembly, to determining temperature profile at each core configuration and power history. The calculated values are based on nominal specifications and also considering the tolerances and uncertainties to obtain the maximum possible values of temperature. Table 2 shows typical values of temperature for fuel assemblies at IEA-R1 with 25 fuel assemblies in core.

Reactor Water Radiochemistry Analysis. Samples of the reactor pool water are weekly collected for radiochemical analysis by gamma spectrometry. Nuclides determination as ^{131}I, ^{133}I, ^{137}Cs, ^{140}Ba, and ^{239}Np can give indication of failure in fuel assemblies. According to the qualification program, if any increase on the historical values of the nuclides concentration in reactor water is observed, sipping tests and visual inspections on all core fuels assemblies have to be done in order to identify the origin of this increase. The reactor pool water has a typical background activity level. More than one hundred irradiated fuels assemblies are storage at the reactor pool and some of them are more than 40 years inside the pool. Some of these old fuels present corrosion pits with ^{137}Cs liberation to water.[5] Up to now no increase on nuclides concentration at the reactor water has been noticed during irradiation using IPEN fuel assemblies at IEA-R1 reactor. Figure 3 shows an example of some radioactive nuclides concentration history for the reactor pool water

Table 1. Irradiation Data (up to July 1998)

Fuel Assembly	U density (gU/cm^3)	In core loading date (dd/mm/yy)	In core discharging date (dd/mm/yy)	Average Burnup (% ^{235}U)
IEA-130	1.9	05/09/1988	02/07/1997	36.1
IEA-131	1.9	16/02/1989	02/07/1997	31.5
IEA-132	1.9	20/02/1990	02/07/1997	27.4
IEA-133	1.9	11/06/1990	02/07/1997	29.7
IEA-134	1.9	21/09/1990	19/12/1997	27.2
IEA-135	1.9	21/09/1990	02/07/1997	30.9
IEA-136	1.9	21/09/1990	19/12/1997	25.4
IEA-137	1.9	21/09/1990	-	28.7
IEA-138*	1.9	25/11/1991	02/03/1998	29.7
IEA-139*	1.9	20/07/1992	02/03/1998	30.3
IEA-140	1.9	01/10/1993	-	24.8
IEA-141	1.9	01/10/1993	-	23.4
IEA-143	1.9	13/12/1994	-	21.4
IEA-144	1.9	16/10/1995	-	17.2
IEA-145	1.9	16/10/1995	-	17.0
IEA-146*	1.9	16/09/1994	-	24.5
IEA-147*	1.9	23/03/1995	-	24.6
IEA-148	1.9	16/10/1995	-	17.1
IEA-149	1.9	10/06/1996	-	15.4
IEA-150	1.9	10/06/1996	-	15.3
IEA-151	1.9	15/11/1996	-	13.4
IEA-152	1.9	12/05/1997	-	7.7
IEA-153	2.3	15/11/1996	-	11.5
IEA-154	2.3	12/05/1997	-	8.5
IEA-155	2.3	08/09/1997	-	4.7
IEA-156	2.3	08/09/1997	-	6.3
IEA-157	2.3	08/09/1997	-	6.5
IEA-158	2.3	08/09/1997	-	6.4
IEA-159	2.3	08/09/1997	-	6.7
IEA-160	2.3	08/09/1997	-	4.6
IEA-161	2.3	08/01/1998	-	5.3
IEA-162	2.3	08/01/1998	-	4.0
IEA-166*	2.3	02/03/1998	-	4.4
IEA-167*	2.3	02/03/1998	-	4.4

*. Control Fuel Assemblies

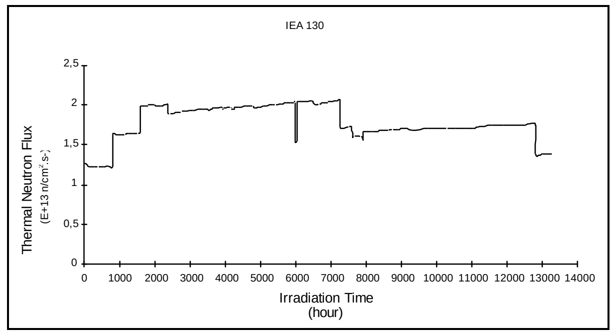

Figure 1. Average Thermal Flux for IEA-130.

