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CONSERVATIVE PERFORMANCE ANALYSIS OF A PWR NUCLEAR FUEL ROD USING THE FRAPCON CODE

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ABSTRACT

In this paper, some of the preliminary results of the sensitivity and conservative analysis of a hypothetical pressurized water reactor fuel rod are presented, using the FRAPCON code as a basic and preparation tool for the future transient analysis, which will be carried out by the FRAPTRAN code. Emphasis is given to the evaluation of the cladding behavior, since it is one of the critical containment barriers of the fission products, generated during fuel irradiation. Sensitivity analyses were performed by the variation of the values of some parameters, which were mainly related with thermal cycle conditions, and taking into account an intermediate value between the realistic and conservative conditions for the linear heat generation rate parameter, given in literature. Time lengths were taken from typical nuclear power plant operational cycle, adjusted to the obtention of a chosen burnup. Curves of fuel and cladding temperatures, and also for their mechanical and oxidation behavior, as a function of the reactor operation´s time, are presented for each one of the nodes considered, over the nuclear fuel rod. Analyzing the curves, it was possible to observe the influence of the thermal cycle on the fuel rod performance, in this preliminary step for the accident/transient analysis.

1. INTRODUCTION

Accidents foreseen for nuclear reactors are called the basic project accidents and severe accidents. Basic project accidents are related to events involving loss of the plant´s primary coolant, also known as LOCA (“Loss of Coolant Accident”), and are also divided in two classes, due to the loss of the coolant, by large or short pipe breaks, in the primary circuit of the nuclear power plant. After the Fukushima accident [1], it was established that events beyond of the basic projects and involving the melting of the nuclear fuel, must also be considered for the safety analysis simulations. Thus, computational programs were developed to carry out those analysis, and to predict the behavior of the nuclear fuel rods, during transient or accident conditions.

Performance analysis of nuclear fuel rods, under conservative and accident or transient conditions, are usually carried out by programs like FRAPCON (“Fuel Rod Analysis Program Conservative”) which simulates fuel rod behavior under normal operation of the nuclear power plants, and FRAPTRAN (“Fuel Rod Analysis Program Transient”), to simulate fuel rod behavior under accident and transient conditions.

FRAPCON [2] is a FORTRAN 90 code designed for the precise evaluation of the light water reactors fuel rods under long burnups, and the assumptions for the power and the boundary conditions variations are such that the system is considered in stationary state. Phenomena modeled by the code are (1) heat transfer from the fuel and cladding to the coolant, (2) plastic and elastic deformation of cladding, (3) fuel-cladding mechanical interaction, (4) fission gas release, (5) swelling and fuel densification, and (6) cladding oxidation and hydration.

FRAPTRAN [3] is also a FORTRAN 90 code, but it is designed to predict the fuel rod behavior under accidents like LOCA and RIA ("Reactivity Initiated Accidents"). The code calculates the temperature and deformation history of the fuel rods, as a function of the power generated by the fuel and from the coolant conditions. It can be used as an independent code, and also as a FRAPCON companion code, from which it takes the results of the conservative simulations as inputs.

In order to contribute to the studies in accident analysis being carried out at IPEN [4], in this paper, some of the results of the sensitivity and conservative analysis of a fuel rod of a pressurized water reactor are presented, using the FRAPCON code as a basic and preparation tool, for the future transient analysis, which will be carried out by the FRAPTRAN code.

Sensitivity analyses were performed by the variation of the values of some parameters, and in this paper we will show the results related to thermal cycle conditions. Emphasis is given on the cladding behavior, since it is a critical barrier to the containment of the fission products generated by the fuel irradiation.

The objective is to obtain the best input data generated by FRAPCON, preparing the simulations for the future analysis of the fuel rod behavior under accident conditions, to be carried out by the FRAPTRAN code.

2. EXPERIMENTAL PROCEDURE

The rod chosen for the performance analysis was simulated at the center of the nuclear reactor core, having typical PWR design characteristics, as given in Table 1. In this table are also shown some of the numerical parameters, fuel pellet dimensions, reactor operation and flow conditions, used in this simulation. The parameter chosen for the sensitivity analysis, among almost one hundred parameters of the FRAPCON code, was thermal cycle. It is a relation of the linear heat generation rate, by the fuel in the rod, with time. Cycles are given in Figs. 1, 2 and 3, for the three simulations named Experiments 1, 2 and 3.

