

Steam Flow-rate Effect on the Transient Behaviour in Phebus Experiment FPT-1

Salwa Helmy¹, Basma. Foad¹

N. Eng. Safety, Dept. of NRRA, Cairo, Egypt¹

Nuclear and Radiological Regulatory Authority (NRRA)

ABSTRACT: In this work the PHEBUS FPT-1 test is simulated and re-analyzed using ASTEC code to investigate fuel bundle degradation in a Pressurized Water Reactor. The main objective is to study the effect of steam flow rate due to increase the inlet steam mass flow in the primary circuit to double value, on fuel degradation, and structure material in the reactor coolant system. During this model; the peak bundle temperatures and amount of magma are decrease to half value as a result of increasing of the inlet steam mass flow. In addition, the fuel and clad temperatures decreased due to more cooling, and the main part of hydrogen, mainly coming from Zr oxidation, is also reduced.

KEYWORDS: severe accidents, ASTEC code, PHEBUS FP, PWR, core degradation; fission products.

I. INTRODUCTION

The PHEBUS Fission Product (FP) programme was initiated in 1988 after major severe accidents in nuclear reactors such as Three Mile Island (TMI) in 1979 and Chernobyl in 1986. The main objective of the FPT-1 programme is to study the transport and retention of Fission product on reactor behaviour during severe accidents for Light Water Reactor (LWR) [1]. The PHEBUS FP programme comprises of six integral experiments as shown in table (1) [2]. For the first technological FPT-0 test fresh fuel was used (but with a pre-irradiation phase). Test FPT-1 test would be similar but with irradiated fuel, and the following tests were numbered FPT-2, FPT-3, FPT-4 and FPT5 [3]. Test FPT-2 will be similar to FPT-1 but the boric acid will be added to the steam and the flow will also be reduced to produce a window of reducing conditions. The tests did not span the whole range of conditions which might be expected in core melt accidents, but it was felt that six tests would provide a good basis for code assessment while remaining within the financial envelope and the residual life time of the reactor [4]. The areas covered by the FPT1 experiment are:

- Fuel degradation, hydrogen production, release of fission products, fuel, and structural materials (bundle part).
- Fission product and aerosol transport in the circuit (circuit part).
- Thermal hydraulics and aerosol physics in the containment (containment part);
Iodine chemistry in the containment (chemistry part).

ASTEC code was used for the simulation of FPT-1 test and it was found that the code is capable to model the main severe accidents phenomena in the fuel bundle, heating and melting.

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

Table 1: Phébus-FP test matrix

No.	Type of fuel	Fuel bundle	Primary circuit	Containment vessel	Date
FPT-0	Fresh fuel, 1 Ag-In-Cd rod, 9 d. pre-irrad.	Melt progression & FP release in vapor rich environment	FP chemistry and deposits in non-condensing steam generator.	Aerosol deposition. Iodine radiochemistry at pH 5.	Dec. 2, 1993
FPT-1	BR3 fuel ~23GWd/tU, 1 Ag-In-Cd rod, re-irrad.	As FPT-0 with irradiated fuel.	As FPT-0	As FPT-0.	July 26, 1996
FPT-2	As FPT-1	As FPT-1 under steam poor conditions.	As FPT-1 with effect of boric acid.	H ₂ recombiner, pH9, evaporating sump,	2000
FPT-3	As FPT-1, but with B4C instead of Ag-In-Cd	As FPT-2	As FPT-0	As FPT-2?	2003
FPT-4	EdF fuel ~33GWd/tU No re-irradiation.	Low volatile FP & actinide release from UO ₂ - ZrO ₂ debris bed, up to melting.	Integral filters in test device Post-test studies on samples		July 22, 1999
FPT-5	Pre-irrad. as FPT-1	Fuel degradation and FP release in air conditions.	Deposition & chemistry of FPs in air conditions.	As FPT-1 or 2	2004-2005

II. FPT-1 EXPERIMENT

PHEBUS facility reproduces at 1:5000 scale a 900 MWe PWR. The FPT1 test bundle as shown in figure (1) includes 18 PWR fuel rods previously irradiated to a mean burn-up of ~23.4 GWd/tU, two instrumented fresh fuel rods and one silver-indium-cadmium (AIC) control rod. The FPT1 test bundle is approximately 1 m in length and contained a total mass of about 11 kg of fuel enclosed in zircaloy cladding. The fuel rods are held in place by two Zircaloy spacer grids, and arranged in a 5x5 square lattice on a pitch of 12.6 mm, without the four corner rods. The spacer grids are of depth ~0.043 m and are located at 0.24 and 0.76 m elevations from the bottom of the fuel column. The grids are linked to four Zircaloy stiffeners.

The bundle is housed in an insulating shroud and introduced into an in-pile tube cooled by a pressurized water circuit. Where the RCS; in which steam generator is represented by an inverted U-tube. The containment represented by a 10 m 3 vessel with a water-filled part for the sump, a gas-filled part and painted surfaces [5].