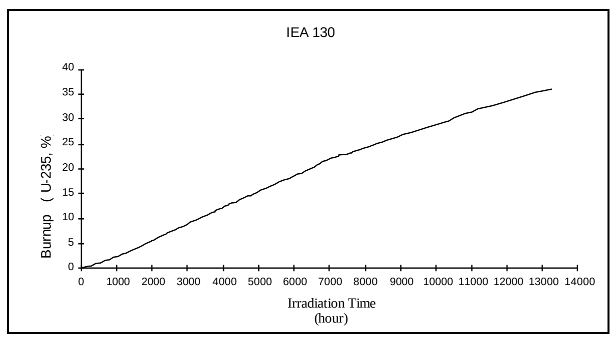

Figure 2. Average Burnup for IEA-130.

Table 2. F.A Temperature Typical Values

	2 MW	5 MW
Cladding Maximum Temperature (nominal)	50 °C	75 °C
Fuel Maximum Temperature (nominal)	52 °C	78 °C
Cladding Maximum Temperature (with uncertainties)	73 °C	94 °C

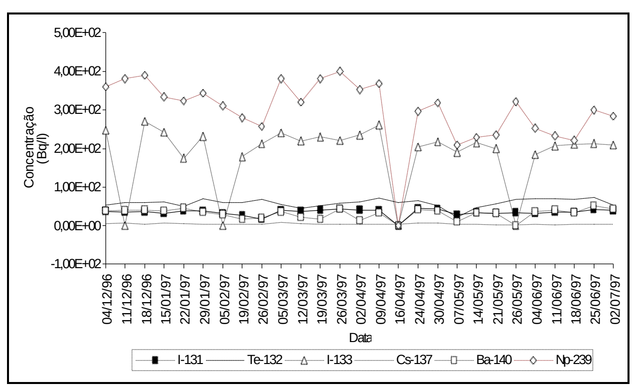

Figure 3. Water Radioactive Nuclides Activity

Water Chemical Analysis. The chemical characteristic of the reactor pool water is important for corrosion rate determination of fuel cladding. The reactor water technical specifications are suitable to lower the general corrosion of aluminum used as cladding of the fuel plates. These specifications are: 5.5<pH<6.5; conductivity < 2 $\mu S/cm^2$; and chlorides < 0.2 ppm. The values observed during all operation time were within the technical specifications.

Visual Inspection. The fuel assemblies with highest burnup are usually inspected by visual means to verify their general condition. Mechanical damage, gross deflections or blisters , and corrosion of the outer plates can be detected by this technique. The fuel assemblies to be inspected are taken out of the core and hung, by a tool, at 2 meters deep from the pool water surface. At the beginning of the qualification program visual inspection was performed with the help of a lens. Now it is used an underwater radiation resistant TV camera for the inspection. Some relevant points observed in these inspections are listed below.

i) *H_2 liberation from the interface between the fuel plate and support plate.* Due to the galvanic pair between the aluminum of the fuel plate cladding (Al 1060) and the aluminum of the lateral support plate (Al 6262 T6), and due to the stagnant water channel in this region of mechanical fixing, there's a significant liberation of H_2 gas as a consequence of the local corrosion of the Al 1060 when the fresh fuel is put inside water.[6] This was first observed at fuel assembly IEA-130 after one day of irradiation at the reactor. A procedure was established to leave the fresh fuel for a period of one week hung inside the pool water before to load it in the core. This allows enough time for the gross amount of H_2 to be released. Figure 4 shows the analysis of H_2 liberation for the fuel IEA-131 before loading in the reactor core.

