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SEVERE ACCIDENT ANALYSIS FOR LOFW OF STEAM GENERATOR COMBINED WITH AN ATWS

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Abstract. *An Anticipated Transient Without SCRAM (ATWS) is a trip failure for severe accident initiated from the transients like a control bank withdrawal, and a loss of feed water. The main concern is whether a power excursion and pressure oscillation can occur in such an event: an ATWS event could produce system over pressurization and increases the core power, producing rapid uncovered and reactor fuel cladding damage and resulting in a release of radioactive fission products to the containment if the fuel in the reactor core was melting. Furthermore, since thermal-hydraulic feedback plays an important role in these accidents, the uncertainty of the reactivity feedback coefficients used can impact the outcome of the analysis for such a power excursion. Since transients accompanied by scram failure would result in serious consequences (and it could be expected to occur in the future population of nuclear plants), the objective of this work was to perform analysis for severe accident scenarios of a generic Pressurized Water Reactor (PWR) using MELCOR v2.2 code, considering a Total Loss of Feedwater (LOFW) scenario combined with a ATWS reactivity accident. The results are compared with ATWS acceptance criteria reported in the NUREG-0800.*

Keywords: *Severe Accident, MELCOR, Loss of Feedwater (LOFW), Anticipated Transients Without SCRAM (ATWS), Pressurized Water Reactor (PWR)*

1. INTRODUCTION

The recent interest in severe accidents analyzes in nuclear power plants has changed after the two most significant nuclear accidents in history, the Chernobyl Reactor Four explosion in Ukraine (1986) and the Fukushima Daiichi accident in Japan (2011). Many countries now require such analyzes as part of the documents for the licensing of Nuclear Power Plants (NPP) (USNRC, 2017). Anticipated Transient Without SCRAM (ATWS) accident is among the list of initiating events that should be considered in the Safety Analysis Reports (SARs) – as given in the US Nuclear Regulatory Commission (NRC) Standard Format (USNRC, 2017). For the regulatory body the SAR is the essential document for the licensing process. The Brazilian Nuclear Energy Commission (CNEN) adopts the same criteria to the licensing process of power reactors (CNEN, 1984).

The NPP behavior is governed by conjugate heat transfer (thermal fluids) in the core and primary coolant loop. Radiation transport and kinetics may play a role if the core remains critical, i.e., during an ATWS event, and fluid/aerosol dynamics and oxidation may be involved if the primary loop is depressurized, intentionally or not. Core integrity, however, is mainly a function of heat and coolant mass transfer which may be quite complex depending on the extent of boiling and buoyancy driven (natural circulation) flow. If the sequence results in significant core damage, neutron kinetics has a very limited role but other physics become important, including: circulation of steam and non-condensable gases, oxidation of core materials (especially stainless steel cladding), loss of core geometry (clad ballooning and rupture, melting of fuel, cladding, and other structures, relocation, embrittlement and fragmentation of corium), heating of the lower head, and release of fission products from the fuel matrix and transport to the coolant. If the primary coolant boundary is breached, ex-vessel phenomena such as containment thermal-hydraulics, hydrogen accumulation, aerosol behavior, and corium spreading and interactions with concrete and other materials must be modeled in a severe accident analysis.

ATWS arises as an event that must be studied in a plant severe accident analysis. ATWS are events in which the SCRAM system (or "reactor trip") – whose is the reactor protection and reactivity control system – fail to operate (e.g., following an anticipated transient such as a loss of the steam generator (SG) feed water, or a complete failure of the rapid insertion of the control rods to shutdown the reactor). If the SCRAM does not occur (or by the actions of other systems), some anticipated transients could result in the reactor core melting – because of possible boiling crisis, leading to fuel and cladding damage. The severity of such ATWS events varies with the design of the reactor and can be modified by the action of the reactor protection systems. The computer code MELCOR version 2.2 can be used to simulate the transient response of PWRs – and other Light Water Reactors (LWRs) – to improve the severe accidents analysis (Humphries, L. L. et.al., 2017).

In a previous paper, the LOFW accident with SCRAM was analyzed for a reference NPP (Gual, M. R. et. al., 2019). For the same reference plant, the present paper brings the results of a model in MELCOR to simulate a LOFW of a SG located in secondary loop of combined with an ATWS, as a hypothetical accident scenario. The following sections present the MELCOR code, the accident scenarios considered and the main results (in this order).