The maximum linear heat generation rate admissible for our simulations was chosen between the values of the realistic and conservative conditions, respectively 38,2 kW/m (11.643 kW/ft) and 53,3 kW/m (16.246 kW/ft)**,** in accordance with the FSAR ("Final Safety Analysis Report") of Angra 2 [5]. Two of the cycles were chosen based on a typical cycle of a reactor operation, and the other to agree with the times for which the FRAPCON code was designed.

After setting up of the input parameters in the FRAPCON input file, curves were generated simulating the performance of the fuel and cladding. For example, curves of fuel (uranium dioxide) and cladding (zircaloy-4) temperature as a function of time, cladding deformation and oxidation, fuel temperature behavior, among others, were studied. The results were summed up in a separate table, Table 3, containing critical (extreme) values for parameters like, burnup, rod internal pressure, fuel rod void volume, fission gas release, fuel centerline temperature, maximum strain rate, to show some of the main results of the simulations.

Values for the limiting cladding conditions under LOCA (which is going to be simulated in future works) which serve as guidance for the conservative simulations, were taken from FSAR of Angra 2 [5]. Among others, they are 2192°F (1200°C) for its external temperature, and 17%, for its local oxidation. Parameters were chosen in order to keep the cladding very far from these critical conditions.

Table 1: Values for some of the FRAPCON simulation parameters (British units).

Rod design		Numerical		Operational	
Parameter	value	Parameter	value	Parameter	value
External diameter	4,2323 e-1	Radial nodes	17	Coolant Pressure	2291.6
Thickness	2.854 e-2	Axial nodes	17	Water temperature	556.34
Gap thickness	3.745 e-3	Simulation steps	48	Coolant mass flow	2.328 e6
Rod length	12.7953	Gas nodes	45	pitch	5.63 e-1
Plenum length	6.2047			LHGR	Figs. 1, 2, 3
Pellet height	4.3307 e-1			time	Figs. 1, 2, 3
Pellet radius	3.5866 e-1				

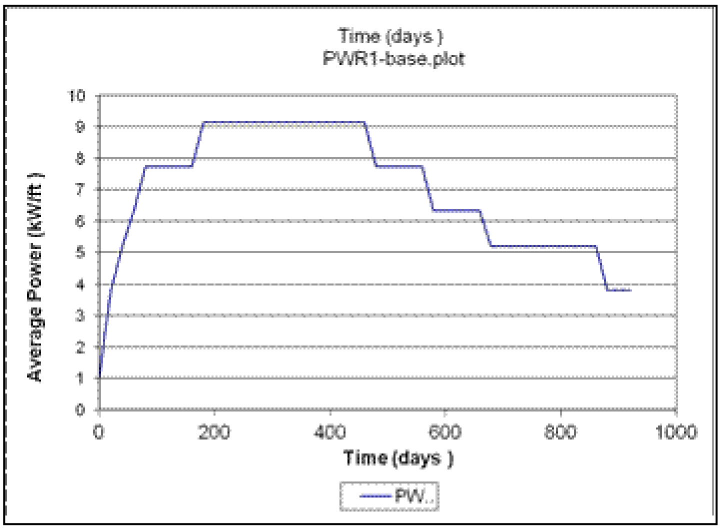

Figure 1: Power cycle for Experiment 1.

Experiment number 1 was designed to obey the requests of analysis of the FRAPCON 3.4 code. The time length of the operational cycle was nearly 900 days, Fig.1, a typical value of some of conservative simulations found in literature [2]. The main objective was to perform the analysis in an extended burnup condition, distinct from the operational ones, to check for fuel and cladding performance.

Experiments 2 and 3 were designed to simulate operational cycles of nuclear power plants, under a typical time interval. New simulations using different LHGR were carried out, in order to keep the same level of burnup in the experiments. Curves of thermal cycles for both experiments are given in Figs. 2 and 3. The main difference between them is the extension over which the reactor operates, in the maximum power level.