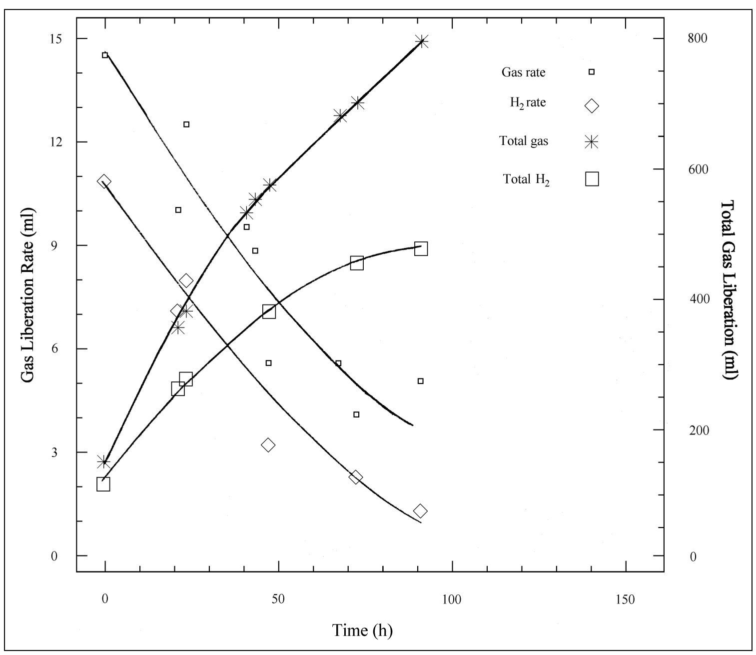

Figure 4 – Gas Liberation from Fuel Assembly

ii) *Shades of the oxide layer at the fuel plate.* The fuel plates show, at their surface, shades that goes from the dark gray to almost white. These shades are associated with the thickness of the oxide layer in the surface of the plate and related to the neutron flux, surface temperature of the plate and aluminum composition. These shades are more intense at the middle of the plate, where is located the maximum surface temperature, and change their intensity along irradiation from gray to white and then reverse from white to dark. These shade turned more visible when the reactor power was increased from 2 to 5 MW showing the dependency with the neutron flux and temperature. These shades can be seen at the fuel plate surface (cladding of Al 1060) but can not be seen at the lateral support plate (Al 6262). Figure 5 shows an example of these shades in a fuel assembly after the reactor started operation at 5 MW. For reactor operation at 2 MW and for the fuel assembly IEA-130 that was taken out of the reactor, these oxide layer were adherent to the plate surface and could no be removed by the soft action of a brush (inside the reactor pool), and did not imposed any limitation for this fuel operation at the reactor.

