Almacenamiento seco prolongado de combustibles nucleares: Influencia del diseño y de las condiciones de su irradiación comercial

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Resumen – *El código BaCo ha sido utilizado para la simulación del comportamiento de un combustible tipo PHWR en condiciones de almacenamiento mostrando una fuerte dependencia con el diseño original del combustible y su historia de irradiación. En particular, el resultado del análisis probabilístico de BaCo indica que la integridad del combustible está influenciada por las tolerancias de fabricación y las solicitaciones termomecánicas durante la irradiación en la CN. El presente estudio concluye que la temperatura del dispositivo deberá controlarse cuidadosamente para mantener condiciones seguras de almacenamiento.*

Abstract – *The BaCo code was applied to simulate the behaviour for a PHWR fuel under storage conditions showing a strong dependence on the original design of the fuel and the irradiation history. In particular, the results of the statistical analysis of BaCo indicate that the integrity of the fuel is influenced by the manufacture tolerances and the solicitations during the NPP irradiation. The main conclusion of the present study is that the fuel temperature of the device should be carefully controlled in order to ensure safe storage conditions.*

I. INTRODUCTION

At present the Atomic Energy National Commission of Argentina (CNEA) and Nucleoeléctrica Argentina (NASA) are evaluating the best solution for the intermediate storage of the irradiated fuels produced for the nuclear power plants (NPP) of our country.

Argentina has two NPPs in operation (Atucha I –PHWR, pressure vessel type– and Embalse –CANDU type–) and one under construction (Atucha II –PHWR, pressure vessel type–).

It is mandatory to guarantee the integrity of the fuels in these temporary systems of storage, to determine the main characteristics of design of that system and to assess the behaviour of the fuels during the entire time of storage without fuel failures and free of fission gas release to the environment [1].

The as fabricated tolerance of the fuel elements are defined in order to improve the economy, to reduce the procedures of manufacturing and to guarantee a success commercial irradiation. The normal rules of fuel design are not taking into account the behaviour and integrity after the commercial irradiation in the NPP. Due to these reasons we need to overview the fuel behaviour evolution at different temperatures of dry storage and to determine if the fuels are enable to resist the new conditions without any failure.

The BaCo code was developed for the analysis of the behaviour and the design of nuclear fuel rods [2]. BaCo is used for the simulation of the fuel rod behaviour at these new conditions, in particular taking into account the range of temperatures of the system of storage in order to determine the most plausible temperature for the system.

We determine that the main change of behaviour between the irradiation in the NPP and the dry storage system is due to the change of the stress-strain state of the cladding. Usually the cladding is working at a compression state (for the presence of the coolant pressure), but at the storage conditions (at approximately normal conditions of pressure) the cladding is working at a traction state. Simultaneously the release of fission gas continues without the production of fission gases but enough to increase the inner fuel rod pressure and we have enough I and Cs in order to induce a stress corrosion cracking (SCC) failure in the fuel rod if we reach a proper value of hoop stress at the cladding. This behaviour depends of the temperature of the system and a high (but technically valuable) temperature could produce this mechanism of failure. This effect was found in experimental WWER fuels stored in these conditions [3].

We focus the present work in the fuel of the Atucha I NPP and a detailed analysis of the fuel rod is performed in order to establish the right correlation between the normal behaviour during irradiation, the as fabricated tolerances and the storage conditions.

II. SPENT FUEL STORAGE AT CNEA

The SNF (“Spent Nuclear Fuels”) are stored into installations of the own nuclear power plants. The Atucha I NPP stores its spent fuels in two existing interim wet storage pools up to the future transference to a temporary dry storage.

Atucha I will have enough room for the storage of spent fuel from the operation of the reactor till December 2014, after completing the rearrangement and consolidation of the spent fuels in the pools. If the operation is extended beyond 2014, or if the reactor is decommissioned, it will need to empty both pools and to transfer the spent fuels to a dry storage facility [4].

The spent fuels of the Embalse NPP are stored in pools for six years and then they are transferred to an intermediate dry system based on a silo made with concrete. These silos have a modular structure; we can build more silos in the case of a life extension of Embalse.

The number of NPPs in Argentina is not enough to justify a centralized system of temporary storage of the spent fuels.

Taking into account the Strategic Plan made for the management of the radioactive waste and the spent fuels, until the year 2030 Argentina must choose between reprocessing or to carrying out a direct final disposition. But, for both cases, we will need a deep final repository approximately for the year 2060. At present, we participate in international programs of the IAEA and geological investigations are performed in order to identify the best place and rocks for this final disposition [5].

