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SIMULATION OF THE EFFECTS OF THE EXTEND FUEL ROD BURN-UP UNDER LOCA SCENARIO

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ABSTRACT

Due to the high burn-up imposed to the nuclear fuel in the last recent years, new challenges become important, including a deep review of the fuel performance under accident conditions. In this sense, available data in the open literature show that some experiments were carried out in order to study the behavior of fuel rods under LOCA (Loss of Coolant Accident) scenario. For instance, a series of experiments, designated IFA-650 series, performed in the Halden reactor in 2010 present data related to zircaloy fuel rods submitted to LOCA conditions. In the tests were addressed issues such as fuel fragmentation, relocation and dispersal for an extended irradiation cycle. In the studied case (IFA-650.5), the LOCA scenario was evaluated after a burnup of 83.4 MWd/kg. The aim of this paper is to compare the experimental data to the fuel performance obtained applying the codes FRAPCON and FRAPTRAN. Different phenomena were evaluated, such as ballooning, burst, cladding oxidation and fuel relocation. Also, the cladding metallurgical phase transformation was considered. The obtained results reproduced in a good way the experimental data, showing that the adopted models are representative of the observed phenomena.

1. INTRODUCTION

The evaluation of fuel performance under accident condition is an essential challenge of safety analysis in nuclear area. In order to perform this kind of assessment, there are computational codes that enable to simulate the irradiation conditions. The developing of these codes was a result of a research program conducted by U.S. Nuclear Regulatory Commission (USNRC). The performance codes contain models capable to describe the nuclear behavior in normal and accident conditions. The code FRAPTRAN-1.5 allows simulating transient conditions, such as observed during loss-of-coolant accident (LOCA) scenarios [1]. Before starting the transient, the steady-state irradiation must be simulated using the code FRAPCON-1.5 to provide the fuel rod parameters at the transient step. In order to validate these tools, it is important to carry out experiments in the same operational conditions. The aim of this investigation is to provide guidelines to simulate fuel rod experiments under transient, such as that from LOCA benchmark obtained from experiments carried out in the Halden reactor in 2006. Some of these data are available in the open literature [2].

In these experiments, segments of fuel rods, previously irradiated at steady state, were then submitted to LOCA conditions. The current fuels have high burnup and linear heat rate. The main goal of the performed tests was to investigate irradiations effects as ballooning, fragmentation, relocation, rupture, oxidation, hydride and pellet and fuel stack during an LOCA transient. A specific concern related to LOCA is that the fuel fragments fill up the ballooned volume promoting the increase of the linear power.

The data used in the present paper are from the experiment IFA-650.5 that corresponds to a Pressurized Water Reactor (PWR), with high burnup (83 MWd/kgU) [3]. The cladding is Zircaloy-4, with heat treatment stress relief annealed (SRA). In the investigation, to calculate IFA-650-5 behavior under the transient condition, it was applied the FRAPTRAN-1.5 and the experimental data available were compared to results obtained in the code simulation. The FRAPCON-3.5 was used to obtain data related to the steady state condition. The obtained results are discussed based on the models and input data utilized in different simulations [4].

2. BACKGROUND

2.1 Acceptance Criteria for Loss-of-coolant Accident

The safety rule applied in worldwide for Light Water Reactor (LWR) is called acceptance criteria of emergency core cooling systems (ECCS) [5]. The USA Code of Federal Regulation (CFR) 10 CFR 50.46 was initially published in 1974. The current norm has important revisions, with amended in 1988, 1996, 1997, and 2007. The document 10CFR 50.46 contains the guidelines of the regulatory norm acceptable using the best-estimate method adopted in 1988.

The calculated maximum fuel element cladding temperature shall not exceed 1204°C. The entire determined oxidation of the cladding shall not exceed 0.17 times the absolute cladding thickness before oxidation (17%) [6]. The total amount of hydrogen generated from the chemical reaction of the cladding with water or steam shall not exceed 0.01 times the hypothetical amount that would be produced if all the metal in the cladding surrounding the fuel reacts. Calculated changes in core geometry shall be such that the core keeps partially the fuel coolability. After any calculated successful initial operation of the ECCS, the estimated core temperature shall be maintained at an acceptably low value, and decay heat shall be removed for the extended period required.