The FPT1 test bundle was pre-irradiated for ~7 days with a mean bundle power of ~205 kW in the PHEBUS reactor before the experimental phase of the test in order to generate short-lived fission products in the fuel. This pre-conditioning phase was followed by a 5 hour transient during which steam at a pressure of ~ 2 bars was injected into the bundle with a flow rate varying between 0.5 and 2.2 g/s. Meanwhile, the bundle nuclear power was progressively increased up to 34.4 kW.

The power generated within the fuel rods produced the heat-up of the test bundle leading to the bursting of the cladding, degradation of the control rod and relocation of the absorber material, cladding oxidation and hydrogen generation, fuel liquefaction, melt relocation and accumulation, and the release of fission products and bundle, structure and control rod material. The main result from the FPT-1 test, as far as the bundle is concerned, was an early fuel relocation leading to a large degradation of the fuel rods with the progressive formation of a UO₂-rich molten pool in the lower part of the bundle [6]&[7].

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

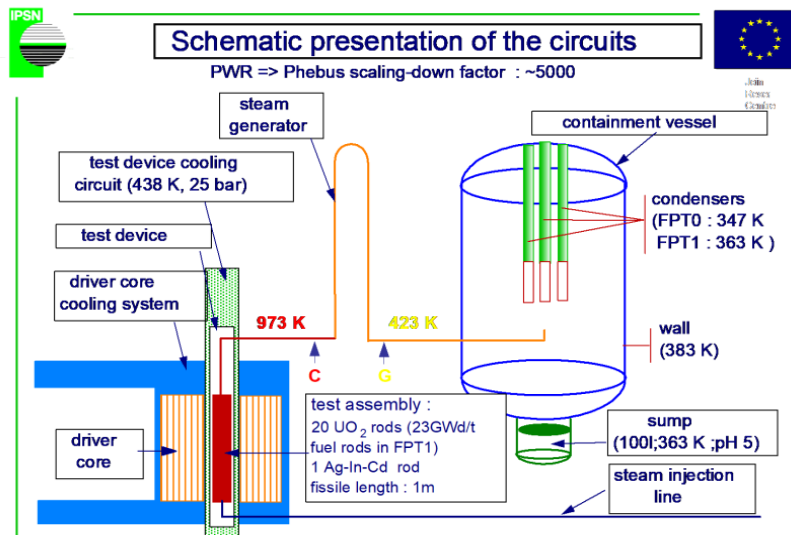
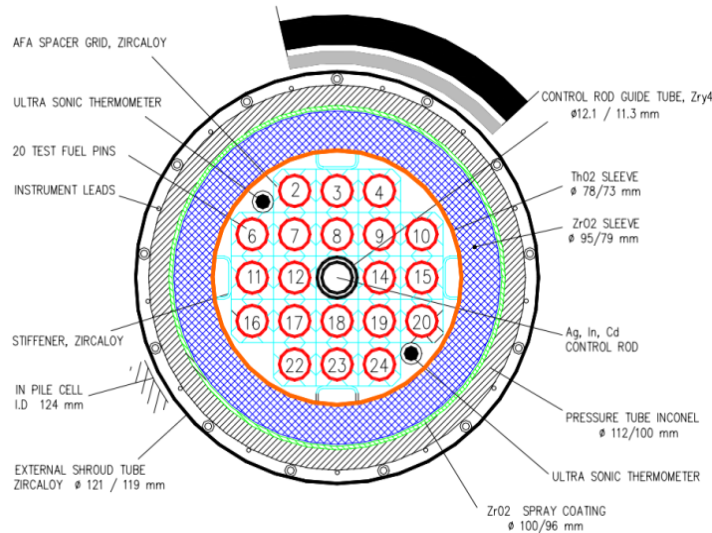


FIGURE 1: PHEBUS EXPERIMENTAL FACILITY [2].

III. ASTEC CODE

The purpose of the ASTEC software package (Accident Source Term Evaluation Code) is to simulate all the phenomena that occur during a severe accident in a water-cooled nuclear reactor, from the initiating event to the possible release of radioactive products (the 'source term') outside the containment.

The code structure is modular, each of its modules simulating a reactor zone or a set of physical phenomena. The design of ASTEC code allows running any module in a stand-alone mode. This mode allows performing the module validation on separate-effect or coupled-effect experiments

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

Description of ASTEC Modules [8]:

CESAR: The CESAR module simulates primary and secondary circuit thermal-hydraulics, intact and degraded core thermal-hydraulics, and porous media thermal-hydraulics (debris bed reflood applications)

ICARE: The ICARE module describes the in-vessel degradation phenomena (both early and late degradation phases). ICARE allows simulating the early-phase of core degradation with fuel rod heat-up, ballooning and burst, clad oxidation, fuel rod embrittlement or melting, molten mixture candling and relocation. It also can simulate the late-phase of core degradation with corium accumulation within the core channels and formation of blockages, corium slump into the lower head and corium behaviour in the lower head until vessel failure

ELSA: The ELSA module aims at simulating the release of fission products and structural materials from the degraded core during a severe reactor accident. ELSA is tightly coupled with the ICARE module which treats the phenomena of the core degradation.