III. THE BACO CODE

The BaCo code (“Barra Combustible”, Spanish expression for “fuel rod”) was developed at CNEA for the simulation of the behaviour of nuclear fuel rods under irradiation. The development of BaCo is focused on PHWR fuels as the CANDU and Atucha ones but we keep a full compatibility with PWR, BWR, WWER and PHWR MOX among advanced, experimental, prototypes and/or unusual fuels.

BaCo assumes azimuthal symmetry in cylindrical coordinates for the fuel rod; the model is bidimensional and angular coordinates are not considered. However, angular dependent phenomenon, as well as radial cracking, is simulated via some angular averaging method [5]. For the numerical modelling the hypotheses of axial symmetry and modified plane strains (constant axial strain) are adopted. The fuel rod is divided in axial sections in order to simulate its axial power profile dependence. The mechanical and thermal treatment and the pellet, cladding and constitutive equations are available from reference [2].

The BaCo code structure and models in its present versions have already been described in references [2], [7]and [8]. A complete application example of BaCo for PHWR fuel design is included in reference [8]. The strategy for the development of the code is presented in reference [7] with a brief description of the code. Statistical analysis, data post-processing and BaCo 3D tools improves the code's performance and analysis of results [6, 9].

On modelling the UO_2 pellet behaviour, phenomena such as elastic deformation, thermal expansion, creep, swelling, densification, restructuring, cracks and fission gas release, are included. For the Zry cladding, the code models the elastic deformation, thermal expansion, anisotropic plastic deformation, and creep and growth under irradiation. The modular structure of the code easily allows to add different material properties. It can be used for any geometrical dimensions of cylindrical fuel rods with UO_2 pellets (either compact or hollow, with or without dishing) and Zry cladding.

BaCo is used for fuel design, analysis of the expected performance and development of a future irradiation tests as in the cases of the CARA Fuel Project [10] and for the fuel for the CAREM reactor [11].

IV. PHWR FUELS BEHAVIOUR

The analysis of a fuel dry storage begins with the understanding of the normal behaviour under commercial irradiation in order to determine the EOL ("End Of Life") fuel morphology. The focus of the present analysis is the fuel of the Atucha I NPP due to the needs of a definition of several issues of the silo for the fuels dry storage, in particular the temperature of the storage.

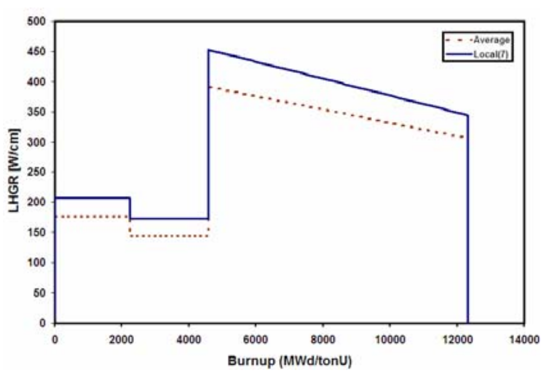

Fig. 1. A real and demanding power history of a SEU fuel of the Atucha I NPP. The doted curve is the average linear heat generation rate (ALHGR) and the other one is the local LHGR at the 7th axial segment of the fuel.

The power history defined for the analysis of an Atucha fuel is based on a realistic irradiation at the NPP. The selected fuel corresponds with the most demanding scenario found in the Atucha I NPP. The discharge burnup is 12000 MWd/tonU (more than 400 days). The Atucha I NPP has a 530 cm fuel stack length fuel, SEU type ("Slightly Enrichment Uranium", 0.82 % of enrichment). The fuel is divided in 10 axial sections for the neutronic calculation. The seventh axial segment (from the top) is the most demanding one (see Fig. 1 where a reduced power history is sketched). There is one on line reshuffling at ~4500 MWd/tonU.

The Fig. 2 shows the pellet centre calculation of the BaCo code. We find a maximum value of ~1600°C. The curve of the Vitanza threshold shows a local FGR ("Fission Gas Release") over 1 %. Pellet surface and inner cladding surface temperatures are included in the Fig. 3.

Equiaxed grains (up to a radius of the forth of the original pellet radius) were found after the first reshuffling and columnar grains appears at the centre of the pellet over ~6000 MWd/tonU (see the Fig. 4).

