# REACTIVITY INITIATED ACCIDENT ASSESSMENT FOR ATF CLADDING MATERIALS

C. GIOVEDI, M.R. MARTINS
LabRisco, University of São Paulo,
Av. Prof. Mello Moraes 2231, 05508-030, São Paulo, SP, Brazil,

A. ABE, R. REIS, A.T. SILVA
Nuclear Engineering Center — CEN, Nuclear and Energy Research Institute - IPEN/CNEN,
Av. Prof. Lineu Prestes 2242, 05508-000, São Paulo, SP, Brazil.

**Abstract**

Following the experience that came from the Fukushima Daiichi accident, one possible way of reducing risk in a nuclear power plant operation would be the replacement of the existing fuel rod cladding material (based on zirconium alloys) by other materials which could fulfill the requirements of the accident tolerant fuel (ATF) concept. In this sense, ATF should be able to keep the current fuel system performance under normal operation conditions; moreover, it should present superior performance than the existing conventional fuel system (zirconium based alloys and uranium dioxide) under accident conditions. The most challenging and bounding accident scenarios for nuclear fuel systems in Pressurized Water Reactors (PWR) are Loss of Coolant Accident (LOCA) and Reactivity Initiated Accident (RIA), which are postulated accidents. This work addresses the performance of ATF using iron based alloys as cladding material under RIA conditions. The evaluation is carried out using modified versions of the coupled system FRAPCON/FRAPTRAN. These codes were modified to include the material properties (thermal, mechanical, and physics) of an iron based alloy, specifically FeCrAl alloy. The analysis is performed using data available in the open literature related to experiments using conventional PWR fuel system (zirconium based alloys and uranium dioxide). The results obtained using the modified code versions are compared to those of the actual existing fuel system based on Zircaloy-4 cladding using the original versions of the fuel performance codes (FRAPCON/FRAPTRAN).

## 1. INTRODUCTION

Accident Tolerant Fuels (ATF) have been studied after the Fukushima Daiichi accident in order to improve the nuclear safety under steady state irradiation and accident scenarios. The most challenge scenarios to be studied are those related to postulated accidents, specifically LOCA and RIA. In this sense, in the framework of the ATF programme, it is necessary to evaluate the performance under postulated accident scenarios of different cladding materials which could be applied to replace the zirconium based alloys currently used. The cladding materials that present higher potentiality to this are iron based alloys such as iron-chromium-aluminum (FeCrAl), and ceramic materials such as silicon carbide (SiC).

There are two accident scenarios of particular interest considering inadvertent insertion of reactivity in nuclear power reactors: the control rod ejection accident in Pressurized Water Reactors (PWR), and the control rod drop accident in Boiling Water Reactors (BWR). These are design basis accidents, which may result in serious consequences if they were not accounted properly in the design of the reactor and related safety systems. The cause of accident is related to mechanical failure of a control rod mechanism housing and the primary coolant pressure ejects a control rod assembly completely out of the core. The main consequence of the control rod ejection is a rapid positive reactivity insertion, which shall occur within about 0.1 s in the worst possible scenario. The fuel rod behaviour during a RIA is mostly influenced by characteristics of the power pulse (amplitude and pulse width), coolant condition (pressure, temperature and flow rate), level of fuel burnup (degree of hydrogen pickup, cladding corrosion, internal fuel pressure), and fuel rod design (cladding thickness, pellet geometry and initial fill pressure). The rapid power excursion (0.1 s) leads to nearly adiabatic heating of the fuel pellets, which immediately deform by solid thermal expansion and depending of burnup degree, the amount of gaseous fission products retained in the fuel will contribute more to the solid pellet deformation. Due to pellet expansion the pellet cladding gap is reduced or closed leading to pellet cladding mechanical interaction (PCMI) with significant mechanical loading on the cladding tube [1].

Fuel performance codes are good tools to be applied to assess the behaviour of different materials under irradiation; however, the conventional fuel codes need to be modified to introduce the properties of the materials, which are being studied.