2.1.1 Loss of coolant accident

The LOCA is an accident caused by a break in the pipe, producing a leaking and reducing the cooling capacity. The hot pipe transports the heated coolant (hot leg) from the pressure vessel to the steam generator and then back to the reactor pressure vessel by cold leg [8]. In PWR, the primary circuit is designed for a pressure of 15 MPa and temperature of 300°C in the Reactor Pressure Vessel (RPV). The core of a PWR is a pressurized vessel with a cylindrical base session, called bottom head, and a hemispherical top head easily removable. Opening the cover of the core allow the refueling of the reactor and perform the change of the fuel. The primary circuit must remove heat from the core reactor. The coolant flows into hot legs to the steam generator. The rupture of hot or cold leg withdraws the cooling capacity of the core for a significant period before beginning the stage of the emergency cooling system [11]. Breaks with flow areas typically less than 0.093 m^2 and greater than 9.5 mm diameter span the category of small breaks (SB-LOCA), since breaks smaller than 9.5 mm in diameter do not depressurize the reactor coolant system. The position of the breakdown in the pipe is important to classify the severity of the event. The small break occurs in the cold leg, but if the rupture is large or in the hot leg, it can be classified as large break (LB-LOCA) [12]. Small break LOCA is about 100 times more probable that large break. The models applied for LOCA must predict pressure decrease, steam formation and separation of steam from water. The especial fluid dynamics also must calculate parameters as coolant speed, direction, pressure and temperature.

The LOCA scenario considers an overheating within of core. The performance of the zirconium alloy during the transient indicates the existence of a broad range of theoretical models, depending on reactor situation. The evolution of the accident will result in internal overheat and consequent core damaged, perhaps even leading to a meltdown of the core.

2.1.2 Accident phases

The blowdown is a term used to describe the rapid loss of coolant from the primary system. The period starts when the coolant leak of the main system through the break. In the blowdown, the coolant loses pressure into the reactor vessel at a pressure below at saturation point [9]. The decreasing of pressure has a low rate due to the creation of a two-phase flow, composed of a steam-water mixture. The loss of pressure forces the phase transition that can evaluate for an explosion due to the coolant change phase from liquid to vapor. The constant loss of coolant from the pressure vessel starts the core dry out downward, then the top region dry before the small session of the core.

The refilling stage occurs when the water from the safety systems begins to refill the lower parts of the pressure vessel. After of open the ECCS, occurs an upward vaporstate flow, between the inlet pipes and outlet flow or hot leg [10]. Around the top of the reactor, it retains a considerable amount of steam. The tendency is that at the top of the reactor to accumulate steam and at the bottom, water.

The reflood phase occurs with the reestablishment of core cooling. The long-term cooling is the stage that the cooling operation is continuing. The reflood phase is most important to avoid mechanical deformation and blockage. In reflood period, occur a slow filling of the hot core under gravity forces, against the resistance provided by steam produced, which must vent to the break. The fuel rod cladding may swell or balloon and partially block some of the coolant sub channels. The blockages affect the heat removal process [13]. During fuel burst, strains occur at the same axial elevation, promoting a distortion in the fuel assembly.

2.1.3 High-Temperature Oxidation Predictions

The Cathcart-Pawel (CP) correlations can describe the oxidation as a function of temperature. The CP model was used for IFA-650.5 as recommended in the FRAPTRAN 1.5. The code has two options to treat the existing oxide, as protective or non-protective. FRAPTRAN-1.5 can calculate internal and external CP for cladding. The correlation called equivalent cladding reacted (ECR) [7], also can offer an upper bound for corrosion, limited at 17%. If the oxidation result is higher than the safety threshold, the material walks to the fragile region. NRC has proposed the 17% $ECR_{C\text{-}P}$ oxidation criterion adjustment that consists in the reduction of tube thickness, calculated according to CP correlation for LOCA temperature course considering the tube thickness that oxidized during corrosion in the reactor. These correlations are presented in Equations 1 and 2.

$$W_{CP} = 0.0602 t^{1/2} \exp(-10050/T) \tag{1}$$

Where W is weight gain in grams per square centimeter of surface area and T is a temperature in Kelvin, and t is time in second.

$$ECR = 87.8\left[\frac{w}{h}\right] \tag{2}$$

Where ECR is in percent, h is the cladding thickness in centimeter, and d is cladding outside diameter in centimeter.