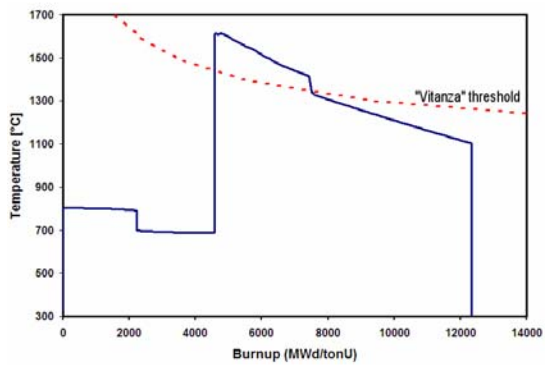

Fig. 2. Pellet centre temperature of the pellet in the 7th axial segment (the most demanding position) of a fuel rod of in the Atucha I NPP. The "Vitanza threshold" is included as a reference for the fission gas release.

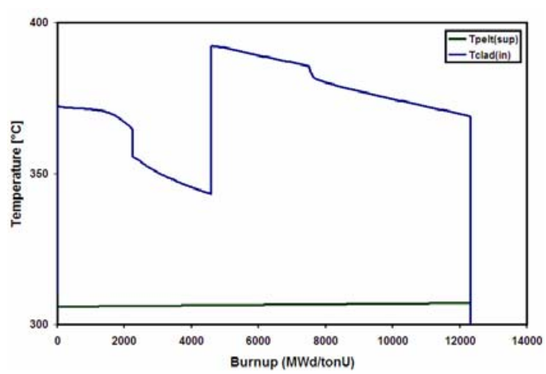

Fig. 3. Temperature in the gap between the pellet (top curve) and the cladding (lower curve).

The FGR ("Fission Gas Release") at EOL is ~0.7 % (less than the 1% due to the power axial profile -see the Figs. 5 and 2-). The power ramp (fuel reshuffling at ~4500 MWd/tonU) releases the most of the free fission gases present during the irradiation. The Fig. 6 shows the dynamics of the free fission gases. We included in that plot the produced gases, the gas at the grain boundary (~650 cm³ at EOL), the grain at the UO_2 matrix (~1000 cm³) and the gas released (~11 cm³). The most of the free gases in the rod at EOL is He (~94 % He, ~5% Xe and ~1% Kr -see the Fig. 7-) due to the Atucha fuel a pressurized one.

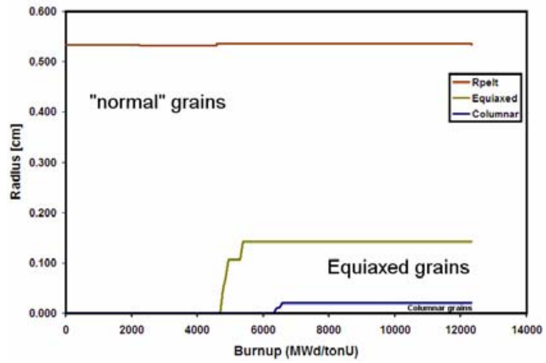

Fig. 4. Grain size evolution during irradiation at the 7th axial section of an Atucha I fuel. The top curve is the evolution of the pellet radius, the middle one is the limit for the equiaxed grains and the lower for the columnar grains. No central hole was found.

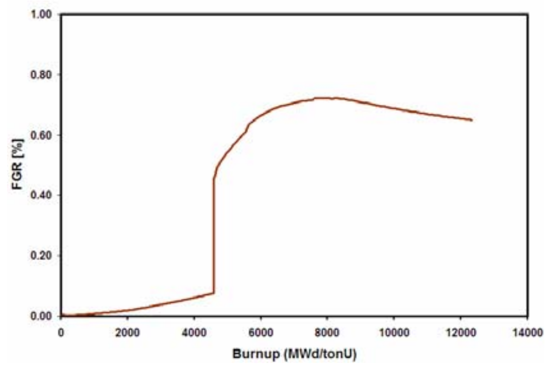

Fig. 5. FGR ("Fission Gas Release") in a fuel rod of the Atucha I NPP. The FGR at EOL is less than 1 %.

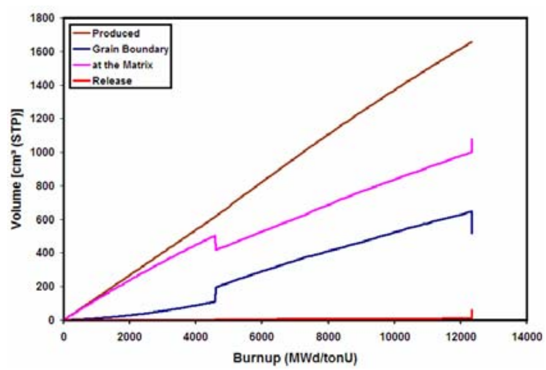

Fig. 6. Fission gases in the fuel rod. The top curve represents the fission gases produced during irradiation. The next curves are the fission gases in the matrix of the UO2 grains and the gases at the grain boundary. The lower one is the fission gas released in the fuel rod.