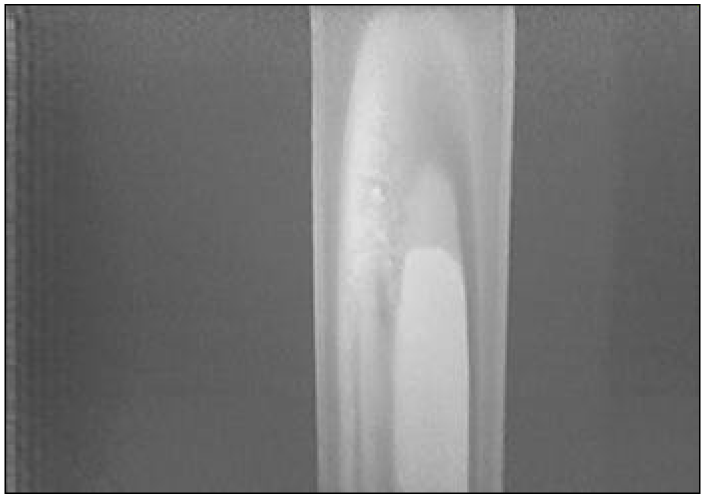

Figure 5 – Shades at Fuel Assembly

iii) *Corrosion pits*. The spent fuel assemblies are stored in stainless steel racks at the reactor pool. Due to the galvanic pair between the stainless steel and the aluminum of the fuel plate cladding, corrosion pits appear at the fuel plates. This was observed in the old fuel assemblies that are stored in pool for many years but also happened to a fresh non irradiated fuel. The fuel assembly number IEA-142 was put inside pool water and stayed for 56 days inside the storage rack. When this fuel assembly was catch to be load in the core it was noticed, visually, that there were many corrosion pits at the external plates in the region where the plate was in contact with the stainless steel of the storage rack. The pits had average diameter of 400 to 500 μmm and 150 μmm of depth. Because of this depth (the cladding has nominal thickness of 380 μmm and minimum of 250 μmm) it was decided not to use this assembly in the reactor core. It was then disassembled. It was observed that only the external plate of the assembly had pits and these pits were just in the outside face that was in contact with the stainless steel. All other fuel plates were in perfect conditions. The internal fuel plates were used to produce a new fuel assembly. This corrosion is not a fuel assembly design problem but a storage rack design problem. Today the spent fuel assemblies that come out of the core are stored in aluminum racks.

iv) *Fuel assembly deformation*. There were no visible deformation at the fuel assemblies as a consequence of irradiation in the reactor core. The visual inspection of water channels between fuel plates did not indicate any visible (gross) dimensional change (no measurement using equipment were done, all analysis were qualitative). Small scratches were noticed at the external surfaces but they were originated by the fuel assembly handling.

Control Rod Drop Test. Control fuel assembly is the one in which the control rod is inserted. This control fuel assembly has less fuel plates than the standard fuel assembly (12 compared to 18) and has the guides for the control blade (fork type) of the control rod, and has also the dashpot system at the top of the assembly. Control rod drop test permits to analyze the time of control element insertion, during scram action, and also the profile of this insertion. Curves of displacement, velocity and acceleration versus time permit to analyze the performance of this safety system, but also permit to have an evidence of the good performance of the fuel assembly under irradiation. Any distortion or excessive deformation of the control fuel assembly would modify the control rod insertion. Six control fuel assemblies were already fabricated by IPEN, and two of them have already been taken out of the core with 30% average burnup. No significant changes have been noticed up to now in these insertion profile. Figures 6, 7 and 8 show an example of the curves obtained by the tests performed at IEA-R1 reactor.

Sipping Test. Although the reactor pool water radiochemical analysis did not indicate any significant change at the historical values, sipping test was performed in those fuels that were discharged from the core. Sipping test is performed in the storage pool with a device that isolates the water of the sipping test from the pool water. Up to now no evidence of ^{137}Cs leakage was detected from IPEN fuel assemblies.

Burnup Determination by Gamma Spectrometry. Neutronic calculation gives an average value for neutron flux and burnup. Using gamma scanning techniques one can measure the gamma activity and to obtain the real average burnup and its profile in the fuel assembly (within the precision of the experimental method). A gamma scanning device at the reactor pool and a procedure of analysis were developed to determine this burnup profile of fuel assemblies.[7] Table 3 shows the results obtained for fuel assemblies with highest burnup. It's seen that the measured average burnup is close to the calculated one, and that the local average burnup in the fuel assembly reached 50%.