2. MELCOR CODE

The thermohydraulic phenomena in the core region directly affect the reactor kinetics or neutronics which in return influence the behavior of the thermohydraulics. In the MELCOR code, thermal-hydraulic data calculated by the Control Volume Hydrodynamics (CVH) and Flow Path (FL) packages provide boundary conditions to other phenomenological packages such as Burn (BUR), Cavity (CAV), Core (COR), Fuel Dispersal Interactions (FDI), and Heat Structures (HS). These packages, in turn, calculate sources and sinks of mass and energy for CVH. Thermal-hydraulic processes interact with and are coupled to all aspects of accident phenomenology. A fluid control volume (CVH package) is also associated with the COR cell and exchanges heat with components in the cell. A heat structure (HS package) may also be associated with the COR cell and exchange heat with components in the cell. The CVH package and the FL package are responsible for modeling the thermal-hydraulic behavior of coolant liquids and gases. Each flow path may be described in terms of a number of segments differing, e.g., lengths, areas, hydraulic diameters, and surface roughness's. The governing equations for thermal-hydraulic behavior in MELCOR are the equations of conservation of mass, momentum, and energy. These equations are presented first as ordinary differential equations for the control volume formulation and then in linearized-implicit finite difference form that is actually solved (Humphries, L. L. et. al., 2017).

The MELCOR code includes thermal-hydraulic response in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heat up, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. Also, it includes impact of engineered safety features on thermal-hydraulic and radio nuclide behavior.

3. MATERIALS AND METHODS

The LOFW combined with an ATWS (LOFW+ATWS) is a hypothetical scenario in order to cause the core melt rapidly and observe the slumping of corium to the lower head (LH), which then leads to heating up of the bottom of the reactor pressure vessel (RPV).

MELCOR reactor point kinetic (PK) model (Humphries, L. L. et. al., 2017) can be used to analyze the ATWS initiated by a LOFW transient of SG. Thus, a point kinetics model was added to allow MELCOR calculation of accident sequences without SCRAM. The model (Humphries, L. L. et. al., 2019): a) include an active neutron source for zero-power reactor startup, and; b) allows for reactivity feedback effects: external, user-specified reactivity insertion (positive/negative), Fuel Doppler feedback; fuel density feedback, graphite density feedback. The primary containment is simulated and external cooling of the reactor vessel is considered.

The developed 6 delayed group kinetics model introduced in MELCOR code was successfully benchmarked using the inhour solution for step reactivity insertions (both positive and negative). Insertion of control rod into the reactor terminates the progression of ATWS events to severe accidents; however, if all neutron absorber systems were assumed to fail, an ATWS could progress to a severe accident.

ATWS results in the over pressurizing of the Reactor Cooling System (RCS) due to the imbalance of heat generation in the core and heat removal by the secondary side of steam generator (SG). In consequence, the pressurizer safety valve (PSV) should reach its set point and open, leading to system depressurization. The safety approach to the treatment of ATWS events for currently licensed PWRs is established in 10 CFR 50.62 (USNRC, 1984) in the form of prescriptive design requirements. 10 CFR 50.62 also provide the acceptance criteria for the fuel centerline temperature and the system pressure, where the centerline temperature must not exceed 2200 °F (1204.4 °C) and the system pressure must not exceed 110% of the design pressure. 10 CFR 50.46 (USNRC, 1988) established that the calculated maximum cladding temperature after the accident should not exceed 1204°C for Emergency Core Cooling System (ECCS).

Next sections present the developed MELCOR model for the reference NPP (including de RCS and the containment), the scenarios selection, and the scenarios description – considering the initial conditions and the events sequences as presented in the plant Level 1 Probabilistic Safety Assessment (PSA).

3.1 MELCOR Model

The reference NPP is a two-loop PWR. The description, the nodalization, the initial steady-state conditions was presented in previous paper (Araújo, N. N. et. al., 2019). The model is illustrated in Fig. 1.