The FRAPTRAN code has two models to predict cladding failure, first failure model is related to RIA event, where deformation is due to pellet cladding mechanical interaction and at relatively low temperature (< 700 K), where PCMI is the driving force for cladding deformation, the model considers basically uniform plastic elongation from irradiated cladding as function of temperature and hydrogen concentration. The second failure model is applicable to LOCA events where deformation is relatively high due to gas overpressure and the temperature of the cladding (> 700 K) [2].

This paper presents some preliminary results obtained using modified versions of the FRAPCON-FRAPTRAN codes containing the properties related to the FeCrAl alloy to simulate the fuel behaviour under steady state and RIA conditions.

## 2. METHODOLOGY

### 2.1. FeCrAl alloy

FeCrAl alloy presents better properties compared to zirconium based alloys specifically concerning to the oxidation rates which are of 1 to 3 orders of magnitude lower [3]. Data from literature [3] based on computational simulations also indicate that FeCrAl alloy maintains acceptable thermo mechanical properties, and fuel-clad interactions under PWR conditions. The thermal, mechanical e physical properties of FeCrAl were obtained from literature [3] and used to modify the fuel performance codes.

### 2.2. Fuel performance code modification

The basis for the code modification applied to study the steady state irradiation was the FRAPCON-3.4 code [4], and for RIA evaluation the FRAPTRAN-1.5 [2]. The main subroutines related to the cladding in the codes modified to introduce the properties in MATPRO [5] of FeCrAl alloy were: CELMOD, CORROS, CREEPR, CSHEAR, and CTHCON. The material properties concerning to each of these subroutines are: CELMOD defines the correlation for the cladding Young´s modulus; CORROS is related to the cladding waterside corrosion; CREEPR; CSHEAR calculates shear modulus of cladding based on type and conditions; and CTHCON defines the correlation for the cladding thermal conductivity.

### 2.3. Test case

The test case applied to compare the fuel performance of Zircaloy-4 and FeCrAl alloy was the HBO-5, which data are available in the FRAPCON-3.5 assessment [6]. This case is part of a large scale RIA experiment carried out at NSSR reactor by JAERI (Japan Atomic Energy Research Institute). The NSRR facility has a TRIGA type reactor, which can generate significantly narrow power pulses, the RIA simulation tests on pre-irradiated fuel segment (test rod) employ a capsule with stagnant water. The HBO-5 fuel rod was refabricated from a standard 17×17 PWR fuel rod with uranium dioxide fuel pellet that was irradiated under steady state condition at OHI#1 reactor comprising a period of about 1360 days until an average burnup of 44 GWd/MTU. After this, the full-length fuel rod was refabricated and submitted to the test under RIA conditions. The HBO-5 fuel rod experienced failure at 0.2113 s.

The input for both codes were prepared following strictly according to the recommendations presented in the assessment volume as presented in Table 1.

TABLE 1. HBO-5 FUEL ROD MAIN CHARACTERISTICS AND BOUNDARY CONDITIONS

| Characteristic/boundary condition | PWR |
|---|---|
| Fuel material | $UO_2$ |
| Fuel pellet outer diameter (mm) | 8.05 |
| Fuel pellet height (mm) | 9 |
| Fuel density (% TD) | 95 |
| Fuel enrichment (w/o %) | 3.2 |
| Cladding outer diameter (mm) | 9.5 |
| Cladding inner diameter (mm) | 8.22 |
| Cladding wall thickness (mm) | 0.64 |
| Fuel-cladding diametral gap (µm) | 170 |
| Fill rod pressure (MPa) | 3.23 |
| Coolant temperature (°C) | 310 |
| Coolant pressure (MPa) | 15.5 |

The average power history and the rod average burnup as function of the irradiation time are presented in Figs 1 and 2, respectively.