2.1.4 Fuel performance code

The code FRAPTRAN-1.5 is the successor to the T-FRAP version, first developed from 1970 to 1980. The FRAPTRAN-1.5 can predict the behavior of the fuel and cladding interactions resulting from the transient state. The time steps are of milliseconds for the transient scenarios.

The methodology applied by coupled codes is based in models as thermal-hydraulic, mechanical, fission gas release, and oxidation. These codes must estimate the thermal evolution, such as heat transfer rate and thermal expansion. FRAPCON-3.5 produces the initialization file for FRAPTRAN-1.5 perform transient calculations. The validation of FRAPCON-3.5 is formed by 133 fuel rods and FRAPTRAN-1.5 based in 43 integral assessment cases.

3. MATERIAL AND METHODS

The simulation data were taken from the open literature to the IFA-650.5 case, a fuel rod pre-irradiated in a commercial PWR 15x15 up to 83 MWd/kgU using Zircaloy as cladding. The LOCA accident was performed at Halden Boiling Heavy-Water Reactor (HBWR). The fuel rod showed a short active length of 0.5 m and it was filled with a gas mixture, formed of 95% Argon and 5 % Helium. The basic idea was to simulate the fission gases released during the steady-state irradiation.

3.1 Analysis of the IFA-650.5

The experimental series conducted in Halden focuses on in-reactor effects, which are different from those obtained in out-of-reactor tests. In special, the process of heat transfer, from fuel pellet to cladding, in disagreement with the external heating models performed out-of-pile. An overview of fuel parameters applied in IFA-650.5 tests is given in Table 1.

Table 1: Geometric properties of fuel rod, IFA-650.5

Fuel type	PWR
Fuel material	UO_2
Fuel grain size (μm)	10
Fuel pellet diameter (mm)	9.132
Fuel pellet length (mm)	11
Fuel dish depth (mm)	0.28
Fuel land width (mm)	1.2
Fuel density (% TD)	94.8
Fuel burnup cycles	6
Fuel enrichment (w/o %)	3.5

The properties of fuel cladding parameters used in IFA-650.5 tests is given in Table 2.

Table 2: Rod fuel properties of IFA-650.5 test rod fuel data.

Cladding material	Zircaloy-4
Cladding heat treatment	SRA
Cladding outer diameter (mm)	10.735
Cladding wall thickness (mm)	0.721
Cladding outer oxide layer (μm)	(65-80)
Cladding hydrogen content (ppm)	650
Fuel rod Burnup (MWd/kgU)	83.4
Fuel rod total length (mm)	480
Fuel rod gap (mm)	0.0805
Fuel rod plenum volume (cm^3)	15
Fuel rod fill gas	90% Ar +10%He
Fill pressure (MPa)	4.0
Fuel rod target PCT (°C)	1100

Considering the adopted models, the level of geometric distortion for fuel cladding was given by FRACAS–I (rigid pellet model). The model BALON2 was applied for failure with empirical stress limits. The gas model applied to estimate the fission gas release was the Massih model. The correlation for high-temperature oxidation used was the CP model.

After initiations of the blowdown phase, in IFA-650.5 are 800 s to the end. An electrically heated flow separator tube surrounds the fuel rod. Exit channels for the coolant circulation from upper to lower are present. The test apparatus provides an electric heating system for fuel rods. The temperature controls offer a separator for heat up of the fuel also using a water spray, which cool the fuel rod with water spraying. The time spent to reach the rupture is detected by dropping in the rod pressure. In sequence, occur in the laboratory an exam detailed of post-irradiation after withdrawing the fuel of the flask. The failure size verified is a narrow rupture of 10 mm reported in the visual examination. The localization of rupture was in the lower region of the rod.