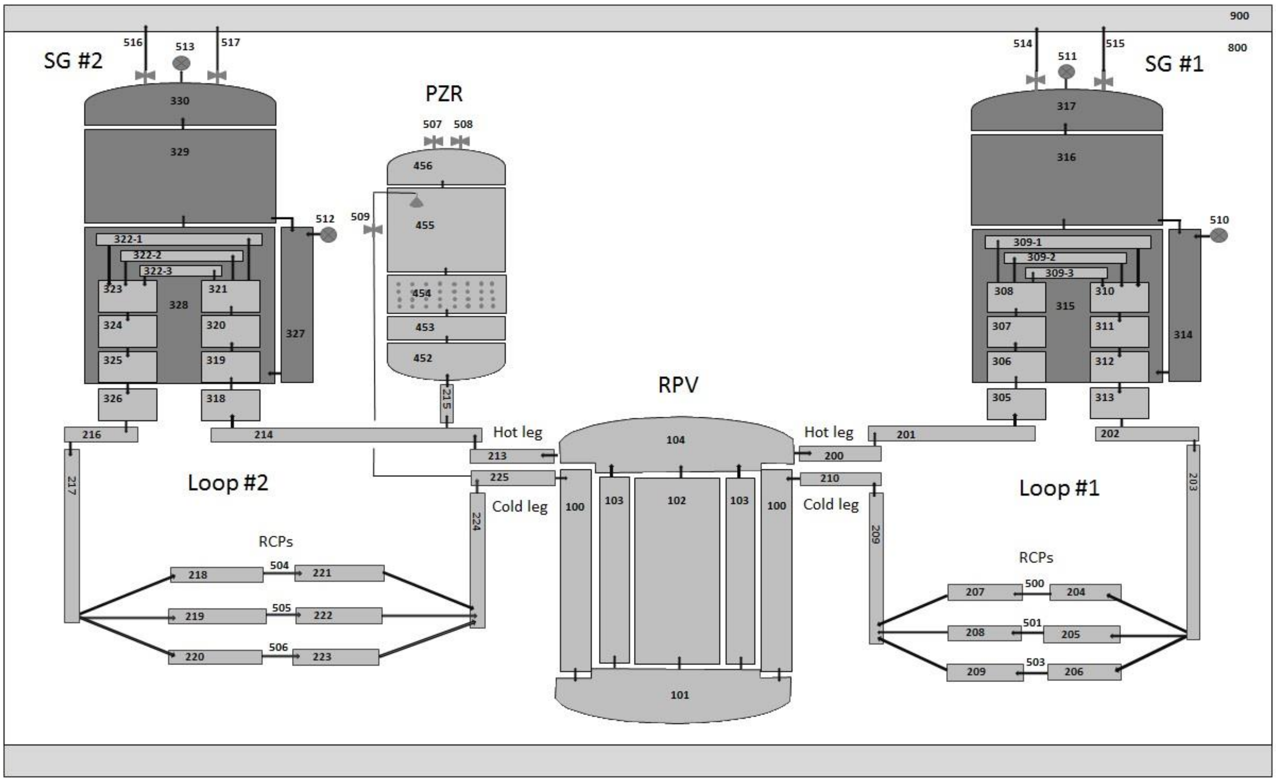

Fig. 1. The reference NPP model. (Araújo, N. N. et. al., 2019)

The main initial parameters taken from (Araújo, N.N. et. al., 2019) are given in Tab. 1.

Table 1. Main initial parameters.

Parameter	Nominal Value
Core Power, MWt	48.00
Primary pressure, MPa	13.10
Secondary side pressure, MPa	3.77
Pressurizer (PZR) pressure, MPa	13.10
Coolant Temperature at the core outlet, K	558.08
Coolant Temperature at the core inlet, K	537.95
Secondary temperature, K	519.15

3.2 Event Selection

This paper takes into account the results of the PSA Level 1 for the reference NPP. The PSA Level 1 analyses accident phenomena in nuclear power plants that regard the initiating events which, in principle, were considered relevant for the calculation of the plant's Core Damage Frequency (CDF). The PSA Level 2 addresses the phenomenological and physical events that can occur during and after core melt. In PSA Level 1 for the reference NPP, it was identified that the contribution of ATWS accidents is equal to 0.15 % of the overall CDF (Maturana, M. C. et al., 2018), as the results presented in Tab. 2.

In the Internal Events Level 1 PSA (for transients) for the reference plant, the LOFW of SG initiating event appears in the cut set with the highest frequency (the highest contribution to the CDF of the plant, therefore) if an ATWS is considered – and almost 50% of the contribution of the sequences with AWTS to the CDF. The events considered in this cut set were presented in Tab. 3.

Table 2. CDF in the Level 1 PSA for the reference NPP (Maturana, M. C. et al., 2018)

Operational Mode	Initiating Event		CDF (/yr)	Percentage of Total CDF
Full Power	Internal Events	Transients	3.99E-06	1.80%
		Loss of Coolant Accident (LOCA)	3.30E-06	1.49%
		ATWS	3.26E-07	0.15%
		Interfacing Systems Loss of Coolant Accident (ISLOCA)	*	0.00%
	External Events	Seismic Events	*	0.00%
		Internal Fire	1.66E-04	74.75%
		Internal Flood	3.25E-06	1.46%
		Tornado	*	0.00%
		External Flood	*	0.00%
		Aircraft Crash	*	0.00%
Low Power and Shutdown	Internal Events	Shutdown	4.52E-05	20.35%
Total CDF			2.22E-04	100%

Table 3. Cut set with the highest frequency in the Level 1 PSA

Event Description	Event Probability	Cut Set Frequency (/yr)
LOFW	1.77E-1	1.63E-07
Plant is at power operation based on annual availability	1.00E+0	
Flag to enable transient events	1.00E+0	
Failure of operator manual trip from control room	5.10E-2	
Reactor protection system (RPS) fail to send reactor trip	1.80E-5	
Transient w/o SCRAM and MTC is not acceptable	1.00E+0	

3.3 Initial Condition of Hypothetical LOFW+ATWS accident

The initial conditions of the hypothetical LOFW+ATWS accident considered for simulation are:

1. The reactor is operating at the initial nominal power 100%;
2. No safety system is available or operational (SCRAM);
3. Reactor protection system (RPS) actuation fails to start;
4. Feed Water Pumps (FWPs) in secondary loop fails to start.

