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FUEL PERFORMANCE OF IRON-BASED ALLOY CLADDING USING MODIFIED TRANSURANUS CODE

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ABSTRACT

The main challenge in the nuclear area since the Fukushima Daiichi accident is to develop fuel materials to be applied in nuclear reactors aiming to increase the safety under normal operation as well as transient and accident conditions. These efforts are concentrated in the Advanced Technology Fuel (ATF) program that has as main scopes to study cladding materials to replace the zirconium-based alloys, and fuel materials presenting higher thermal conductivity compared to the conventional uranium dioxide fuel pellet. In this sense, iron-based alloys, which were used with a good performance as cladding material in the first Pressurized Water Reactors (PWR), have becoming a good option. The assessment of the behavior of different materials previously to perform irradiation tests, which are time consuming, can be performed using fuel performance codes, but for this, the conventional fuel performance codes must be modified to implement the properties of the materials that are being studied. This paper presents the results obtained using a modified version of the well-known TRANSURANUS code, obtained from the implementation of the stainless steel 348 properties as cladding material. The simulations were performed using data available in the open literature related to a PWR irradiation experiment. The results obtained using the modified version of the code were compared to those obtained using the original code version for zircaloy-4. The performance of both cladding materials was evaluated by means of the comparison of parameters such as gap thickness, fuel centerline temperature, internal pressure, and cladding stress and strain.

1. INTRODUCTION

After the Fukushima Daiichi accident on March 2011 [1] the main challenge in the nuclear area is to develop nuclear fuels that can reduce the generation of hydrogen and fuel melting that characterized this accident. The main assumption is to increase the amount of time before operator action that is necessary in the event of a loss of cooling [2]. These activities involve research institutes, universities, and industry in a collaborative work.

In the framework of the Advanced Technology Fuel (ATF) program, the main issues are to develop new cladding materials and to improve the fuel thermal conductivity. Concerning to the cladding, iron-based alloys represent a good option to replace the conventional zirconium-based alloys mainly due to the low cost and good thermal and mechanical properties; then, materials such as stainless steel and iron-chromium-aluminum (FeCrAl) alloy [3] are being evaluated as possible cladding candidates in ATF. Regarding to the fuel, efforts are being made in order to evaluate possible materials to be used as addictive in the conventional uranium dioxide fuel to improve the thermal conductivity under irradiation [4, 5].

The complete development of ATF requires carrying out experimental tests with these new materials, to apply computational simulation, and to perform in-pile tests. Considering that irradiation tests are time consuming, the use of appropriate computational tools can reduce the way to assess new cladding materials. In this sense, fuel performance codes play an important role to evaluate the behavior of new materials under different irradiation scenarios. However, taking into account that the conventional fuel performance codes available just consider zirconium-based alloys as cladding material, the first step to use these tools is the code modification in order to introduce the properties of the material to be studied allowing to evaluate its performance under irradiation.

The aim of this work is to present preliminary results obtained using a modified version of the well-known TRANSURANUS code considering stainless steel (AISI) 348 as cladding material and to compare its performance to zircaloy-4 using the original code version using data available in open literature for a fuel rod under steady-state irradiation.

2. METHODOLOGY

The code, the modification implemented in the code, and the test case used for simulation are described below.

2.1. TRANSURANUS

The basis for the code modification was the well-known TRANSURANUS code developed at the European Institute for Transuranium Elements (ITU). This is a computer code for the thermal and mechanical analysis of fuel rods in nuclear reactors [6].

The properties of the AISI 348 were collected in the literature aiming to satisfy the set of physical, thermal, and mechanical data which were necessary to be introduced in the TRANSURANUS code. The data selection has been made in order to use reliable data, when necessary data are not available either values coming from similar stainless steel (AISI 347) or typical values (i.e. applicable for a variety of stainless steel) were used [7, 8].

Considering that TRANSURANUS code contains in its library properties and models for AISI 316, the modification process carried out for AISI348 considered that the correlations already programmed in TRANSURANUS for AISI 316 are acceptable and validated enough. In this sense, the adaptation for AISI 348 aimed to keep the same structure available in original code version for AISI 316 in order to avoid possible incoherencies.

2.2. Fuel Performance Code Modification

The properties related to the cladding modified in the TRANSURANUS code to introduce the properties of AISI 348 are: thermal conductivity, specific heat, elasticity constant, Poisson´s ratio, strain due to swelling, thermal strain, creep strain, yield stress, rupture strain, burst stress, density, and melting temperature.

2.3. Test Case

The assessment was carried out using experimental data available in the open literature [9] related to a zircaloy/UO_2 fuel rod identified as TSQ002. This fuel rod was part of a program to develop more efficient fuel management concepts and an increase in the burnup of discharged fuel.