SOPHAEROS: The SOPHAEROS module simulates transport of FP vapors and aerosols in the RCS through gas flow to the containment

CPA: The CPA module simulates thermal-hydraulics and aerosol behaviour in containment. It simulates simple or multi-compartment containments (tunnels, pit, dome...) with possible leakages to the environment or to normal buildings, with more or less large openings to the environment.

IODE: The IODE module simulates iodine behaviour in the containment. It describes in a kinetic way the chemical transformations of iodine.

DOSE: The DOSE module allows evaluating the dose rate in the containment.

ISODOP: The ISODOP module simulates decay of FP and actinide isotopes in different zones of the reactor
In the present studies: ICARE, CESRA, CPA, SOPHAERO, DOSE and ISODOP modules are activated (where ELSA module is activated automatically as vessel is called).

IV.FPT-1 MODEL FOR ASTEC CODE

The reactor core is represented by 20 fuel rods, analogous to those in a PWR, along with a 1-metre-long absorber rod, and two grids. In the model fuel bundle of 1 control rod, 20 fuel rods and 2 grids is modeled. Two channels are defined for the whole test section. Every channel contains fuel rods each of which composed of two macro-components: FUEL and CLAD. The primary circuit is represented by 12 volumes; where the upper part - consists of 3 volumes (B_UP), and the hot-leg consists of 2 volumes (H_L). The steam generator is divided to 6 volumes (SG), and finally the cold-leg consists of 1 volume (CL_1). In addition, the bottom part is represented by VOLBOT volume.

A steam circulation from the bottom to the top with a flow rate is varying with time according to the experimental data.

V. RESULTS AND DISCUSSION

The Results consists of two parts, the base case results which is previously described and the results due to increasing the inlet steam flow rate in the primary circuit to double value to investigate the effect of the steam flow rate in on fuel degradation.

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

Base Case Results

Figure (2) presents ASTEC Primary circuit nodalization of FPT-1 from the core up to the cold-leg.

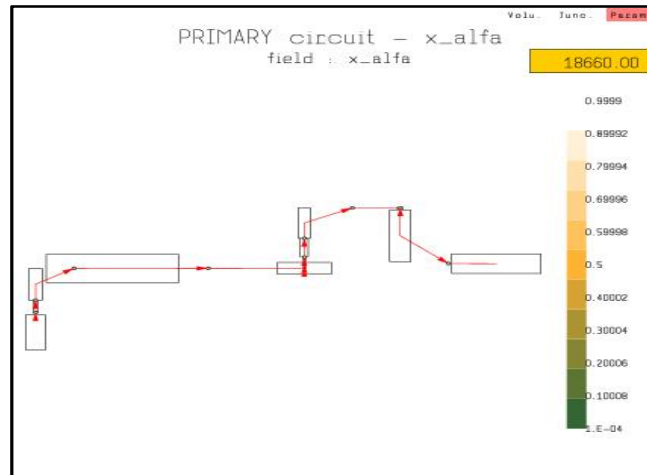


Figure 2: Primary circuit

Figure (3) compares the inlet flow rate between the experimental and ASTEC calculation. The rate of the steam at a pressure of ~ 2 bars was injected into the bundle with a flow rate varying between 0.5 and 2.2 g/s. The evolution of the bundle nuclear power is investigated in figure (4).

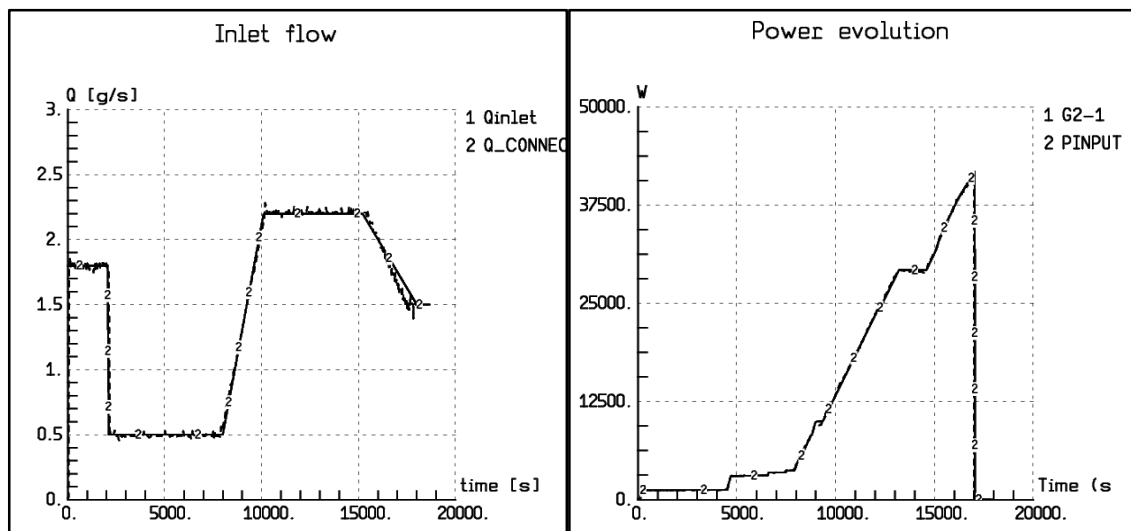


Figure 3: The steam inlet flow rate

Figure 4: The bundle power

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

The bundle degradation phase consisted of two main periods:

- 1- The thermal calibration of the bundle ~7900 seconds. During this period, the bundle power and the steam flow rate were increased step by step in order to check the thermal response of the bundle. The steam flow rate was first reduced from 1.8 to 0.5 g/s, when the bundle power was increased in steps from 0 to 3.90 kW.
- 2- The second period lasting from ~7900 to ~17000 seconds was the real bundle degradation period specifically devoted to the release of fission products, and bundle, structure and control rod material in order to study their transport and retention in the experimental circuit.

During the degradation phase substantial hydrogen production was observed. The degradation phase was terminated by shutdown of the nuclear power at 17039 s, and the bundle was cooled in a steam flow of 1.5 g/s. At the end of this cooling phase the containment was isolated at 18660 s.

The power generated within the fuel rods produced the heat-up of the test bundle leading to the bursting of the cladding, degradation of the control rod as shown in figure (5), where the schematic diagram of the degradation figure explains evolution of the bundle nuclear power, fuel & shroud temperature and H₂ production. After cladding oxidation and hydrogen generation, fuel liquefaction, and relocation of the absorber material, Figure (6) investigates the amount of mass of magma which reached to 3kg at 18000sec.

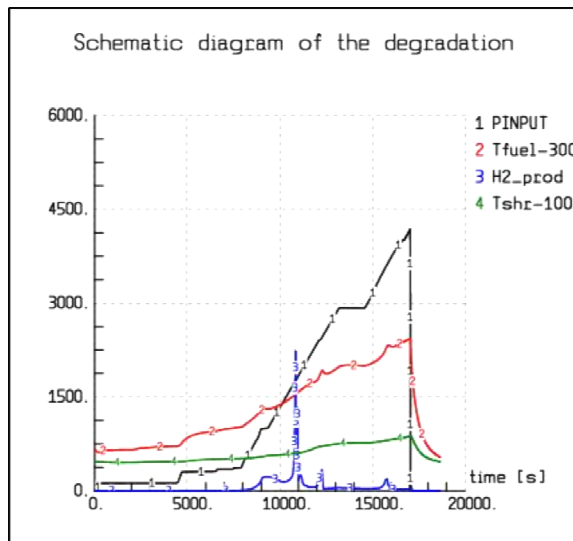


Figure 5: Schematic diagram of the degradation

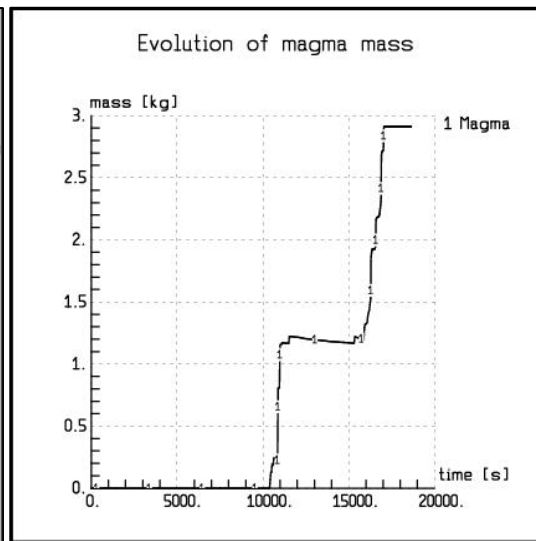


Figure 6: Evolution of magma mass

Figures (7) and (8) show the temperature comparison between the experimental and ASTEC calculations for fuel3 and clad3 (channel 2) at 600 and 700 mm. The temperature of fuel3 reached 3400 K and the temperature of clad reached 2400 K, where TCW refers to experimental results and T_{fuel} refers to ASTEC results.