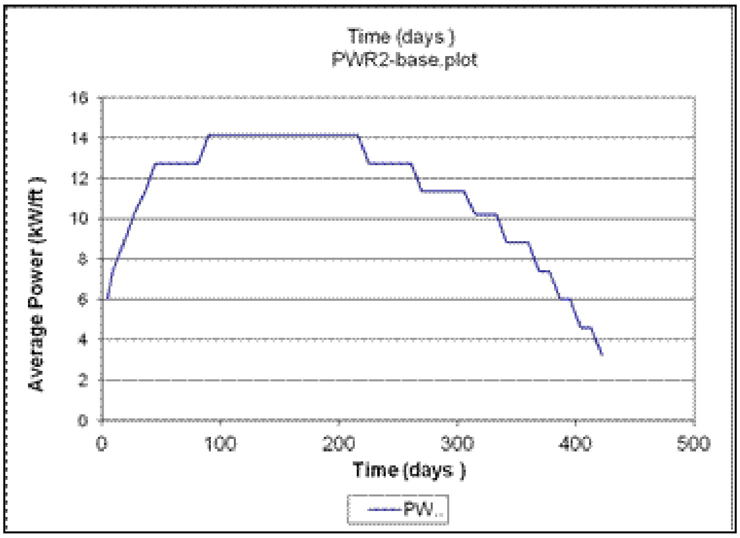

Figure 2: Power cycle for Experiment 2.

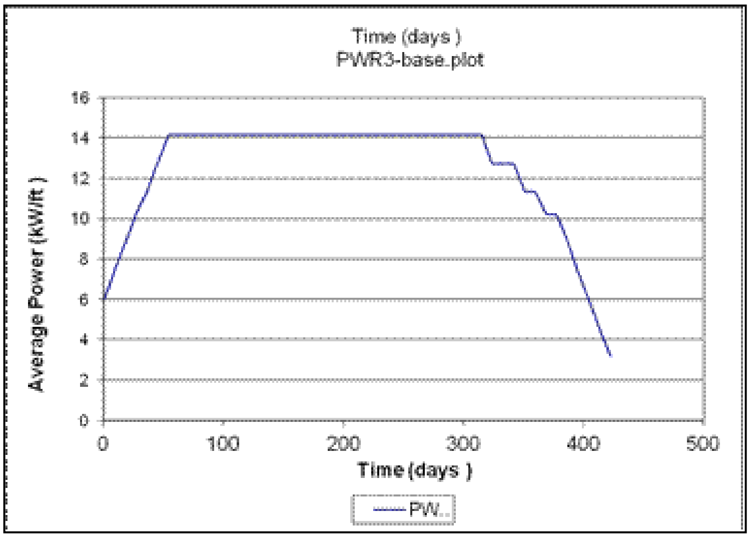

Figure 3: Power cycle for Experiment 3.

3. RESULTS AND DISCUSSIONS

Some of the phenomena covered by FRAPCON are represented in the Figs. 4 to 15, mainly related to the cladding mechanical properties. Table 2 shows a summary of the critical values of some of the properties, resulting from the simulations for the fuel, fuel rod and cladding.

From the curves of cladding average temperatures, Figs. 4 to 6, and from Table 2, the values of the temperatures maxima obtained for the 3 experiments were very far from the melting point of the cladding. A value of 394,25°C was reached at Experiment 3, the most severe one, against 381,34°C from Experiment 1 and 387,36°C from Experiment 2. Most important is that temperature did not grow above the safety limit, even in a condition near the realistic and above the conservative.

It can also be seen from a comparison of Figs.1 to 3 to Figs.4 to 6 that temperature and LHGR are linearly dependent, resulting from the model used by FRAPCON, to simulate the heat transfer phenomena.