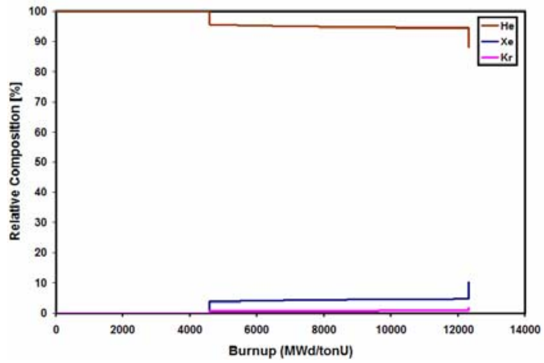

Fig. 7. Relative composition of the free gases in the empty places of the fuel rod (the Atucha I fuels is pre-pressurized)

The inner pressure due to the free gases in an Atucha I fuel rod is included in the Fig. 11. That pressure is lower than the coolant ones during all the time of the commercial irradiation of the fuel as it was originally designed.

The Fig. 12 shows the pellet cracks opens for tangential stresses. We find cracks opens from the surface up to the approximately the middle of the pellet radius. The central zone of the pellets is under a compression state.

The calculation of the fuel stresses at the most demanding axial segment of the fuel shows a negative hoop stress in the cladding (tangential stress at the inner surface) during all the time of the irradiation (see Fig. 13). That means the cladding is under compression, at least by using the nominal values of design of the Atucha I fuel. The peak at EOL is due to the change from the reactors conditions to discharge ones outside the pressure vessel.

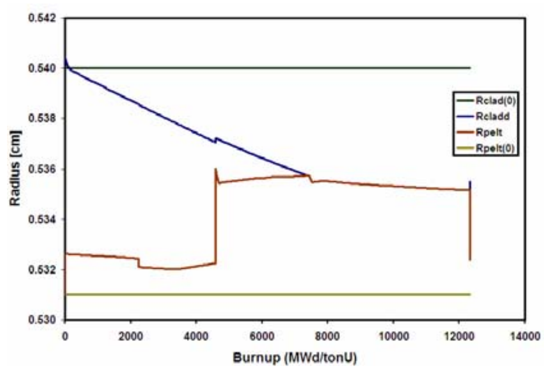

Fig. 8. Fuel rod radius evolution at the 7th segment of an Atucha I fuel during the irradiation. The parallel lines are the as fabricated pellet and inner clad radius. The top curve is the inner clad radius and the other one is the pellet radius. Pellet-clad contact is found over 6500 MWd/tonU.

The Fig. 8 shows the evolution of the pellet and cladding radius (where we include the original radius as a reference value). We observe the creep-down of the cladding. A peak at reshuffling produces a plastic deformation of the cladding. A decrement in the fuel radius is done for the reduction of temperature at ~7500 MWd/tonU due to the contact between pellet and cladding. The Fig. 9 shows the calculation of the cladding radius. The radial and axial deformations of the cladding were less than ~0.3 % at EOL. The Fig. 10 shows the thermal conductance in the gap pellet-cladding. It appears as the reflex of the radius evolution and the fission gases dynamics. The best values are after pellet-cladding contact.

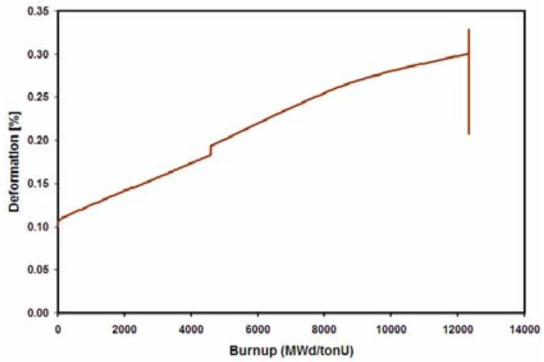

Fig. 9. Radial deformation of the cladding of an Atucha I fuel at the 7th axial segment.

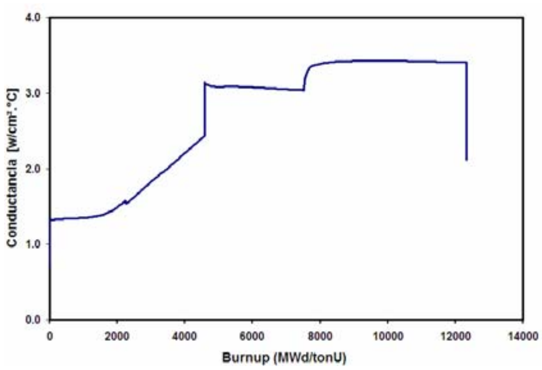

Fig. 10. Thermal conductance in gap pellet-cladding at the 7th segment of an Atucha I fuel.