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

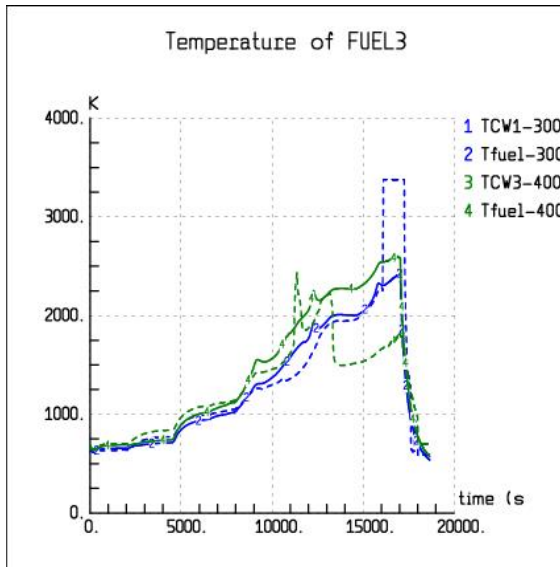


Figure 7: Temperature of Fuel3

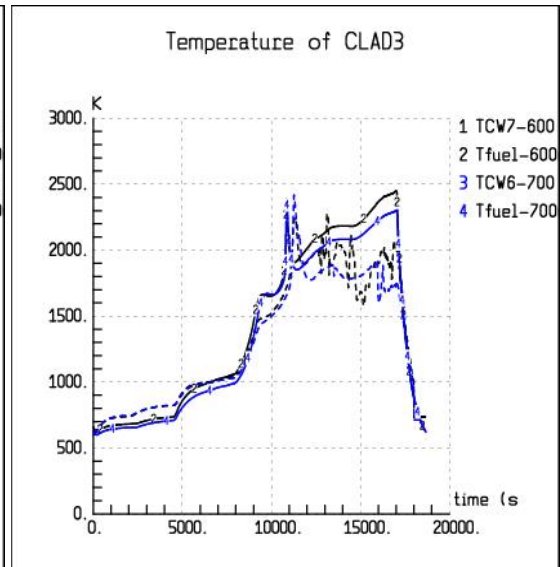


Figure 8: Temperature of clad3

Figure (9) shows that the bundle void fraction at the end of run (time 18660sec) equals 1.0, therefore the whole core bundle was completely vapor. In addition, the core field temperature lies between 800- 1000 K as indicated in figure (10).

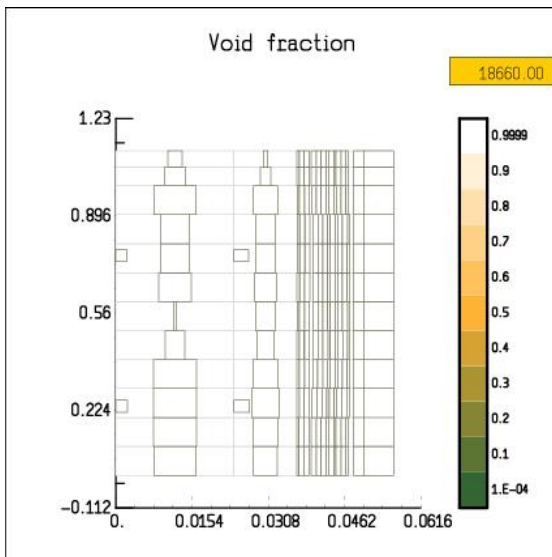


Figure 9: Bundle void fraction

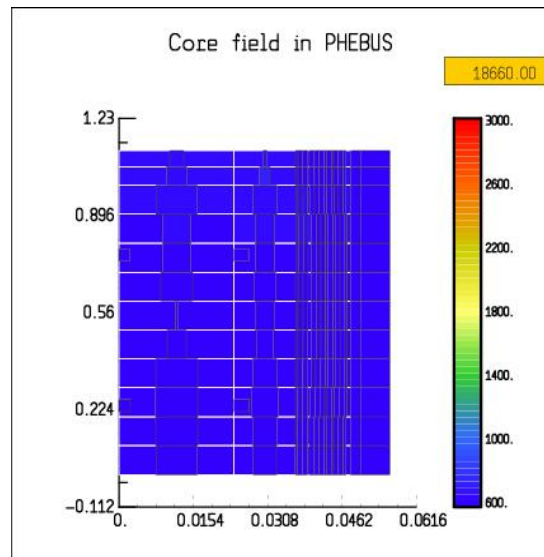


Figure 10: Core field temperature

Hydrogen production

Figure (11) explains the in-vessel production of H₂ located in all ASTEC volumes, i.e inside the primary circuit, the vessel and the containment. As it is expected; the main part of hydrogen is mainly coming from Zr oxidation (Zr

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

claddings, Zr guide tubes, and U-OZr relocated mixtures) where H₂ production. Figure (12) shows the rate of hydrogen generation where the maximum rate is 0.2 g/s at 10000.0 sec.