The fuel rod TSQ002 was part of a standard 16 x 16 Pressurized Water Reactor (PWR) fuel assembly. The main characteristics of TSQ002 fuel rod are presented in Table 1. It accumulated an end-of-life (EOL) rod-average burnup of 56.1 GWd/MTU. The rod-average LHGR varied from 2.75 to 6.95 kW/ft with the higher values near beginning-of-life (BOL).

Table 1: Characteristics of TSQ002 Test Fuel Rod

Parameter	Value
Rod Outside Diameter (cm)	0.97028
Cladding Thickness (cm)	0.0635
Gap size (diametral) (cm)	0.00889
Fuel Rod Pitch (cm)	1.27
Fuel Pellet Diameter (cm)	0.8255
Fuel Pellet Density (%)	95
Fill Gas Internal Rod Pressure (MPa)	2.62
Active Fuel Length (cm)	381
U235 Enrichment Degree (%)	3.48
Coolant flow rate (g h-1)	3.6E+6
Coolant temperature (°C)	290 (inlet)
Coolant Pressure (MPa)	15.5
Average LHGR (kW m^{-1})	22.802 (time peak) 15.533 (time ave.)
Maximum Cladding Temperature Surface (°C)	324.93 (Zircaloy) 320.89 (AISI 348)
Average Burnup (MWd tU^{-1})	9.47
Final Fluence Level (n cm^{-2})	1.05E+22 (ave., fast) 1.12E+22 (max., fast)

3. RESULTS AND DISCUSSION

The figure of merits related to gap thickness, fuel centerline temperature, internal pressure, hoop stress, and hoop strain evolution are presented in Figures 1 to 5, respectively.

Figure 1 shows that the gap thickness is higher for AISI 348 due to the higher thermal expansion experienced by AISI 348 under irradiation. As a consequence of this, zircaloy fuel rod presents gap closure after about 27000 while for the AISI 348 fuel rod the gap remains open during all the irradiation time at the studied conditions.

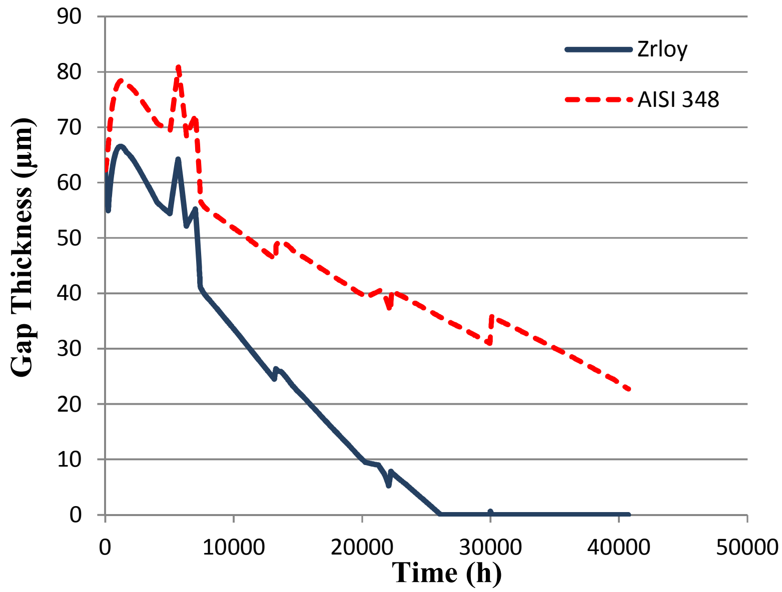

Figure 1: Gap thickness evolution as function of irradiation time using: original TRANSURANUS code for zircaloy-4 and modified version for AISI 348.

As a consequence of the higher gap thickness observed for the AISI 348, the fuel centerline temperatures are about 100ºC higher than those observed for the zircaloy fuel rod as can be observed in Figure 2.