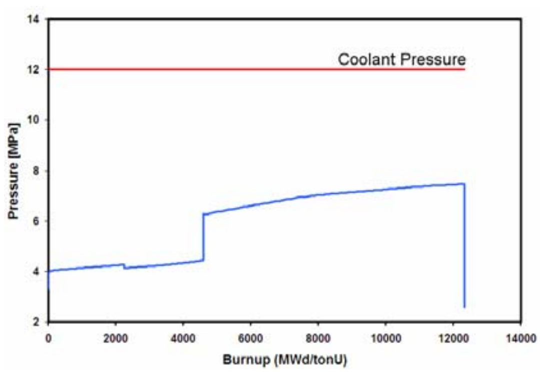

Fig. 11. Inner pressure of the fuel rod due to the free gases in an Atucha I fuel rod. The top curve is the coolant pressure of the Atucha I NPP. The fuel is under safe conditions (Pcoolant > Pgas).

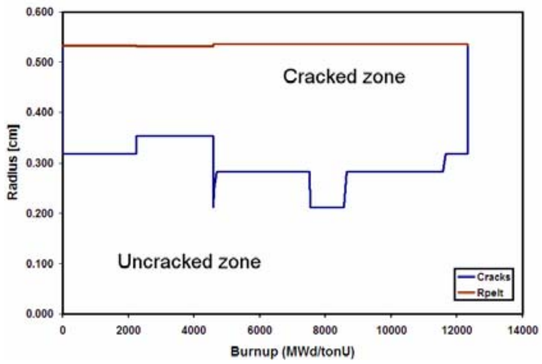

Fig. 12. Fuel pellet cracks opens due to tangential stresses at the most demanding axial position (7th from the top) of an Atucha I fuel. These cracks are not present in the centre of the pellet.

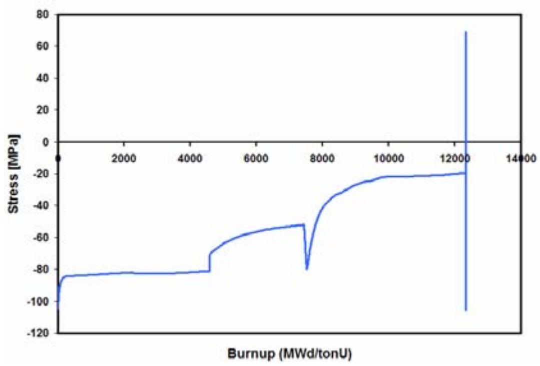

Fig. 13. Hoop stress (tangential stress) at the inner surface of the cladding at the 7th segment of an Atucha I fuel during the irradiation. The cladding is under a compression state up to the end of the irradiation.

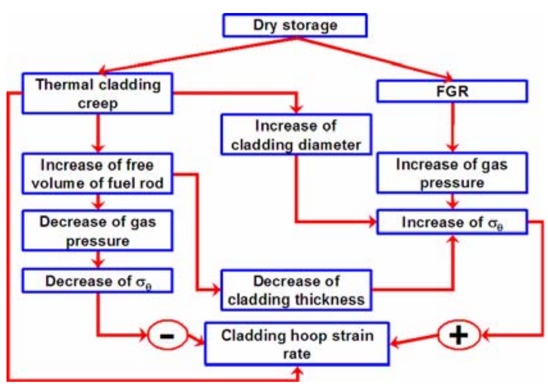

Fig. 14. The simplified flux diagram of the fuel rod behaviour under dry storage conditions (based on ref. 3).

IV. A FUEL UNDER DRY STORAGE CONDITIONS

The creep and fission gas dynamics in UO_2 is a function of temperature and burnup among others [12]. We not cumulate burnup for the new dry storage conditions after the commercial fuel irradiation in the NPP. The storage conditions can be reduced to steady state behaviour without a nuclear inner heat source and simplified boundary conditions, at least from the point of view of the BaCo code. That behaviour will be strongly conditioned for the accumulated burnup, the fission product inventory in the rod and the pellet morphology at EOL.