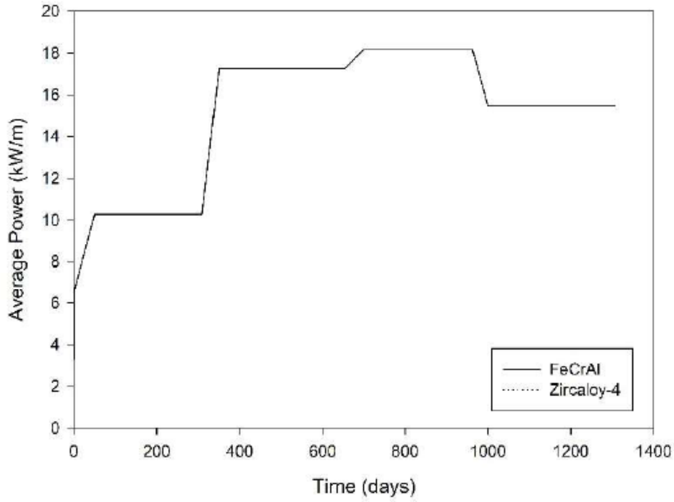


*FIG.1. Average power history for HBO-5 test case.*

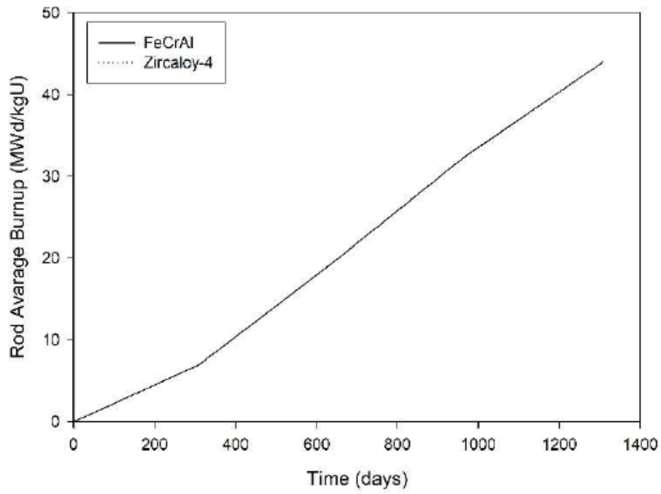


*FIG.2. Rod average burnup as function of irradiation time for HBO-5 test case.*

## 3. RESULTS AND DISCUSSION

The main results obtained comparing the performance of Zircaloy-4 and FeCrAl alloy as cladding under steady state irradiation for the HBO-5 fuel rod are presented in Figures 3 to 7.

Figure 3 shows that FeCrAl fuel rod presents fuel centreline temperatures higher than Zircaloy-4 fuel rod. This occurs due to the higher thermal expansion of the FeCrAl. Then, the gap thickness is larger, as shown in Figure 4, and, consequently, the fuel temperature reaches higher values.

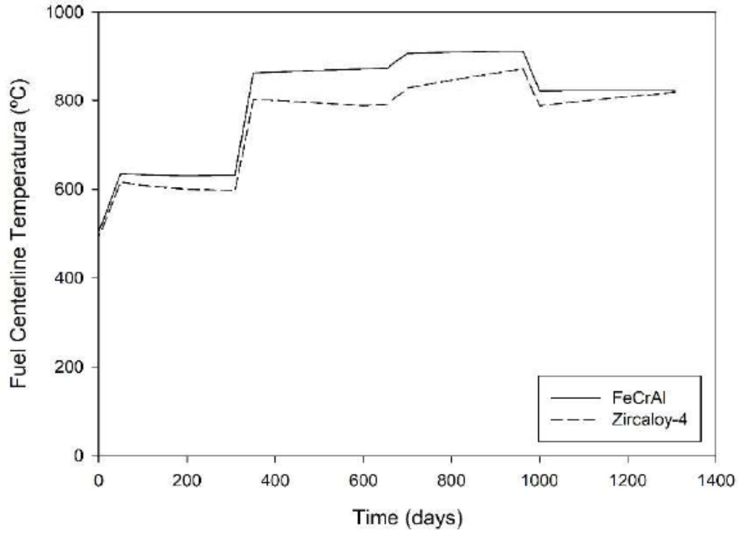


*FIG.3. Fuel centerline temperature evolution under steady state irradiation as function of time for HBO-5 test case considering as cladding: Zircaloy-4, and FeCrAl alloy.*