Table 4 listed the boundary conditions applied in the calculations.

Table 4. Capacities and actuation of safety systems.

Parameter		Condition actuation
Reactor SCRAM signal	Low level of PZR	1.13 m
	High level of PZR	2.37 m
	Low level an any SG	2.13 m
	High level an any SG	2.68 m
	Containment pressure	> 0.13 MPa
PZR safety valve	Opening set point- #1	15.15 MPa
	Closing set point- #1	13.40 MPa
	Opening set point- #2	15.96 MPa
	Closing set point- #2	14.36 MPa
PZR spray system (determined by an over-pressure signal)	Opening set point	13.45 MPa
	Closing set point	13.17 MPa
Other boundary conditions	Primary circuit pressure (PI)	< 13.1 MPa
	Primary cooling rate	<115.0 kg/s

4. RESULTS

The point kinetic model is used to analyze the transient behavior as long as the reactivity change occurs in the entire core region by ATWS event. In this study is analyzed the loss of feed water in the SG#2 (in secondary loop, see Fig. 1). The AptPlot 6.8.0 software was used to plotting the parameter values in curves.

The results show that with the loss of feed water in SG#2 the water level drops, the pressure inside the primary circuit exceeds the design-basis pressure, as shown in Figs. 2 and 3. The primary peak pressure occurs at 640s.The limit for ATWS acceptance criterion is P < 110%. After that, the pressure of the reactor is decreased rapidly and the pressure of the steam generator is slightly decreased by actuation of safety valves of SG and PZR and spray system.

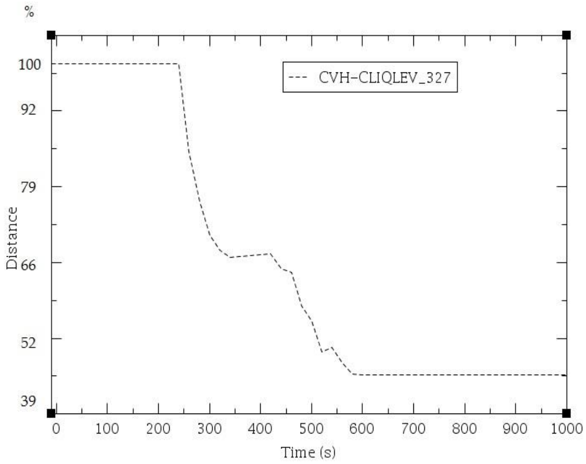

Figure 2. Collapsed water level in the SG downcomer.

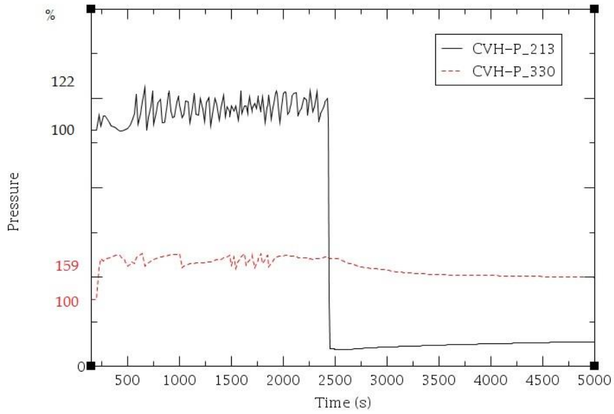

Figure 3. Primary and secondary pressure.

The water level of the reactor core is decreased continuously and the coolant temperature is increased (Fig. 4). If coolant temperature is increased, then the void fraction is increased and the coolant density is decreased. This is seen in Figs. 4-6.

Fig. 7 shows the collapsed water levels level for four volumes inside of RPV: core (CV 102), downcomer (CV 100), LH (CV 101) and upper head (CV 104). The reactor coolant inventory drops in the LH gradually until LH content is emptied at 2439 s when RPV failed.