4 RESULTS

The simulation has two stages with the first one carrying out the simulation of the steady state irradiation. In this first step, it is calculated the burning cycle in a commercial reactor, after this stage the rods are withdrawn and remanufactured for the transient. In the second step, it is simulated the transient. The results achieved through simulation with FRAPCON-3.5 are very close of the obtained in the experimental phase. The quantities of hydrogen buildup showed a small difference with the simulation. The simulation presents 488 ppm while the experiment value was of 650 ppm. Table 3 presents results of IFA-650.5 under burnup cycle simulated with FRAPCON-3.5.

Table 3: Simulation results from IFA-650.5 with FRAPCON-3.5 code.

IFA-650.5 Burnup cycle	Results
Cycle of burnup (days)	2064
Burnup (MWd/kgU)	83.47
Thickness layer ZrO_2 (μm)	73.9
Hydrogen (ppm)	488
Fission gas cumulative fraction release (%)	14
ZrO_2 weight gain ,(g/m^2)	8.62
Maximum fuel rod internal pressure (MPa)	9.09
Maximum fuel centerline temperature (°C)	1547
Maximum strain increment (elastic and plastic) (%)	0.074364

During the transient phase, when starting the IFA-650.5 test, during blowdown phase the coolant pressure falls and the coolant mass flow is reduced. The stage of depressurization produces an increasing of the temperature of the cladding from 350 °C to 850 °C. The overheating can reach the region of the transition of zirconium, (alpha+beta) between 840 to 980 °C, in about 40 s. The period of depressurization and overheating must start the emergency cooling system to supply coolant, but present a continued rise of temperature. The pellet cladding temperature was 1100 °C, and the failure occurred at 750 °C. The distribution of fragment can fall to the lower part of the fuel bundle, but the uranium dioxide cracked increases the fission rate. The event produces fragmentation and relocation in small scale. The geometric deformations, as ballooning, contribute for relocation. The hard contact was achieved during short periods of the irradiation time, so occurs the pellet clad interaction as shown in Figure 1. Figure 2 also indicates LOCA phase and rupture of cladding at 179 s.

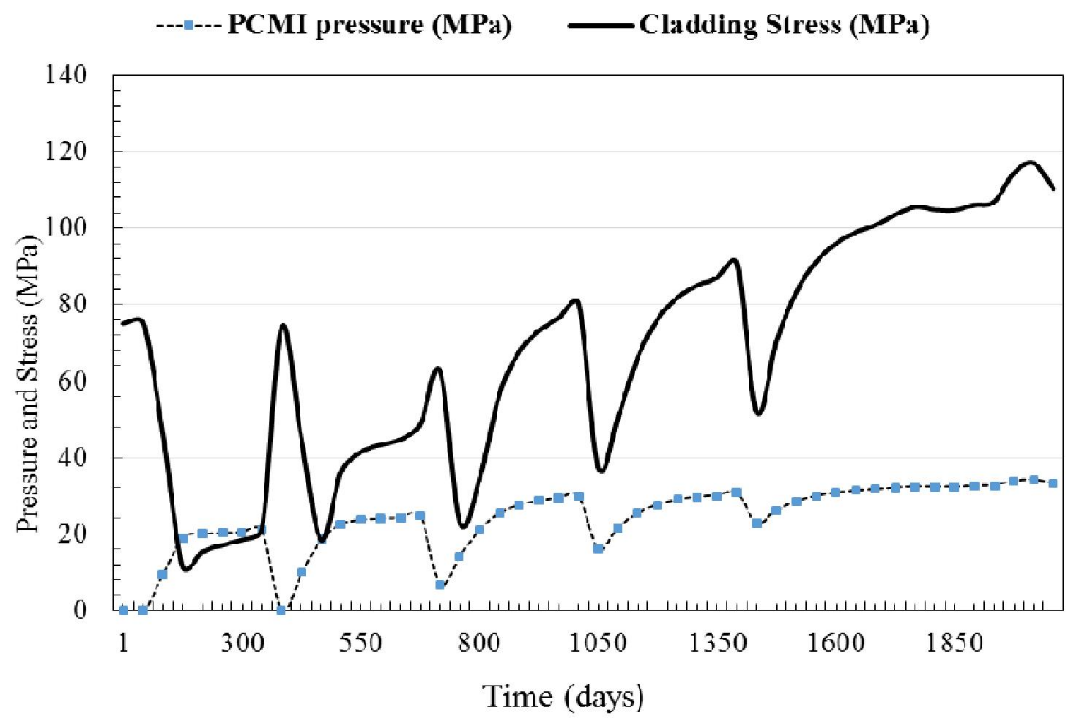

Figure 1: IFA-650.5 results obtained in the steady-state simulation.