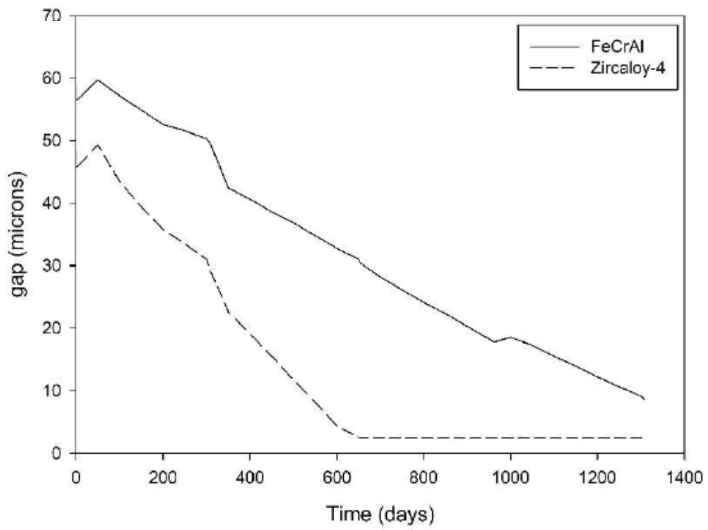


*FIG.4. Gap thickness evolution under steady state irradiation as function of time for HBO-5 test case considering as cladding: Zircaloy-4, and FeCrAl alloy.*

Also, it can be observed in Figure 5 that FeCrAl fuel rod presents a lower plenum pressure compared to the Zircaloy-4 fuel rod due to the higher internal free volume available that is a result of the higher thermal expansion of the FeCrAl alloy.

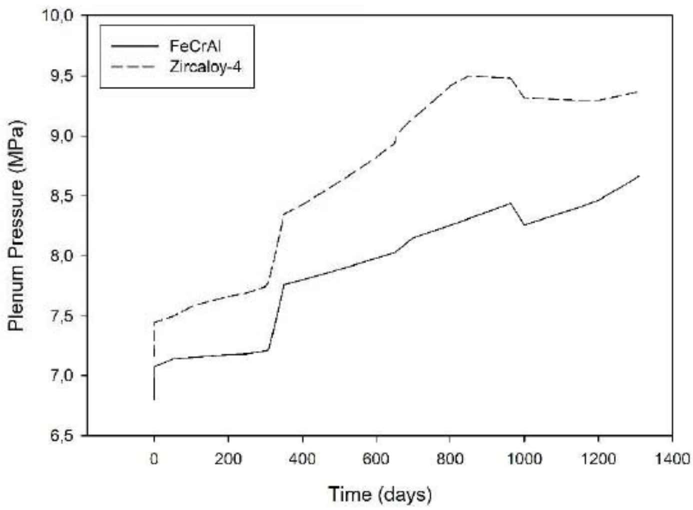


*FIG.5. Plenum pressure evolution under steady state irradiation as function of time for HBO-5 test case considering as cladding: Zircaloy-4, and FeCrAl alloy.*

Due to the gap closure around 600 irradiation days observed for the Zircaloy-4 fuel rod, the cladding hoop stress, as shown in Fig. 6, changes from a compressive to a tensile state. This is not observed for the FeCrAl fuel rod because the gap remains open during all the irradiation time, also, as a consequence of the higher thermal expansion compared to Zircaloy-4.