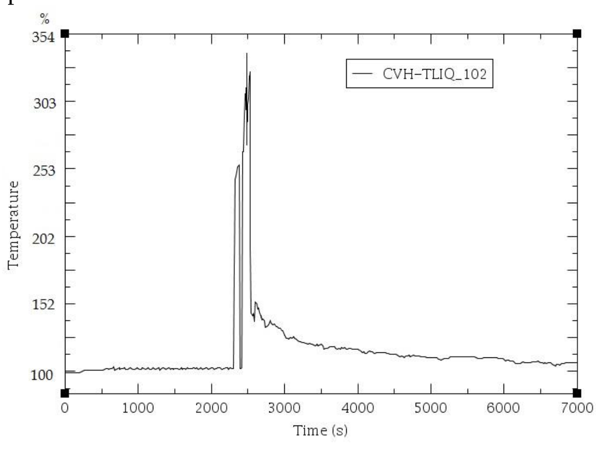

Figure 4.Coolant temperature in core.

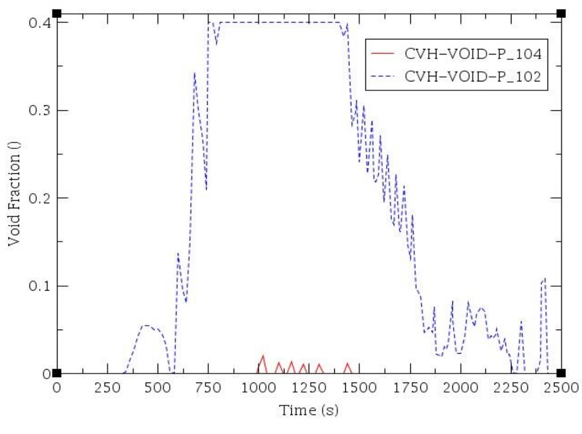

Figure 5. Void fraction in upper head (CV 104) and core (CV 102).

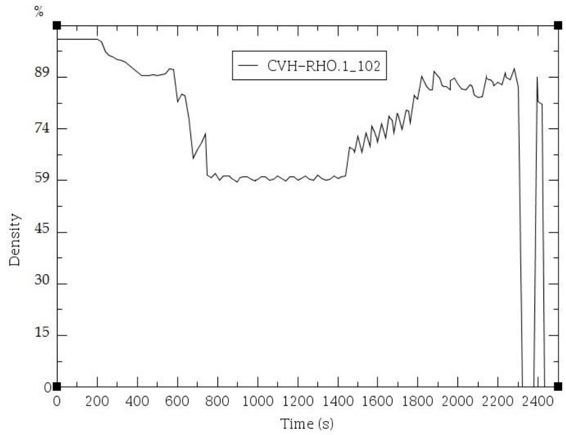

Figure 6. Coolant density.

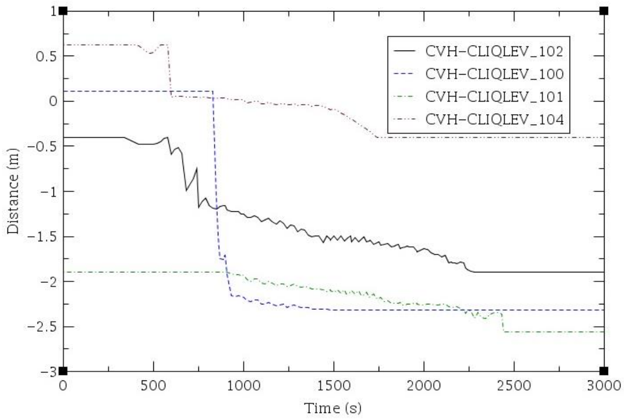

Figure 7. Collapsed water level in the reactor.

As the fuel temperature is increased, the fuel cladding temperature is increased due to the decreased cooling water level. The maximum peak fuel and cladding temperature during accident occurs at 1860 s after that dropped to 0 when fuel and cladding failed. The UO2 limitation temperature of PWR is 1204.4 °C (1477.55 K). This is seen in Fig. 8. Fuel cladding temperature integrity at elevated temperature is affected. The peak fuel cladding temperature during accident occurs at 1844 s. The cladding (Zircaloy) temperature was greater than 2500 K, which is the failure temperature of the cladding in MELCOR code.

The accumulated oxidation energy generated during the core degradation contributes to RPV failure (see Fig. 9). The oxidations begin at 1180 s.

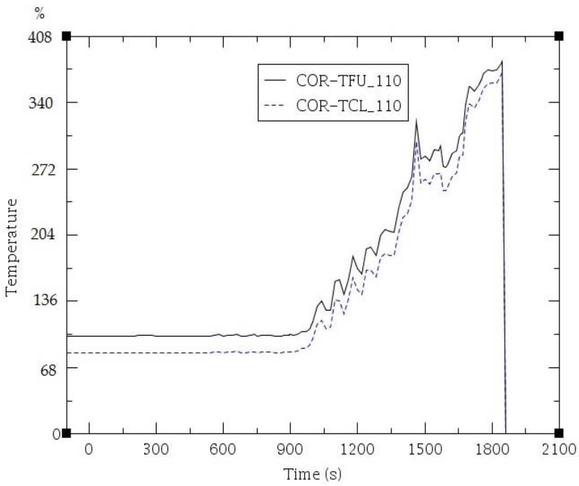

Figure 8. Temperature in fuel and cladding.