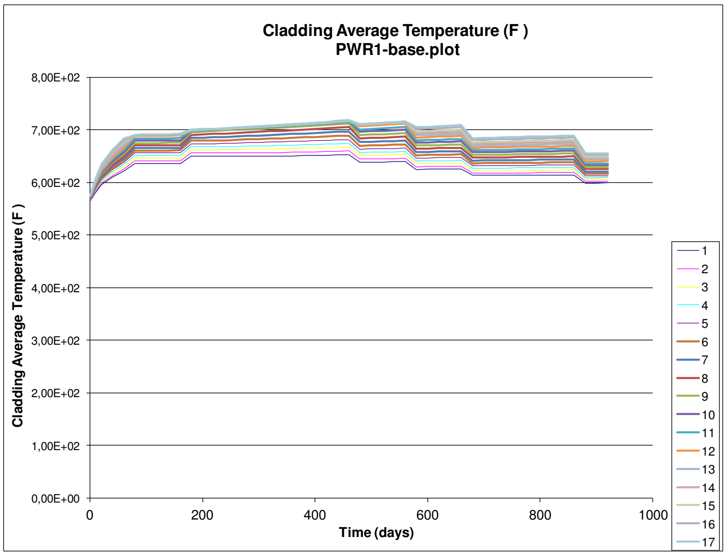

Figure 4 - Cladding average temperature, Experiment 1.

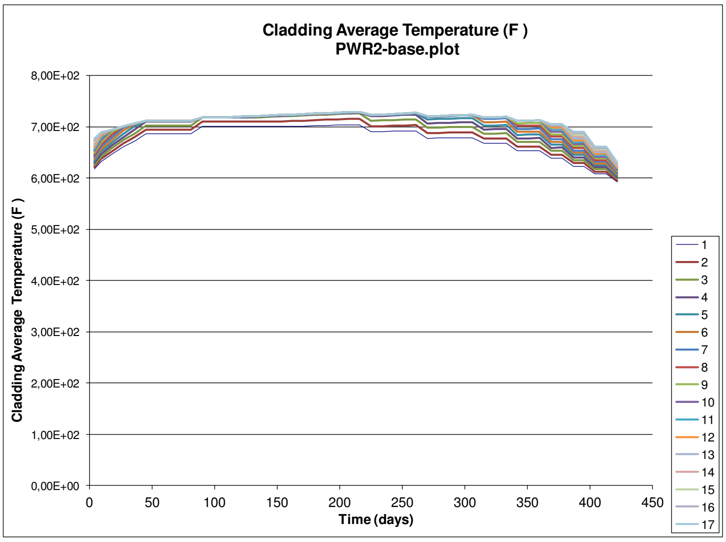

Figure 5 - Cladding average temperature, Experiment 2.

From the curves of Figs. 7 to 9, a contraction of the cladding and, consequently, of the total rod void volume, for times up to 220 days, was observed in Experiments 2 and 3. The same phenomenon was observed in Experiment 1, but with times up to 360 days. According to the results of the simulations, a relation between the end of the contraction of the cladding and the start of the phenomena of creeping was observed, in all three conditions.

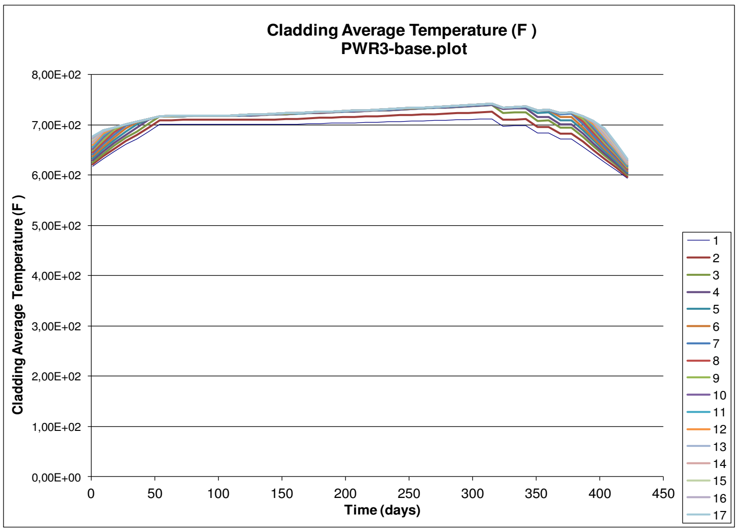

Figure 6 – Average cladding temperature, Experiment 3.