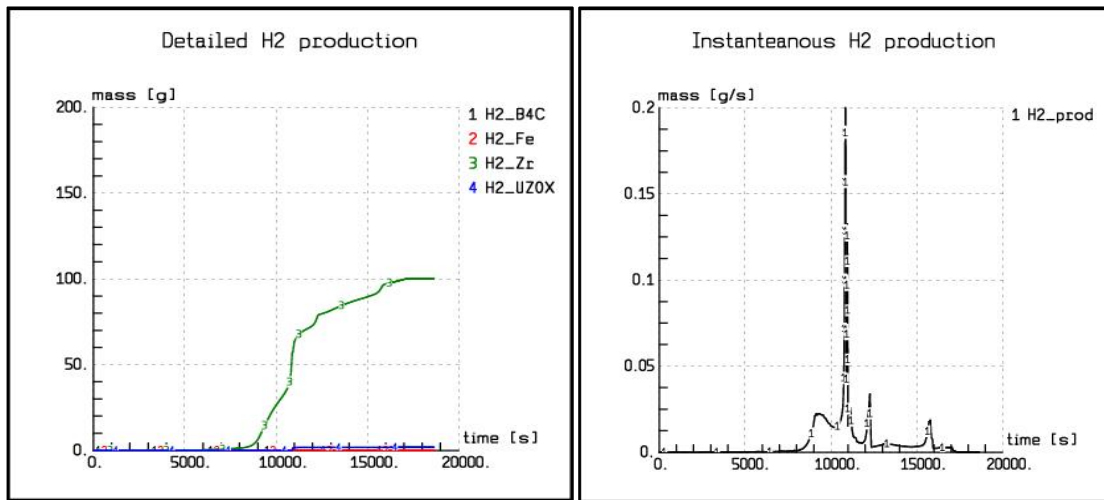


Figure 11: Detailed Hydrogen production

Figure 12: Instantaneous Hydrogen production

VI. PARAMETRIC STUDY RESULTS

In this Study the inlet steam flow in the primary circuit increased to double value (relative to the base case). Figure (13) compares the inlet steam mass flow which injected in the primary circuit for the parametric study and the base case. The base case is denoted by (case1) while the parametric study case is denoted by (case2).

The input power, fuel and shroud temperatures, and the hydrogen production and mass are represented in figure (14) and (15). Due to the increase of the steam injection rate the progression of the Zircaloy cladding oxidation (represented by the hydrogen production) was faster and lower in case2 than in case1. Accordingly the fuel and shroud temperatures for case2 are lower compared to the base case.

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

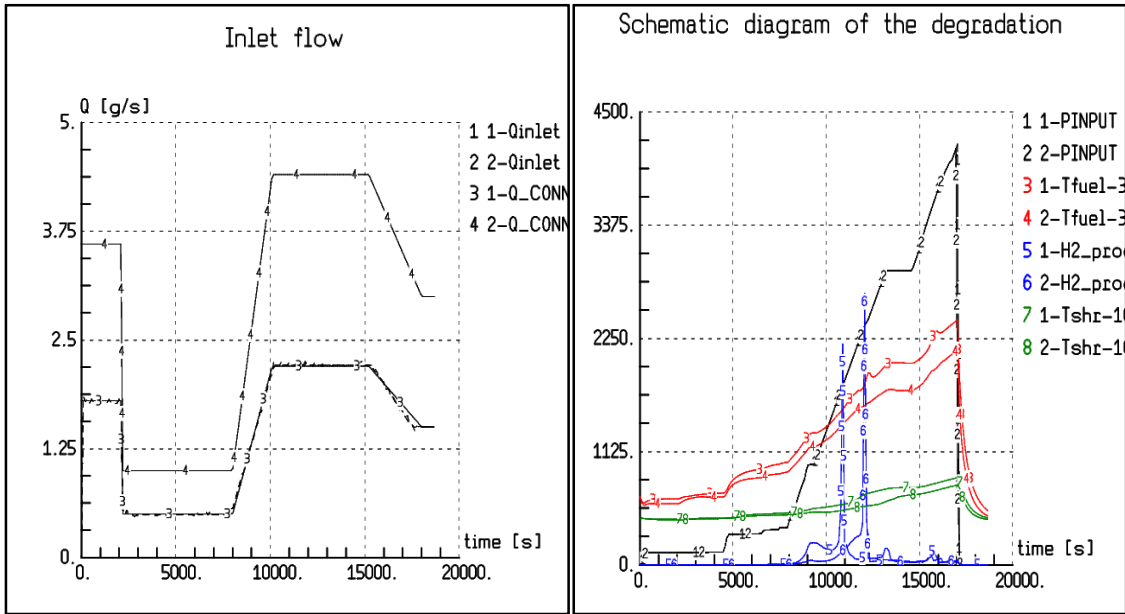


Figure 13: Inlet flow

Figure 14: Schematic diagram of the degradation

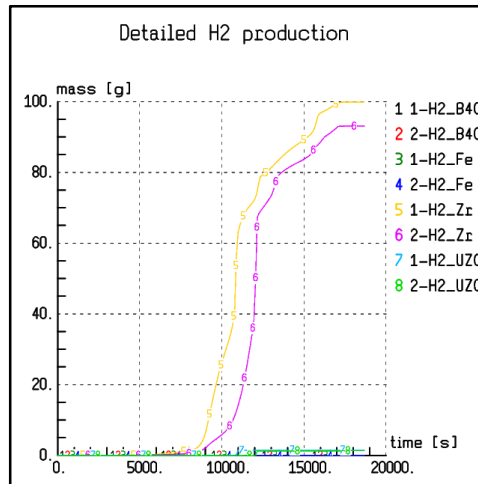


Figure 15: Hydrogen mass

Figure (16) compares the clad temperatures profile for both cases. As can be seen the clad temperature for case2 is lower than case1 due to the increase of the steam injection rate. The evolution of magma mass is shown in figure (17). Where the amount of magma for case2 was half the value of case2 (about 1.5 Kg) due to the increase of the cooling rate.