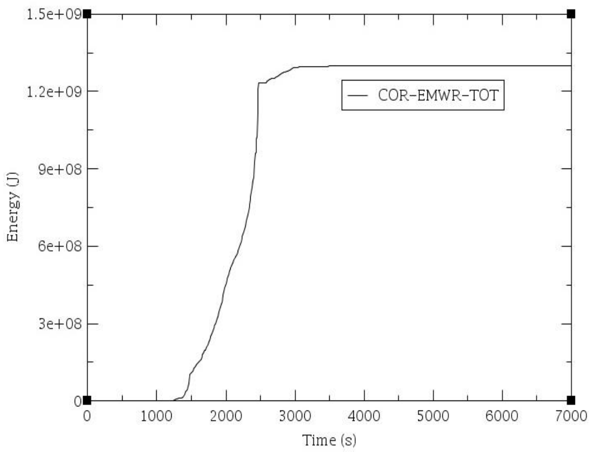

Figure 9. Total cumulative oxidation heat generated in core.

Fig. 10 shows the total mass in the reactor core reducing and the faster reducing to 2700 s is due to RPV melt. If RPV failure occurs due to insufficient core cooling, total mass of the core decreases due to melting and relocation into the reactor lower plenum at 2439 s (see Fig. 15). The components of the reactor core get significantly damaged; through chemical interactions and/or melting after 2700 s (see Fig. 10).

The hydrogen is produced from the zirconium (ZIRC) and stainless steel (SS) oxidation in the reactor core at 1180 s, as shown in Fig. 11. Hydrogen production shows correlation with core collapse.

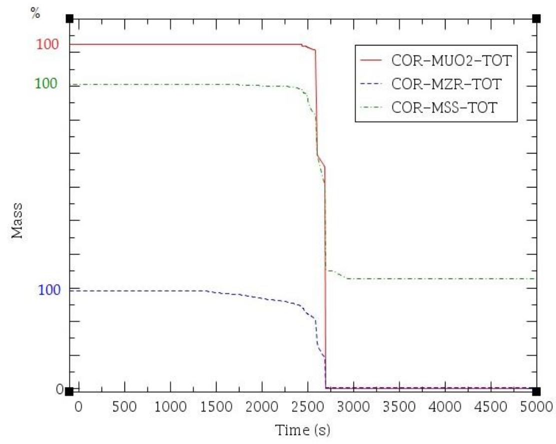

Figure 10. Total mass in the core.

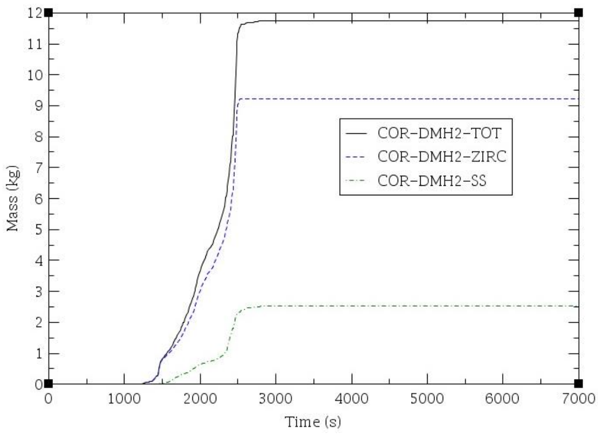

Figure 11. Hydrogen production by oxidation in the core.

The debris mass is being relocated to the lower head of containment (LHC) at 2439 s, as shown in Fig. 12. The heat transfer to the coolant is increased again after start of debris quench, and the heat transfer to structures in the reactor is increased slightly after the end of debris quench (see Fig. 13).

More energy accumulates in LHC during late stage of accident. More total energy means higher debris temperatures (see Fig. 13).The most material relocated from the core to the LHC had temperatures in the range of 4880 K to 6392 K (see Fig. 13).

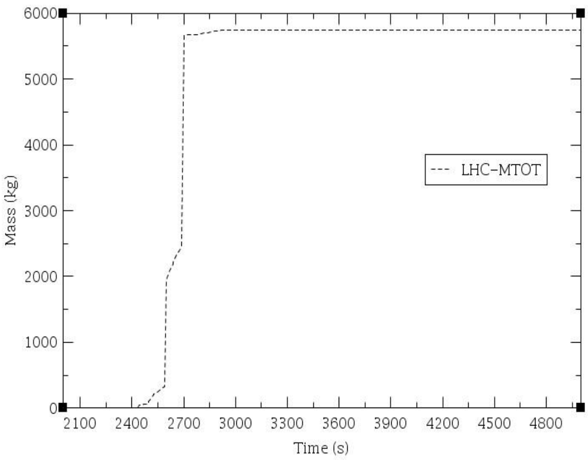

Figure 12. Total debris mass in LHC.