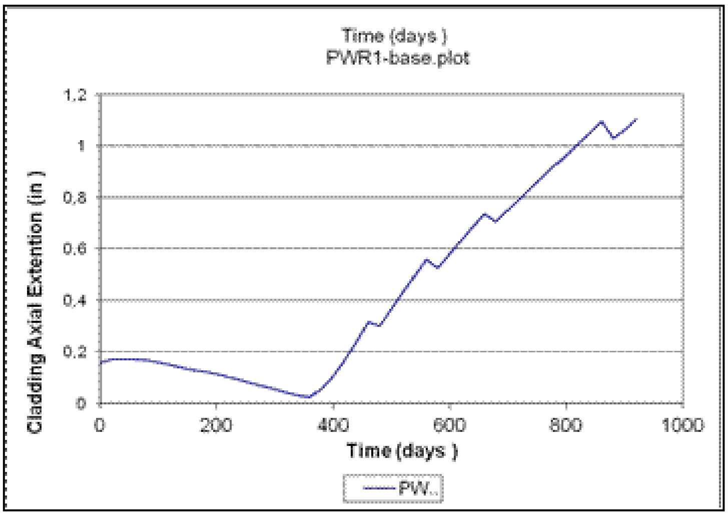

Figure 7 - Cladding extension, Experiment 1.

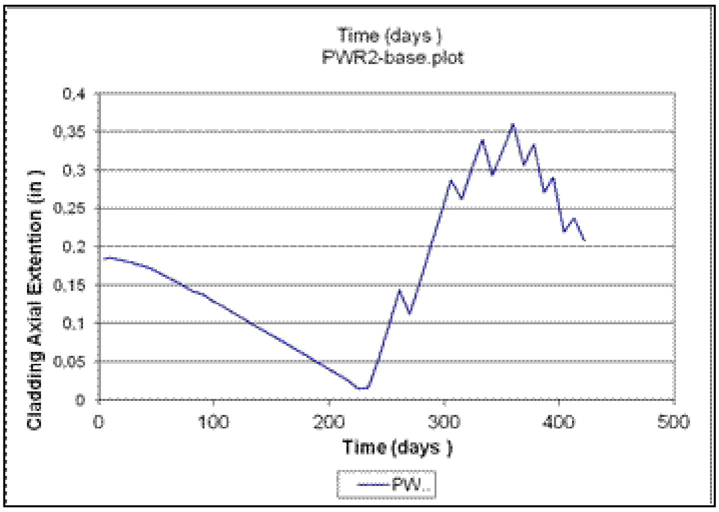

Figure 8 - Cladding extension, Experiment 2.

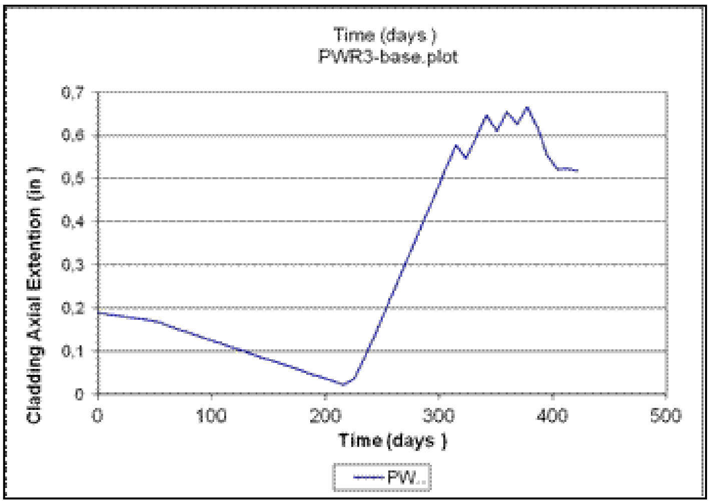

Figure 9 – Cladding extension, Experiment 3.

Differences in the behavior of the cladding appeared more intense in the next experiments. For the cladding axial stresses, it is shown in Figs. 10 to 12 that, despite of the longer time length of the Experiment 1, the cladding behaves uniformly, as also shown from the curves of Experiment 2.