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

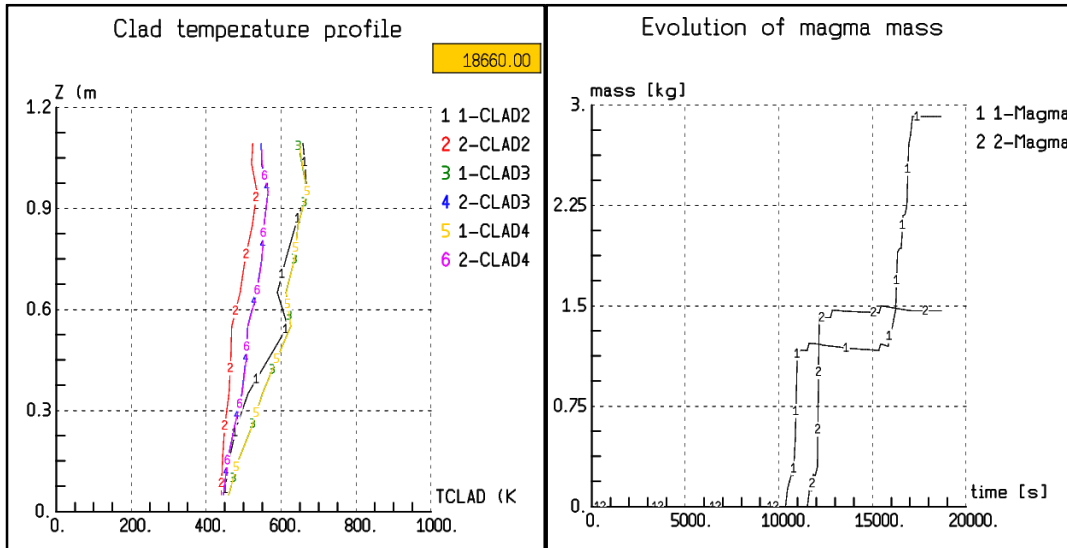


Figure 16: Clad temperatures profile

Figure 17: Evolution of magma mass

Figure (18) and (19) (case 1 & case 2) show the core degradation and core magma saturation, where the material relocation events inside the bundle occurred slower compared to case 1.

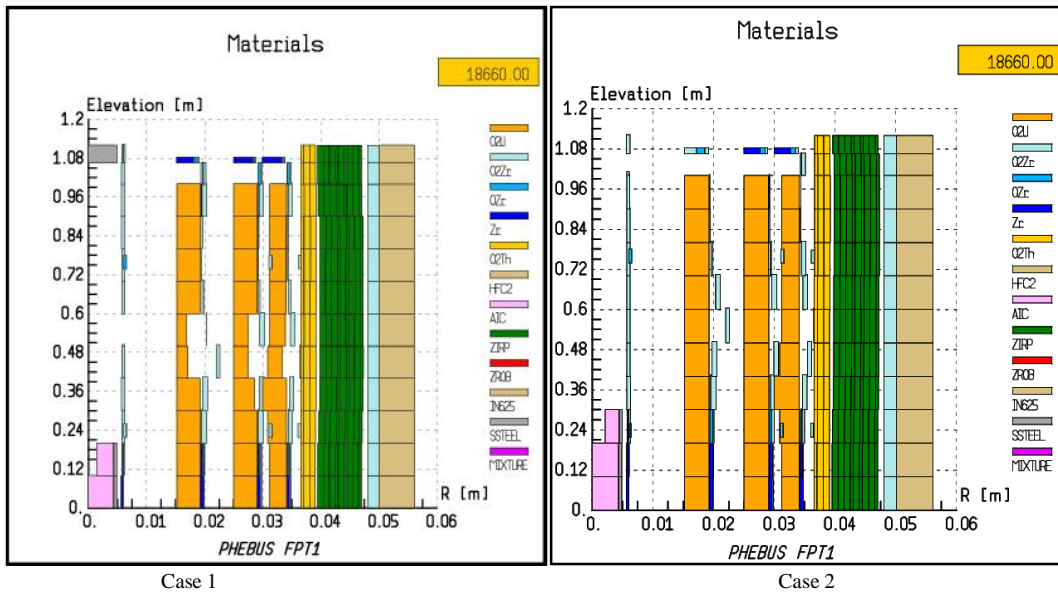


Figure 18 Material Core degradation

International Journal of Innovative Research in Science, Engineering and Technology

(A High Impact Factor, Monthly, Peer Reviewed Journal)