The Fig. 14 shows the change in the fuel rod characteristics under the dry storage conditions on fuel cladding hoop strain [3]. The amount of corrosive fission product is enough to initiate SCC ("Stress Corrosion Cracking") if a hoop stress threshold is reached in the cladding. A value of $\sigma_{\theta,SCC}$ of 170 MPa is accepted as a lower limit for the fuel of the Atucha I NPP [13]. Severe power ramps are normal for the PHWR fuels due to the on line fuel reshuffling. We could find that the $\sigma_{\theta,SCC}$ limit was allowed during those operations (usually 2 per fuel during irradiation). Then that limit could be empirically reduced in the time of the dry storage.

The temperature for the dry storage device must take into account the reduction of the stresses up to a safe limit. Those stresses and limits could be the results of the fuel performance and the classical rules of nuclear fuel design.

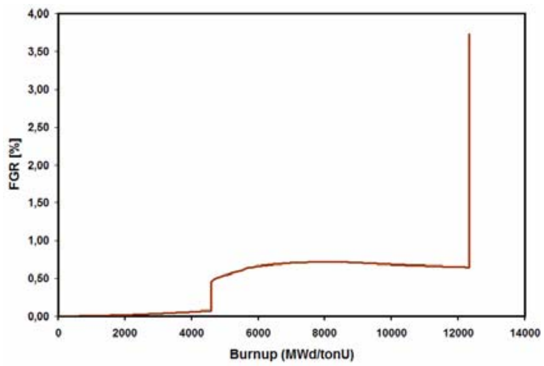

Fig. 15. Fission gas release (FGR) during irradiation and dry storage conditions for an Atucha I fuel in function of the average burnup. The FGR at EOL ("End Of Life") is less than 1 %. The final peak is the release during dry storage at 300ºC (without irradiation conditions –no burnup-).

VI. DRY STORAGE TEMPERATURE

Taking into account the flux diagram of the Fig. 14 we can determine that the temperature will be the main parameter for the final design of the dry storage system. We can simulate this condition with the BaCo code by using an extension of the normal power history under irradiation (see Fig. 1). We continue the calculation after EOL with the new boundary conditions (in particular pressure and temperature of the dry system) and without an inner source of heat in the pellet (zero power and burnup). Then the calculation is strongly simplified and the behaviour will just present small variations but with a widest time scale. The Fig. 15 includes de FGR against burnup with this new condition (see the Fig. 5). The peak at EOL represents the FGR for ~10 years of storage in dry conditions (mainly a temperature of 300°C). There is an increment of the FGR due to the thermal condition of storage. The Fig. 16 is the same of the previous one but against the time. FGR increases continuously in time due to a process thermally activated.

The inner pressure of the fuel rod due to the free gases in an Atucha I fuel rod is included in the Fig. 17 by using different values for the temperature from 100°C to 400°C. This sensitivity analyses will help us in the definition of the temperature for of the dry system. We observe that during irradiation $P_{coolant} > P_{gas}$ (where $P_{coolant}$ is the coolant pressure and P_{gas} gas pressure) and, during the dry storage, $P_{atm} < P_{gas}$ (where P_{atm} is the atmospheric pressure). This means a different stress-strain state in both step of the fuel life (the cladding is under compression during irradiation and traction at the dry storage). Following the flux diagram of the Fig. 14 we find a small and continuous increase of the pressure.

The analysis of the stresses in the fuel is focussed in the hoop stress of the cladding (inner surface). The Fig. 18 shows the hoop stress for the same temperatures of the previous analysis. The cladding is in the state that we observed with the previous analysis. The values of the hoop stresses are under the limits for a SCC event, Nevertheless the fuel included the mechanical works over the cladding during irradiation (stress reversal, PCI –"Pellet Cladding Interaction" –, PCI-SCC –"Stress Corrosion Cracking"–, etc.) and a corrosive inventory of fission gases (I, Cs, etc).

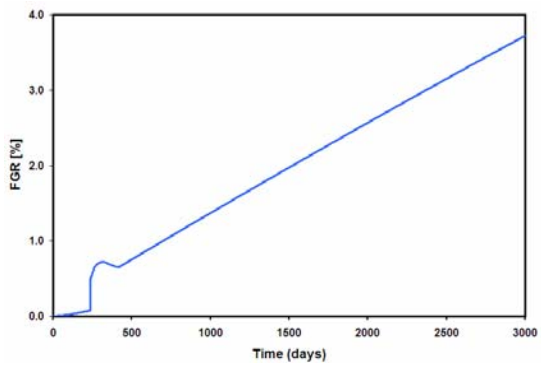

Fig. 16. The plot of the Fig. 15 but vs. time. FGR is due to a process thermally activated. Temperature of storage 300°C.