However, a possible condition of rupture or numerical divergence was observed in Experiment 3, node 2, with a value of 14517 psia, after 400 days of operation, for the stress in this particular node. Despite this event, the program continued to run, indicating no failure of the rod, or no failure in the convergence method employed for the analysis.

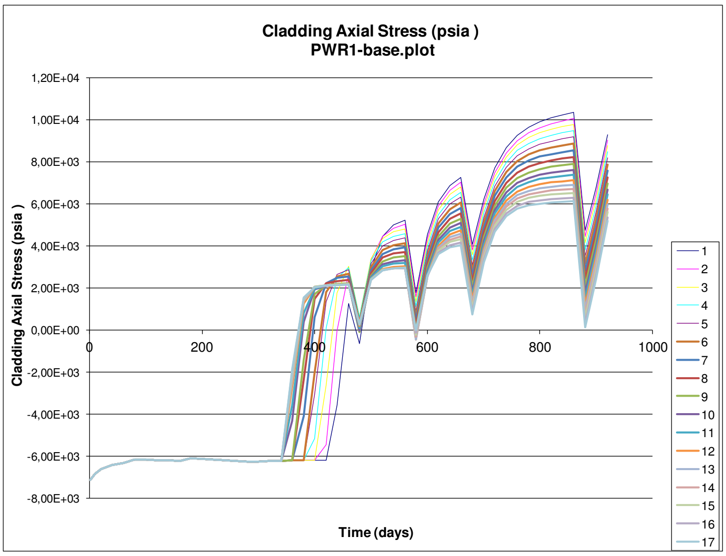

Figure 10 – Cladding axial stress, Experiment 1

Times for the start of the oscillations of the cladding axial stresses, for the 3 experiments, were observed to be related with the creep of the cladding, according to the results of the simulations.

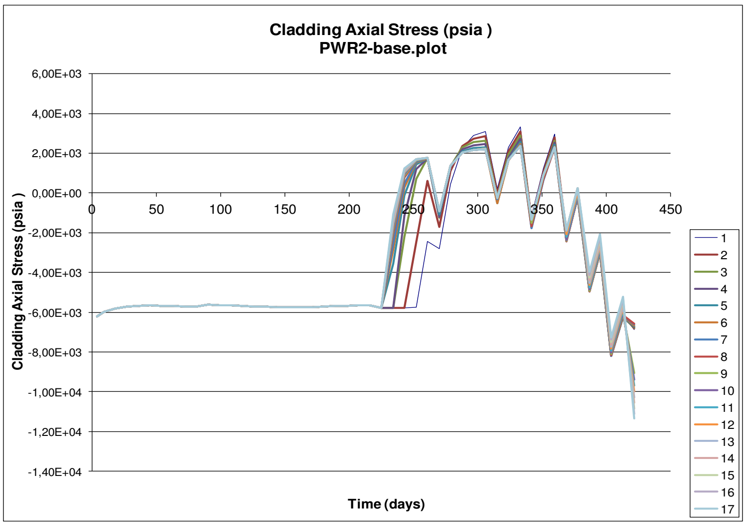

Figure 11 – Cladding axial stress, Experiment 2.

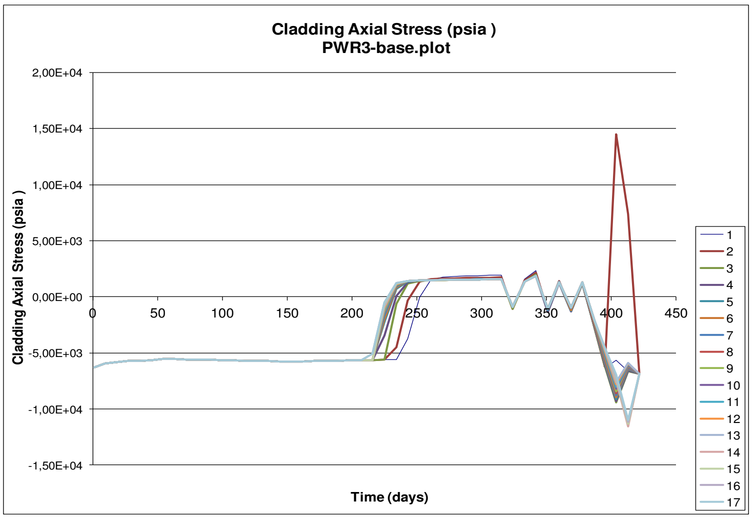

Figure 12 – Cladding axial stress, Experiment 3.