Visit: www.ijirset.com

Vol. 8, Issue 2, February 2019

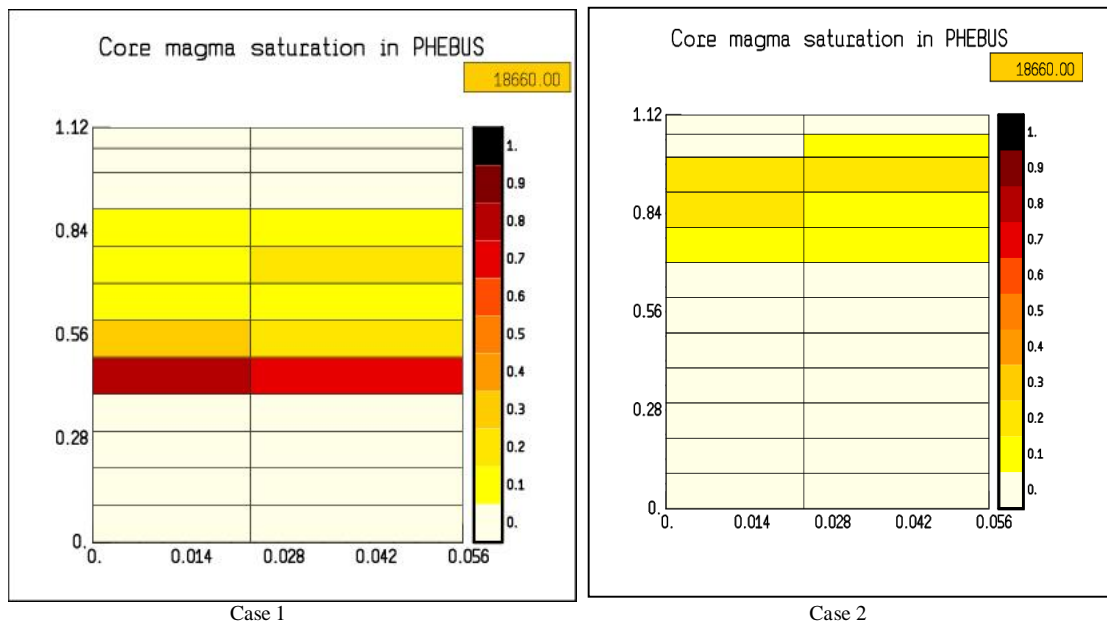


Figure 19 Core magma saturation

VII. CONCLUSION

Our paper is mainly focus on ASTEC code in studying the transient behavior of PHEBUS experiment due to its importance in investigating the fuel bundle degradation of Pressurized Water Reactor. It was found that ASTEC code was capable to model the main severe accidents phenomena in the fuel bundle, heating and melting of FPT-1 experiment. In addition, the consequence of increasing the steam flow rate in the primary circuit to double value was also analysed. Increasing the steam flow rate was leading to a delay in the fuel degradation, with the formation of less amount of magma (about half value). Therefore the peak bundle temperatures, fuel and clad temperatures, decreased as a result of increasing the cooling flow rate. Finally, the hydrogen mass and production rate were compared with the base case, and it was shown that the hydrogen coming from Zr oxidation was reduced due to the decreasing of the steam flow rate.

REFERENCES

- [1] V., Hardt, P., Jones, A.V., Lecomte, and C., A., Tattegrain, "Nuclear Safety Research The Phebus FP Severe Accident Experimental Programme," *Nuclear Safety*, Vol. 35(2), July-, pp. 187-205 December, 1994.
- [2] S., Marco, G., Aleksander A., Luca "Circuit and Containment Aspects of PHÉBUS Experiment FPT-2", Final Interpretation Report, European Commission, 2015.
- [3] M., Schwarz and B., Clement "Practical Lessons from the First Three PHEBUS FP Tests", JAERI-Conf 2000-015
- [4] V., Vileiniskis, and A., Kaliaika, "Best estimate analysis of PHEBUS FPT1 experiment bundle phase using ASTEC code ICARE module", *Kerntechnik*; 76, 4; 254-260; 2011.
- [5] "FPT1 Final Report Volume 1 Phebus Progress report", April-September 2004
- [6] L. Bosland, G. Weber, W. Klein-Hessling, N. Girault and B. Clement "Modeling and Interpretation of Iodine Behavior in PHEBUS FPT-1 Containment with ASTEC and COCOSYS Codes", *Nuclear Technology Issue 1 Volume 177*, 2012.
- [7] G., Repetto, J., Birchley, T., Drath, and H., Austregesilo, "Analysis of the PHEBUS FPT3 Core Degradation Using Severe Accidents Codes (ICARE/CATHARE, ATHLET-CD, MELCOR)", in *TLEAR SORANSACTIONS- AMERICAN NUCCIETY*; 98; 270-27; 2008.
- [8] P. Chatelard, N. Reinke, "Overview of the integral code ASTEC V2.0", *Rapport Technique DPAM/SEMCA-2009-149*, Technische Notiz GRS ASTEC 2009.