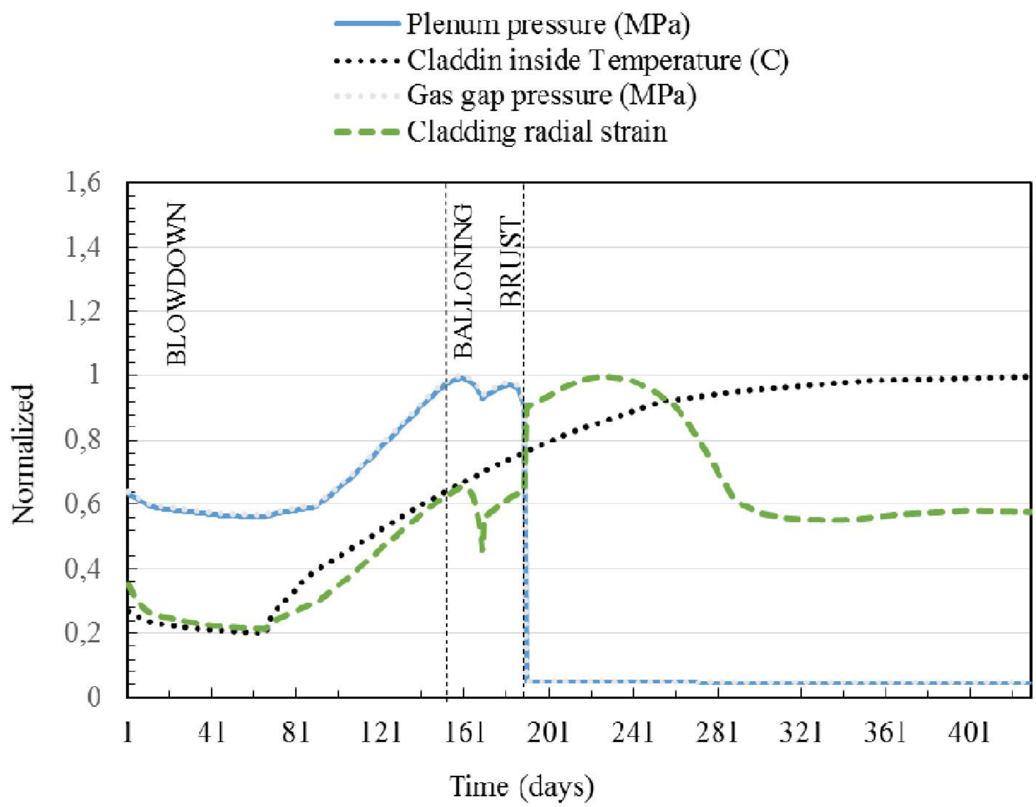

Figure 2: IFA-650.5 results obtained under LOCA conditions.

The plastic deformation is predominant during blowdown and refill phases. After the stage of irradiation simulated with FRAPCON-3.5, the transient state occurs, with imposed cladding outside temperatures. The focus on the fuel response was the internal rod pressure, ballooning, burst and oxidation limits. The simulation showed small differences of the experimental data. Table 3 presents results of IFA-650.5 under accident conditions using FRAPTRAN-1.5.

Table 3: IFA-650.5 results from FRAPTRAN code under LOCA condition.

Items	IFA-650.5
Time to rupture after start blowdown (s)	178
Axial location of rupture, (mm)	(70-80)
Axial length of rupture, crack (mm)	10
Average rod pressure from blowdown (MPa)	7
Rod pressure at rupture (MPa)	7.2
Cladding diameter increase close to rupturing (%)	12
Cladding temperature at start of heat-up (°C)	210
Temperature increase rate (°C/s)	(5.0-5.5)
Cladding temperature at rupture (°C)	750
Temperature upper thermocouple position (°C)	1002
Temperature lower thermocouple position (°C)	1042
Cladding outer surface oxide layer (μm)	11