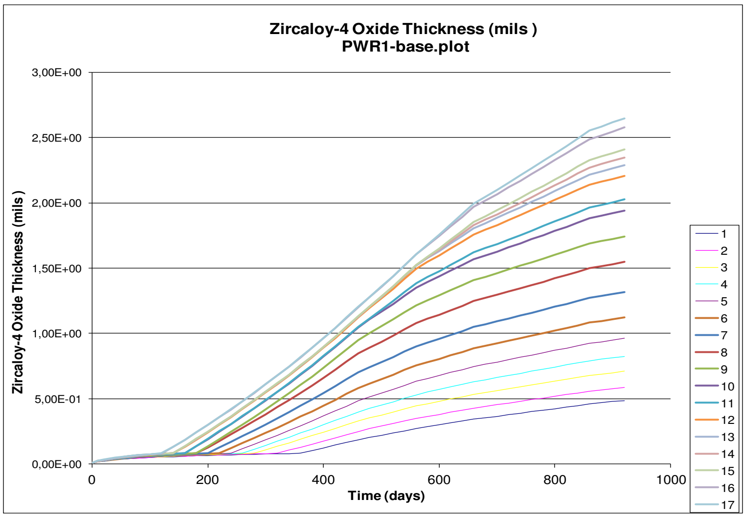

Figure 13 – Cladding oxidation, Experiment 1.

From the oxidation or oxide thickness curves of Figs. 13 to 15, it was obtained that cladding oxidation from Experiment 1 was larger than the obtained from Experiments 2 and 3, an expected result, due to the largest time of exposition under water of Experiment 1. When comparing Experiments 2 and 3, we see that the oxide cladding thickness of the Experiment 3, presented an increase of only 11%, even considering that there was a doubling of the exposition time of the fuel rod in the maximum value of the linear heat generation rate.

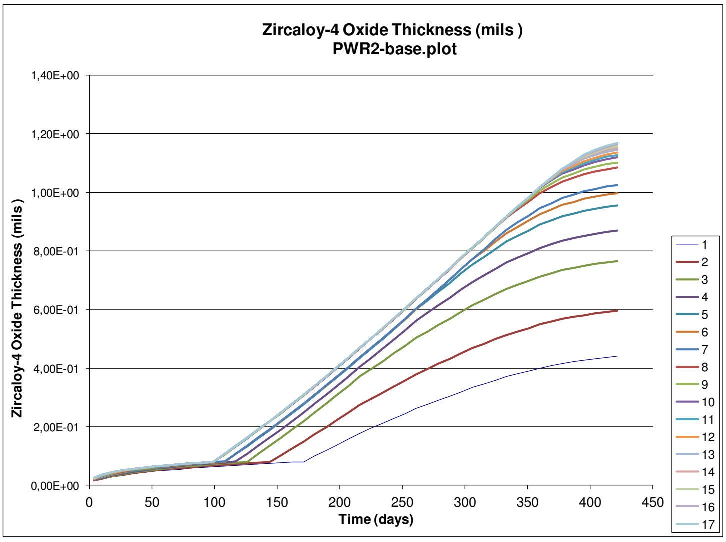

Figure 14 – Cladding oxidation, Experiment 2.