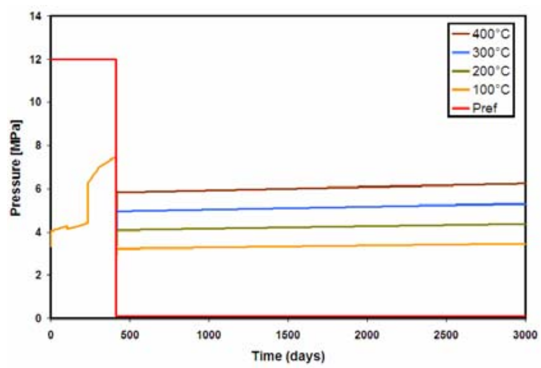

Fig. 17. Inner pressure of the fuel rod due to the free gases in Atucha I. The red curve is the pressure out of the cladding. We take into account four different temperatures for the storage.

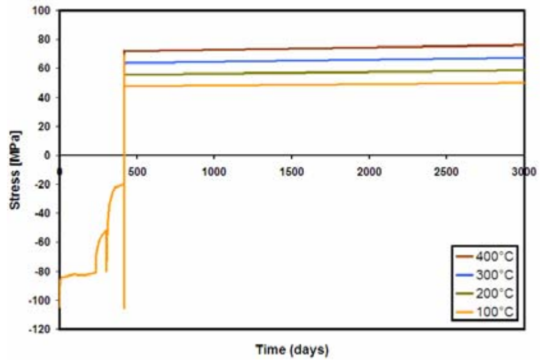

Fig. 18. Hoop stress (tangential stress) at the inner surface of the cladding at the 7th segment of an Atucha I fuel during the irradiation and the dry storage. The cladding is under a traction state at the dry storage conditions.

VII. STATISTICAL ANALYSES

The previous calculations were performed by using the nominal value of the fuel rod parameters, in particular the rod geometry. Then we obtained a conservative an average result compatible with the classical design of fuel elements. Nevertheless those parameters have included as fabricated tolerances due to the process of manufacturing, the economy, the performance and the safety. The “statistical or probabilistic analysis” included in the BaCo code is a M-C (“Monte-Carlo”) technique, which combine several random of fuel parameter (input data) with its statistical distribution. Each BaCo code calculation included random fuel data compatible with the as fabricated tolerances and its distribution. Then each input data represent a real segment of the fuel rod, and the series of M-C calculation have a significant impact on the calculated results [9].

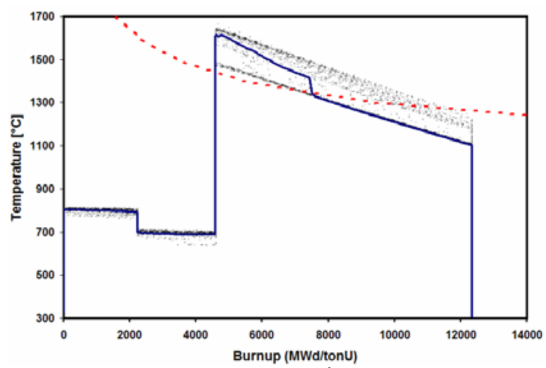

Fig. 19. Statistical analysis [9] of the fuel centre temperature at the 7th segment of an Atucha I during commercial irradiation. The cloud of dots is a representation of the dispersion of the thermal calculations of BaCo.

The Fig. 19 represents the statistical calculation of BaCo for the pellet centre temperature for an Atucha I fuel at the most demanding axial segment of the rod. We find a band of temperature of ~250ºC. Meanly the dispersion is the result of the gap size between pellet and cladding. This gap could be closed before or after the power ramp or directly open during the irradiation.

The Fig. 20 shows the statistical dispersion of the FGR during the irradiation and the dry storage. The Fig. 21 includes the statistical calculation of the inner pressure of the gases in the rod for a 300ºC at the storage device plus the normal results for the previous temperatures. We find that during the irradiation the pressure is increasing its dispersion and at the storage conditions the dispersion remains without major changes. The minimum and maximum results of the calculation are over the next (and previous) range of temperature.

The statistical hoop stress calculation shows a peak during the on line reshuffling due to the presence of small gaps pellet-cladding compatibles with the tolerances. Those cases produce the strongest events of PCI ("Pellet Cladding Interaction"). We obtained values slowly under the limits of the $\sigma_{\theta,SCC}$. We mentioned that the threshold for PCI-SCC does not mean a failure; it means the probability of a failure after that event. We have not PCI during the dry storage due to the boundary conditions. Nevertheless, an event of SCC and a failure could be done during the dry storage for the cumulative damage during the commercial irradiation of the fuel.