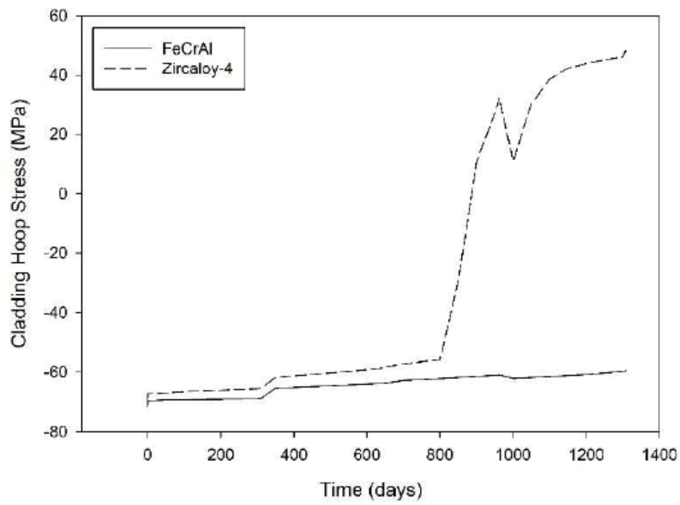


*FIG.6. Cladding hoop stress evolution under steady state irradiation as function of time for HBO-5 test case considering as cladding: Zircaloy-4, and FeCrAl alloy.*

Regarding to the fission gas release, Figure 5 shows that the behaviour is exactly the same for both studied materials. This shows that the differences observed in the fuel centerline temperatures are not enough to affect the fission gas release behaviour for both studied materials.

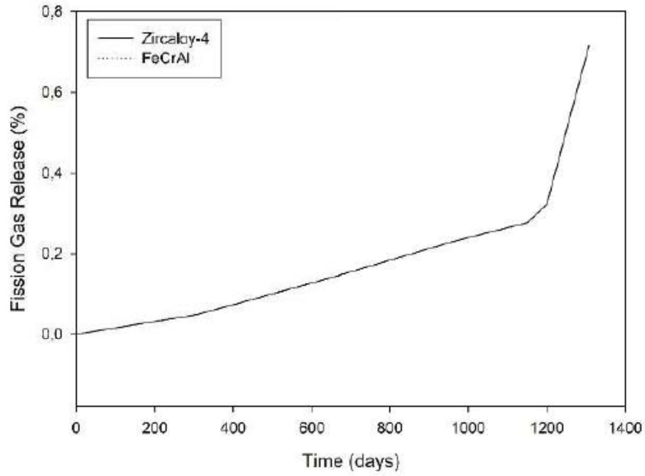


*FIG.7. Fission gas release evolution under steady state irradiation as function of time for HBO-5 test case considering as cladding: Zircaloy-4, and FeCrAl alloy.*

After steady state irradiation, the RIA condition was simulated using FRAPTRAN code in order to reproduce the experimental test for the HBO-5 fuel rod. Figure 8 presents the cladding hoop stress after RIA, considering that the RIA starts at time equal to 0. It can be observed that the Zircaloy-4 fuel rod experiences failure 0.21 s after starting the RIA, result that is very similar to that observed in the experiment. This confirms that the simulation was appropriately carried out. The same simulation was performed under RIA conditions using the transient code modified considering FeCrAl as cladding material. And, in these preliminary results the FeCrAl fuel rod experiences failure at a time very close to that observed for Zircaloy-4 due to the fact that the failure criterion adopted in the FRAPTRAN code for Zircaloy-4 cannot be appropriate to evaluate the FeCrAl fuel rod behaviour under RIA conditions.