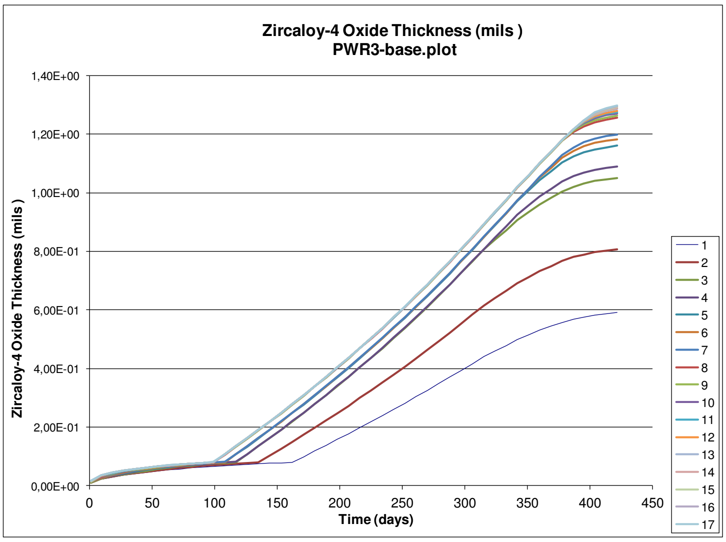

Figure 15 – Cladding oxidation, Experiment 3.

The results of some of the simulated properties are given in Tables 2, 3 and 4, for the cases where maximum values of internal pressure, fuel centerline temperature and maximum strain increment were obtained. It is shown the expected dependences with cycle length and maximum allowed LHGR. For example, maximum fuel centerline temperatures are dependent of the chosen LHGR, and similarly for the maximum fuel rod internal pressure. Burnups of 47.05 and 52 GWD/MTU were similar to the discharge burnup of some of light water reactor nuclear power plants. However, it is important to note on what nodes those maxima occurred.

Table 2 – Output summary, rod internal pressure, Experiments 1, 2 and 3.

Parameter	Experiment 1	Experiment 2	Experiment 3
Maximum fuel rod internal pressure	1023.37 psia	1177.91	1226.81
Rod average burnup	47.05 GWD/MTU	12.79	52.78
Fission gas release	1.04%	0.09	1.66
time	460 days	90	315

Table 3 – Output summary, centerline temperature, Experiments 1, 2 and 3.

Parameter	Experiment 1	Experiment 2	Experiment 3
Maximum fuel centerline temperature	1286.7°F	1602.86	1667.57
Axial node	9	3	3
Rod average burnup	15.43 GWD/TU	12.79	7.17
time	180 days	90	54

Table 4 – Output summary, strain increment, Experiments 1, 2 and 3.

Parameter	Experiment 1	Experiment 2	Experiment 3
Maximum strain increment	0.010334%	0.007057	0.008422
Axial node	10	15	16
Rod average burnup	38.02 GWD/MTU	39.06	37.05
time	380 days	243	225

To confirm the results of Figs. 13 to 15, Table 5 shows that more than the double of the mass of zircon oxide per unit area of the cladding´s surface was gained by the rod of Experiment 1, after 900 days of operation, when compared to the other 2 experiments, after 422 days. Also, it is confirmed that the mass per unit area incorporated by the rod of Experiment 3 is nearly 11% bigger than that of the Experiment 2, as discussed earlier in terms of cladding oxide thickness.

Table 5 - Output summary, oxide weight gain, Experiments 1, 2 and 3.

Parameter	Experiment 1	Experiment 2	Experiment 3
ZrO_2 weight gain	98.04 e-4 g /cm^2	43.27 e-4	48.09 e-4

4. CONCLUSIONS

FRAPCON was used in this work to simulate the performance of a nuclear fuel rod under stationary or conservative conditions. Sensitivity analysis was performed having as a parameter the cycle length. Three conditions of linear heat generation rate and times were used for the simulations. A conservative one, adapted to the features of the FRAPCON code, to run for long burnups, and the other 2, more severe in terms of the allowable LHGR, to simulate operational conditions for light water reactor nuclear power plants.

The results provided by the program, by means of curves of the mechanical properties like stresses, elongation, burnup, fuel and cladding temperatures, showed the expected dependences with time and, thus, with the cycles studied. An abnormal behavior of the cladding axial stress with time, for node 10, had no effect in the progress of the Experiment 3. But it still must be studied, in order to determine its causes.

Another set of simulations are planned to be carried out, in order to define the best input conditions for the transient analysis, from the conservative simulations. The goal is to perform some LOCA simulations, having FRAPCON as a source of inputs for the FRAPTRAN transient analysis.

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