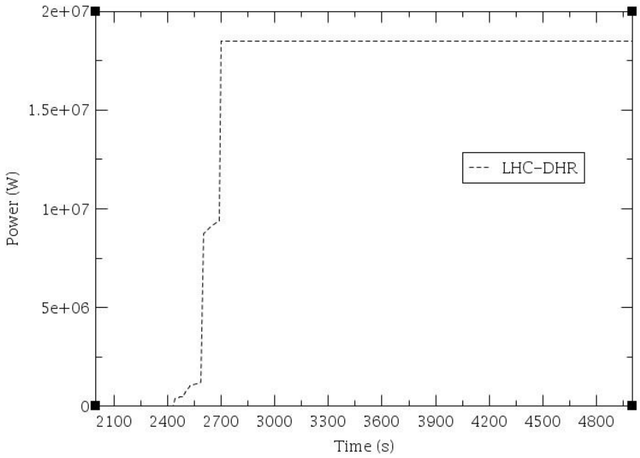

Figure 13. Total decay heat in LHC debris inventory.

Mass of the debris ejection (5898 kg) from failed RPV into the LHC is shown in Fig. 12, and Fig. 15 presents the total debris mass ejected through vessel breach. The debris mass of reactor core gradually increases corresponding to the timings of the RPV failure.

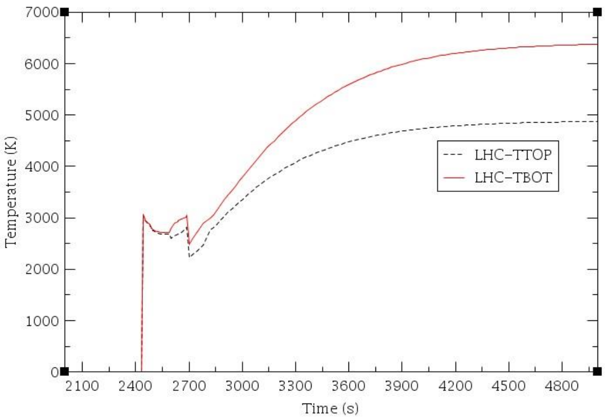

Figure 14. Debris temperature in bottom and top surface.

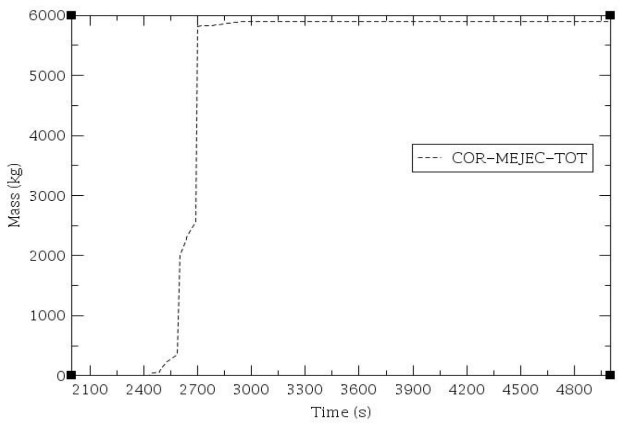

Figure 15. Total debris mass ejected through vessel breach.

The temperature is increased after hydrogen generation in RPV (solid line in Fig. 16) and in the containment (dash line in Fig. 16). The pressure is increased after hydrogen generation in containment (solid line in Fig. 17) and slowly after containment failure in shielding pool (dash line in Fig. 17).

In this accident, as shown in Fig. 18 the containment structural integrity is keeping, because not exist H2 releases outside the containment (in shielding pool).

Additional hydrogen in containment is produced after RPV failure, as shown in Fig. 19. The maximum amount of hydrogen produced in-vessel is 0.37 Kg, and ex-vessel is 11.61 Kg.

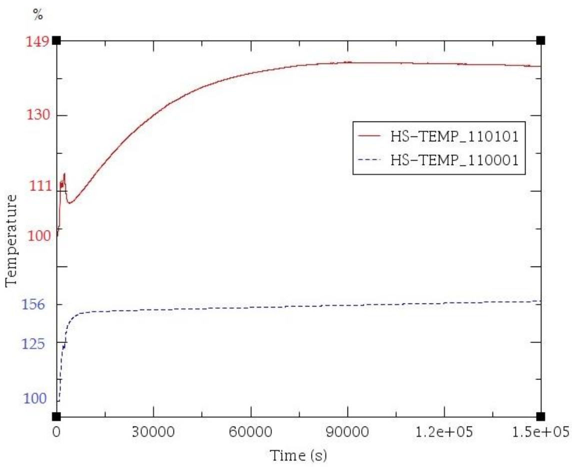

Figure 16. Vessel and containment temperature.