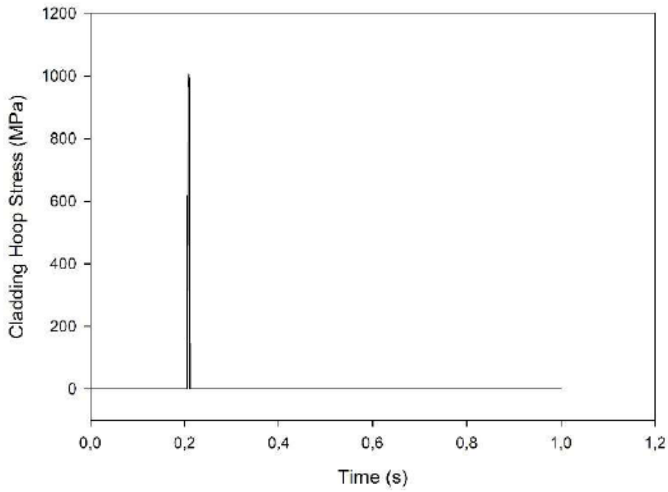


*FIG.8. Cladding hoop stress after RIA (t=0) for HBO-5 test case.*

## 4. CONCLUSIONS

The obtained results for the HBO-5 test case indicate that under steady state irradiation FeCrAl present a good performance compared to Zircaloy-4 showing: higher fuel temperatures (about 100°C), lower internal pressure, open gap during all the irradiation time, and lower cladding hoop stress.

The simulations carried out to under the RIA scenario reproduced exactly the experimental result registered for the studied test case, occurring the failure of the fuel rod at the same time verified in the experiment.

Although the results of simulation obtained in the steady state condition have shown differences, mainly in the fuel gap parameter, the fuel rod test case failure at similar time in the transient simulation, consequently it is necessary to evaluate and establish a better and consistent failure criterion for FeCrAl cladding.

The strain-based failure criterion used in the original code version for Zircaloy-4 considers the effects of hydride content and temperature [7]. In this sense, it is necessary to carry out experiments in order to obtain data to better predict the cladding failure for FeCrAl behaviour under RIA conditions.

## 5. NOMENCLATURES

| | |
|---|---|
| ATF | Accident Tolerant Fuel |
| BWR | Boiling Water Reactor |
| FeCrAl | Iron-Chromium-Aluminum alloy |
| HBO-5 | Experimental test case |
| LOCA | Loss-of-Coolant Accident |
| PCMI | Pellet Cladding Mechanical Interaction |
| PWR | Pressurized Water Reactor |
| RIA | Reactivity Initiated Accident |
| SiC | Silicon Carbide |

## ACKNOWLEDGEMENTS

The authors are grateful to the technical support of USP and IPEN-CNEN/SP, and to the financial support of IAEA to attend the Technical Meeting.

## REFERENCES

[1] RASHID, Y.R., ZANGARI, A.J., LIN, C.L., Modeling of PCI Under Steady State and Transient Operating Conditions, IAEA Technical Committee Meeting on Water Reactor Fuel Element Computer Modeling in Steady State, Transient and Accident Conditions, IAEA-659/IWGFPT/32 Preston England, (1988) 91.

[2] GEELHOOD, K.J., LUSCHER, W.G., CUTA, J.M., FRAPTRAN- 1.5: A Computer Code for the Transient Analysis of Oxide Fuel Rods, PNNL-19400, 1, Rev.1 (2014).

[3] GAMBLE, K.A., BARANI, T., PIZZOCRI, D., HALES, J.D., TERRANI, K.A., PASTORE, G., An Investigation of FeCrAl Cladding Behavior Under Normal Operating and Loss of Coolant Conditions, Journal of Nuclear Materials **491** (2017) 55.

[4] GEELHOOD, K.J., LUSCHER, W.G., BEYER, C.E., FLANAGAN, M.E., FRAPCON-3.4: A Computer Code for the Calculation of Steady state Thermal mechanical Behavior of Oxide Fuel Rods for High Burnup, U.S.NRC, NUREG/CR-7022, Washington (2011).

[5] ALLISON, C.M., BERNA, G.A., CHAMBERS, R., CORYELL, E.W., DAVIS, K.L., HAGRMAN, D.L., et al., SCDAP/RELAP5/MOD3.1 Code Manual Volume IV: MATPRO—A Library of Materials Properties for Light-Water-Reactor Accident Analysis, NUREG/CR-6150, EGG-2720, Washington (1993).

[6] GEELHOOD, K.J., LUSCHER, W.G., FRAPCON-3.5: Integral Assessment, PNNL-19418, 2, Rev.1 (2014).

[7] NEA/CSNI/R (2010) 1, Nuclear Fuel Behaviour under Reactive Initiated Accident (RIA) Conditions, OCDE (2010) 179.