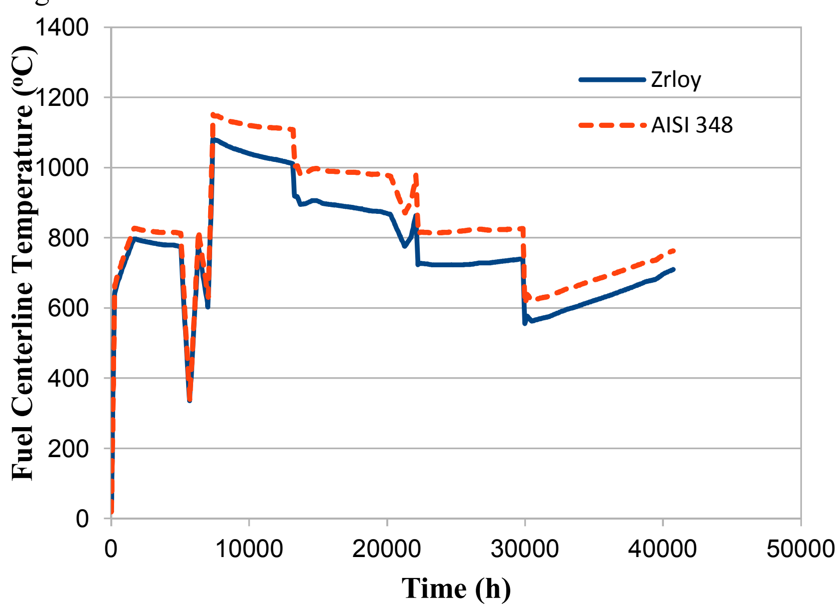

Figure 2: Fuel centerline temperature evolution as function of irradiation time using: original TRANSURANUS code for zircaloy-4 and modified version for AISI 348.

The higher thermal expansion observed for AISI 348 fuel rod compared to the zircaloy fuel rod also affects the internal pressure evolution. Then, AISI 348 experiences a slightly lower internal pressure compared to the zircaloy fuel rod, as presented in Figure 3.

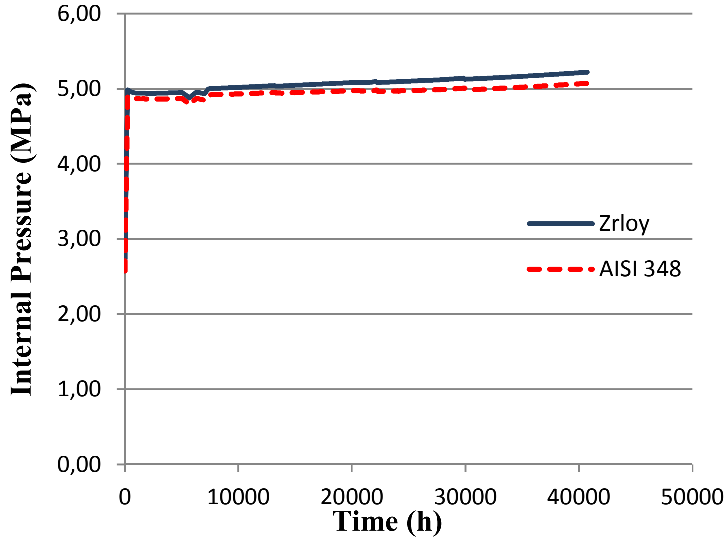

Figure 3: Internal pressure evolution as function of irradiation time using: original TRANSURANUS code for zircaloy-4 and modified version for AISI 348.

Due to the fact that the gap remains open during all the irradiation time, it is not observed any change in the mechanical loading for the AISI 348 fuel rod; on the other hand, for zircaloy, it is observed changes in the stress and strain of the cladding, which are associated to the gap closure, as can be observed in Figures 4 and 5.

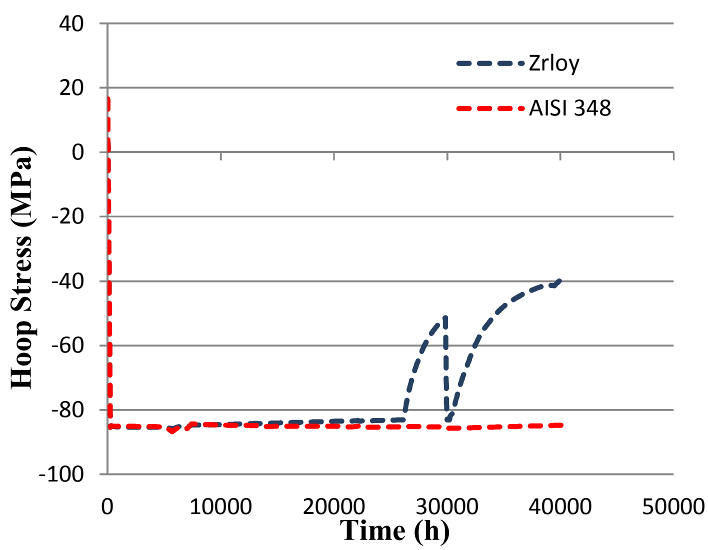

Figure 4: Hoop stress evolution as function of irradiation time using: original TRANSURANUS code for zircaloy-4 and modified version for AISI 348.