At present the fuel elements design do not take into account the integrity of the cladding after the irradiation in the NPP.

The Fig. 20 to 22 clearly shows a similar dispersion during all the time at the storage dry system. The dispersion is done during irradiation conditions due to the as fabricated tolerances.

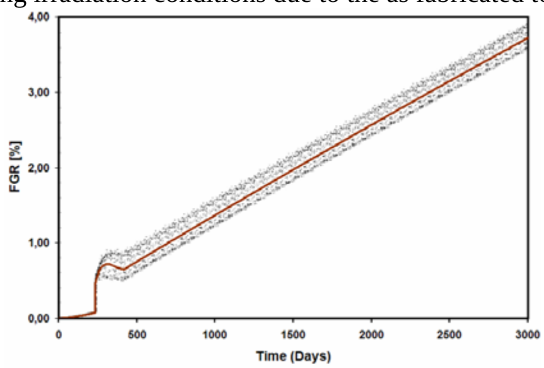

Fig. 20. The plot of the Fig. 16 including the statistical analysis of the fission gas release (FGR) of an Atucha I fuel during irradiation and dry storage (300°C of temperature for the storage device). The cloud of dots is a representation of the dispersion of the FGR calculations of BaCo.

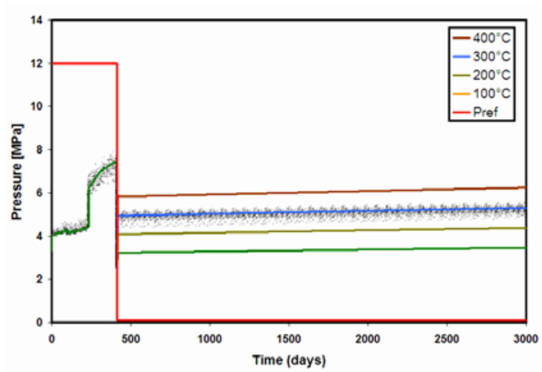

Fig. 21. The plot of the Fig. 17 including the statistical analysis of the inner pressure of the fuel rod due to the free gases in an Atucha I fuel rod for a storage temperature of 300°C.

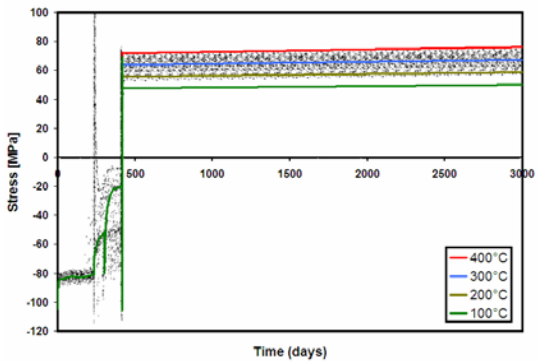

Fig. 22. The plot of the Fig. 18 including the statistical analysis of the hoop stress during the irradiation and the dry storage for an Atucha I fuel rod with a storage temperature of 300°C.

VIII. AN OVERVIEW OF EXPERIMENTAL TESTS

Experimental evidence [3, 4] suggest a temperature for the dry storage of ~300°C. The statistical analysis suggests a reduction of that temperature up to a conservative value of ~200°C taking into account the dispersion of the M-C calculations.

The short term experiments performed for WWER fuels [3] show a small increment of the hoop stress, hoop deformation and FGR. Those increments have a qualitative agree with our first estimations.

IX. CONCLUSIONS

We presented a detailed analysis of the behaviour of a PHWR fuels as the Atucha I ones because that is the starting point for the understanding of the needs for a safe operation with the dry storage system, in particular the determination of the temperature of the system.

The design of the dry system taking into account the possible transients, accidents and/or abnormal events that can occur in SNF storage is not enough to guarantee a conservative state during all the time of that period. We demonstrated the close relation between the integrity of the SNF with the history of the fuel previous to the storage conditions, in particular the as fabricated tolerances and the events during the irradiation. The power transients during irradiation could be the way to increase the probability of the lost of integrity of the fuel rod during the dry storage. A right fuel element design taking into account a safe operation during irradiation is not enough to guarantee safe storage conditions. Abnormal or severe operations of the fuels during irradiation could require other extra conditions more conservative.

We must to improve the Zry creep, the FGR and the clad corrosion models for a better understanding of the general phenomena taking place in the dry storage system. Nevertheless we strongly remark the needs to include the statistical analysis as the BaCo way for safety storage and an enhanced design of a dry storage system.

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