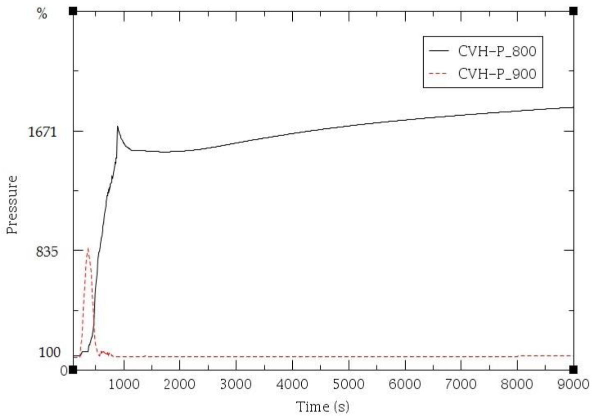

Figure 17. Pressure in containment and shielding pool.

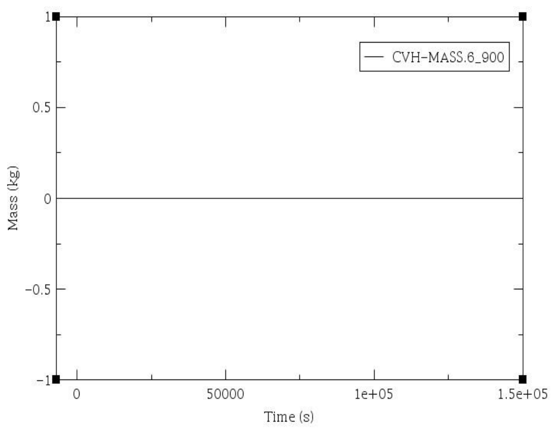

Figure 18. H2 mass in shielding pool.

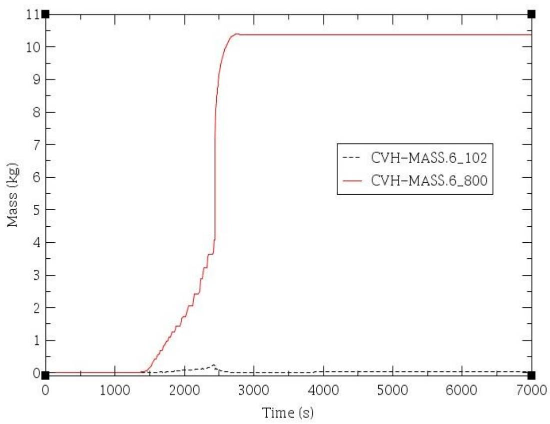

Figure 19. Hydrogen generation in core and containment.

4.1 Events Sequence

Table 5 presents the sequence of the events, systems and devices operation, for the LOFW+ATWS accident, obtained from the results presented in this section (Figs. 2-19).

Table 5. Sequence of events, systems and devices operation in LOFW+ATWS.

Time (s)	Event
0.0	Null transients starts
220	Feed water flow of SG#2 (in secondary loop) start decrease (Disconnection of FWPs)
340	Core water level reduction is initiated (Fig. 7)
520	Low level an any SG and no reactor shutdown (Fig. 2)
640	Primary pressure peak occurs (Fig. 3)
857	Maximum void fraction in the upper head (Fig. 5). Start of core uncover (Fig. 5)
1180	Start the oxidation of zirconium and H2 generation(Fig. 11)
1260	Begins to decrease the zirconium mass in the core (Fig. 10)
1440	Water level in downcomer is minimal (Fig. 7)
1844	Fuel temperature limit exceeded and Peak fuel cladding temperature occurs (Fig. 8)
1860	Fuel cladding collapsed (Fig. 8)
2320	Water level in core is minimal (Fig. 7)
2439	LH content is emptied (Fig. 7). RPV collapsed. Debris mass ejected through vessel breach (Fig. 15)
2700	Core material melt
2800	Maximum hydrogen produced by oxidation in core (Fig. 11)
1.5E+05	End of calculation

5. CONCLUSION

In the (LOFW+ATWS) study for a generic PWR NPP by MELCOR code, the peak fuel cladding (Zircaloy) temperature exceeded the safe margin of 1477.55 K and failed. The system pressure exceeds 110% of the design pressure during ATWS accident and RPV failed, then the reference PWR NPP not fulfillment of acceptance criteria.

Posteriorly, with the experience gained in this study – resulting in a better understanding of the phenomena involved in the LOFW+ATWS accident, and of the working with MELCOR code –, the group will simulate different types of severe accident, and will study how mitigate the consequences of accident.

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