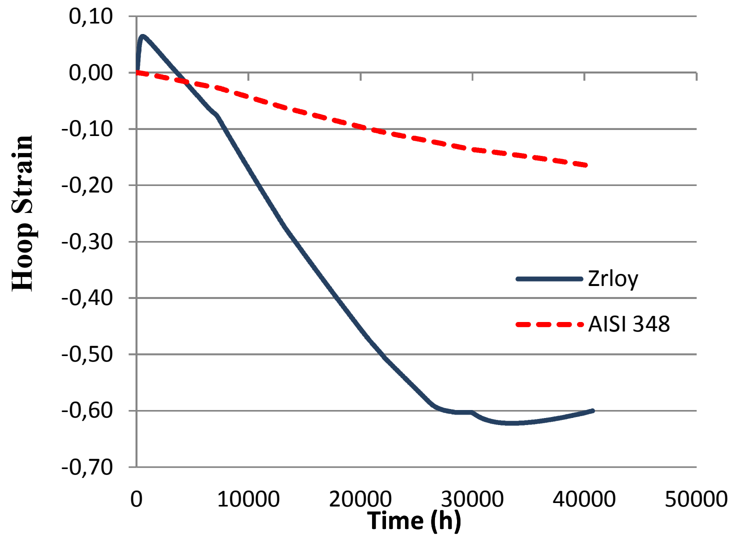

Figure 5: Hoop strain evolution as function of irradiation time using: original TRANSURANUS code for zircaloy-4 and modified version for AISI 348.

4. CONCLUSIONS

The preliminary results obtained in the simulations performed using the TRANSURANUS modified code version for AISI 348 compared to those obtained using the original code version have shown that the parameters evaluated are in agreement with the properties of both material. Then, the AISI 348 fuel rod presents higher gap thickness during all the irradiation time with no gap closure. As a consequence of this, the fuel centerline temperatures observed for AISI 348 are about 100^{o}C higher compared to the zircaloy fuel rod. Concerning to the internal pressure, despite of the higher fuel centerline temperatures experienced by AISI 348, its higher thermal expansion compared to zircaloy results in lower internal pressure. The gap closure experienced by zircaloy fuel rod induces the change in the hoop stress as well as in the hoop strain values which can lead to the pellet cladding mechanical interaction (PCMI) at high burnups.

These preliminary results obtained using the TRANSURANUS modified code version shall be verified also at different irradiation scenarios even considering accidents as Los-of-Coolant Accident (LOCA) and Reactivity Insertion Accident (RIA), which are essential in the assessment of different materials to replace the convention zirconium-based alloys as cladding material in the framework of ATF program.

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REFERENCES

1. N. Akiyama, H. Sato, K. Naito, Y. Naoi and T. Katsuta, "The Fukushima Nuclear Accident and Crisis Management-Lessons for Japan-U.S. Alliance Cooperation", Sasakawa Peace Foundation, Tokyo (2012).
2. "Advanced Technology Fuels – Potential Nuclear Industry Game Chang", https://nextevolutionfuel.com/2017/03/advanced-technology-fuels-potential-nuclear-industry-game-changer (2019).
3. K. A. Terrani, S. J. Zinkle and L. L. Snead, "Advanced oxidation-resistant iron-based alloys for LWR fuel cladding", *Journal of Nuclear Materials*, 448, pp.420-435 (2014).
4. R. Liu, W. Zhou, P. Shen, A. Prudil and P. K. Chan, "Fully Coupled Multiphysics Modeling of Enhanced Thermal Conductivity UO_2–BeO Fuel Performance in a Light Water Reactor", *Nuclear Engineering and Design*, **295**, pp.511 (2015).
5. K. Y. Spencer, L. Sudderth, R. A. Brito, J. A. Evans, C. S. Hart, A. Hu, A. Jati, K. Stern and S. M. Mcdeavitt, "Sensitivity Study for Accident Tolerant Fuels: Property Comparisons and Behavior Simulations in a Simplified PWR to Enable ATF Development and Design", *Nuclear Engineering and Design*, **309**, pp.197 (2016).
6. K. Lassman, "TRANSURANUS: a fuel rod analysis code ready for use", *Journal of Nuclear Materials*, **188**, pp.295-302 (1992).
7. A. Abe, C. Giovedi, D.S. Gomes, A. Teixeira e Silva, "Revisiting stainless steel as PWR fuel rod cladding after Fukushima Daiichi accident", J. Energy Power Eng., **8**, 973 (2014).
8. C. Giovedi, M. Cherubini, A. Abe and F. D´Auria, "Assessment of stainless steel 348 fuel rod performance against literature available data using TRANSURANUS code", EPJ Nuclear Sci. Technol., **2**, pp.27 (2016).
9. K. J. Geelhood, W. G. Luscher, C. E. Beyer and M. E. Flanagan, "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).