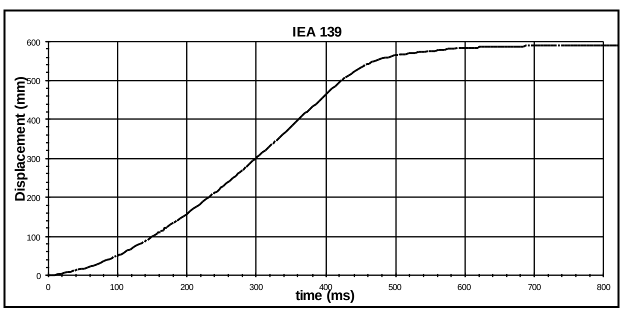

Figure 6 – Control Rod Drop Displacement

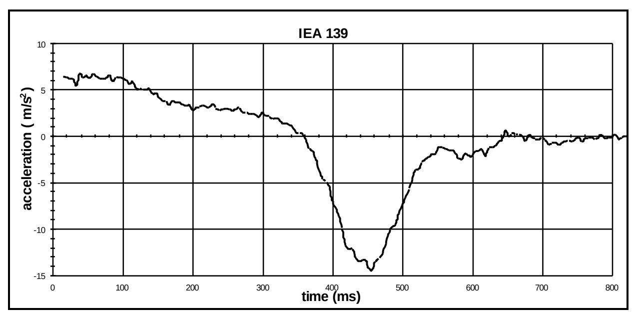

Figure 8 - Control Rod Drop Acceleration

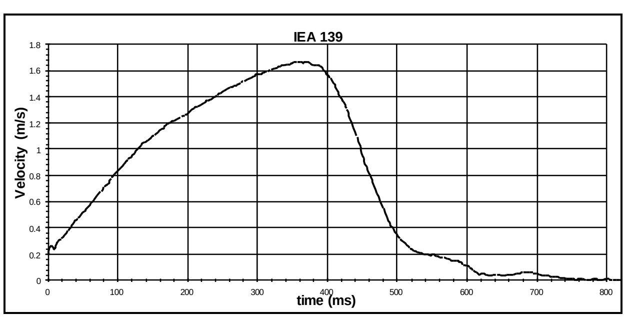

Figure 7 - Control Rod Drop Velocity

Table 3. Burnup Determination Results.

Fuel Assembly	Calculated Average Burnup (% of ^{235}U)	Measured Average Burnup (% of ^{235}U)	Measured Average Local Burnup (% of ^{235}U)
IEA-130	36.1	37 ± 5	53 ± 5
IEA-131	31.5	29 ± 5	39 ± 5
IEA-132	27.4	31 ± 5	36 ± 5

CONCLUSION

In September 1988 the first complete fuel assembly of IPEN production was put in the IEA-R1 Reactor. It irradiated in the reactor core, with the reactor at 2 MW of power, up to July 1997. A qualification program was established for verifying its irradiation. Average burnup of 30% and peak burnup of 50% was achieved by this fuel assembly demonstrating good performance under these conditions.

REFERENCES

[1] Perrotta,J.A; **Estratégia de Utilização de Combustíveis Tipo MTR de Alto Teor de Urânio para Operação do Reator IEA-R1 a 5 MW**; VI CGEN, Rio de Janeiro, Brazil, October 1996.

[2] Perrotta, J.A;Lainetti,P.E.O; **Program of Converting IEA-R1 Brazilian Research Reactor from HEU to LEU**; 19th RERTR International Meeting, Seoul, Korea, 5-10 October 1996.

[3] Perrotta,J.A; Mattos,J.R.L; Hayashi,C.S; Silva,A.T; Riella,H.G; **Qualificação sob Irradiação dos Combustíveis tipo MTR de U3O8-Al de Fabricação IPEN/CNEN-SP no Reator IEA-R1**; III CGEN, Rio de Janeiro, Brazil, July 1990.

[4] **Acompanhamento da Irradiação dos ECs Nacionais no Reator IEA-R1**; Internal Reports Divisão de Engenharia do Núcleo - Diretoria de Reatores - IPEN/CNEN-SP.

[5] Perrotta,J.A; Terremoto,L.A.A; Zeituni,C.A; **Experience on Wet Storage Spent Fuel Sipping at IEA-R1 Brazilian Research Reactor**; IAEA Technical Committee Meeting to Collect and Evaluate Information on Procedures and Techniques for the Management of Failed Fuels from Research and Test Reactors; Budapest, Hungary, 29-31 October 1996.

[6] Perrotta,J.A; Yoshimoto,A; Riella,H.G; Souza,J.A.B; Durazzo,M; Jardim,E.A; **Fenômenos de Liberação de Hidrogênio de Elementos Combustíveis tipo MTR Observados no Reator IEA-R1**; III CGEN, Rio de Janeiro, Brazil, July 1990.

[7] Terremoto,L.A.A; Zeituni,C.A; Perrotta, J.A; **Espectrometria Gama em Elementos Combustíveis Tipo Placa Irradiados**; XI ENFIR, Poços de Caldas, Brazil, 18-22 August 1997.