5. CONCLUSIONS

The code FRAPTRAN-1.5 has a slight constraint to predict the time mark of rupture of the fuel, under transient. The fission gas can elevate the internal rod pressure, which will be filling the plenum volume. The thermodynamic model have weak coupling, so producing a delay time between outcomes for experiments and simulations. The failure reported in the Halden experiment was at 179 s, but in the simulation with FRAPTRAN-1.5 was calculated to 169 s. The simulation achieved good results about the experiment, considering that the instrumentation applied in the experimental device is quite complex to simulate the accident perfectly. However, it is clear that after extended burnup, the cladding fail in low temperatures regarding the 1204°C criterion. It is noteworthy that this criterion, proposed in 1974, was established based on tests conducted in the 1970 decade, using fresh fuel, not taking into account the material degradation.

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REFERENCES

1. LUSCHER, W. G.; BEYER, C. E.; CUTA, J. M. “FRAPTRAN 1.4: a computer code for the transient analysis of oxide fuel rods”. US Nuclear Regulatory Commission, Office of Nuclear Regulatory Research, (2011).

2. MANNGÅRD, Tero; MASSIH, Ali; STENGÅRD, Jan-Olof. "Evaluation of the Halden IFA-650 loss-of-coolant accident experiments 2, 3 and 4". (2012).
3. AOUNALLAH, Yacine. "Simulation of HALDEN IFA-650 loss-of-coolant accidents tests with TRACE". Kerntechnik, Volume 77, n. 5, pp. 316-323, (2012).
4. MIETTINEN, Jaakko; STENGAARD, Jan-Olof; KELPPE, Seppo. "FRAPTRAN-GENFLO calculations on the IFA-650 LOCA test." *Institutt for energiteknikk, OECD Halden Reactor Project, Halden* (Norway), pp. 187, (2004).
5. 10 CFR 50.46, "Acceptance criteria for emergency core cooling systems for light water nuclear power reactors," Appendix K, "ECCS Evaluation Models", to 10 CFR Part 50, Code of Federal Regulations, (1996)
6. ZIMMERMANN, Martin A.; BART, Gerhard. Review of the licensing basis for RIA and LOCA transients in light of new evidence on high burnup fuel behavior. *CHIMIA International Journal for Chemistry*, **Volume. 59,** n. 12, pp. 950-956, (2005).
7. LIANG, Thomas KS; SCHULTZ, Richard R. "Development program of LOCA licensing calculation capability with RELAP5-3D in accordance with appendix K of 10 CFR 50.46". *Nuclear technology*, **Volume. 133**, n. 3, pp. 355-358, (2001).
8. NEA, OECD "Nuclear Fuel Behaviour in Loss-of-coolant Accident (LOCA) Conditions. State-of-the-art Report" Technical Report NEA-6846 pp 373, (1972)
9. PETIT, M. et al. IRSN R&D, "Studies On Fuel Behaviour Under Loca Conditions". *Fuel Behavior and Modelling Under Severe Transient and Loss of Coolant Accident (LOCA) conditions*, Proceedings of Technical Meeting Held, Japan, pp. 179, (2013).
10. WIESENACK, W. "Halden Reactor Project Activities", *Achievements and International Collaboration. In: Proc. 6th International Conference on WWER Fuel Performance, Modelling and Experimental Support,* Albena, Bulgaria, (2003).
11. BILLONE, M. C.; YAN, Y.; BURTSEVA, T. "Overview of ANL LOCA programs". *In: Proceedings of the ANL LOCA Program Review Meeting*. (2007).
12. ARKOMA, Asko et al. "Statistical analysis of fuel failures in large break loss-of-coolant accident (LBLOCA) in EPR type nuclear power plant". *Nuclear Engineering and Design,* Volume 285, pp. 1-14, (2015).
13. AMMIRABILE, Luca; WALKER, Simon P. "Dynamic ballooning analysis of a generic PWR fuel assembly using the multi-rod coupled MATARE code". *Nuclear Engineering and Design,* **Volume 268**, pp. 24-34, (2014).
14. OLANDER, Donald R.; MOTTA, Arthur T. "A new book:'light-water reactor materials'". *Nuclear Engineering and Technology,* **Volume. 37**, n. 4, pp 309-316, (2005).