DC-POWER AVAILABILITY ANALYSIS WITH MELCOR: INSIGHTS ON A SBO IN A 3-LOOP PWR-W NPP


RAFAEL BOCANEGRA
IDOM NUCLEAR SERVICES


GILBERTO VILLESCAS
IDOM NUCLEAR SERVICES

The Fukushima nuclear accident highlighted the fact that external events can significantly impact the restoration of AC power supply at nuclear power plants. This can result in longer recovery times than initially anticipated. As a result, it may be necessary to evaluate Extended Loss of Alternating Current Power (ELAP) conditions and determine whether existing batteries can operate for a sufficient duration to support core cooling. In this study, a MELCOR evaluation model of a generic 3-loop PWR is employed with the aim of assessing the effect of battery capability during an unmitigated station blackout scenario during the first 24 hours. A total of 5 cases have been analysed, varying the DC power duration depending on the battery's capability and the current consumption. The available safety systems considered have been limited to the accumulators and the auxiliary feed water system, which is controlled via valves fed by the DC power supply. The sensitivity analysis showed that the DC power supply is critical in managing the last heat sump available. Loss of DC power in the early stages of an SBO scenario may lead to a harder severe accident, containment failure, and even a major release of radionuclides to the environment.

INTRODUCTION

Since the last severe accident registered (1), the nuclear community has been focused on strengthening the Nuclear Power Plants (NPPs) capabilities to withstand scenarios beyond the design basis criteria. Stress Tests (2) and many other programs, such as (3), have been carried out to improve the reliability, and to maintain the high levels of safety of current NPPs. In fact, many countries now require for new plants such analyses in a newly defined 19 section in the Final Safety Assessment Report (FSAR) referred as "Severe Accident Evaluation" (4), where the Station BlackOut (SBO) is one of the scenarios to be assessed.

The SBO has been extensively studied during the last 10 years applied to all kind of plant designs, mostly to Pressurized Water Reactors (PWRs) (5) and Boiling Water Reactors (BWRs) (6) because its deployment all around the world. The motivations of these studies have been also diverse. Some of them have been focused on the PAR's efficacy (7) , combustible gas distribution (8), source term (9), forensic analyses (10), FLEX strategies (11), and many others.

In this study, an unmitigated SBO is analyzed in a generic 3-loop PWR-W with the MELCOR code, with the focus set on the DC capacity and its effects on the plant thermal-hydraulics. The document is structured firstly in a brief introduction, followed by the plant modelling with the MELCOR code. Then, the scenario is depicted, followed by the discussion of the results obtained. The document is closed with the main findings and remarks.

PLANT MODELING

The plant object of this study is a generic 3-loop PWR-W (12). The plant is modelled with the MELCOR code (13) following the SOARCA's best practices (14). MELCOR is a fully integrated, engineering-level computer code developed by Sandia National Laboratories for the U.S. Nuclear Regulatory Commission (NRC), with the primary purpose to estimate accident progression and source terms during postulated severe accidents in light water reactors.

The core is nodalized in the COR package into 5 radial ring and 10 axial levels, plus 2 additional axial level for representing the Lower Plenum (LP), and another axial level for the Upper Plenum (UP). An additional radial ring is employed for representing the LP downward region. The nodalization in the CVH package (Figure 1) is similar to that in the COR package but having half of the nodes, so that it is a CV for every radial subdivision, and a CV for every two axial COR nodes, making a total of 25 CVs for representing the core region, a CV for the bypass, another one for the LP, two for the UP and the upper head (UH), and one for the DC.

The Reactor Cooling System (RCS) is nodalized to allow natural circulation, as is also shown in Figure 1. Hot legs (HLs) are subdivided into 2 CVs to allow the modelling of an eventually Loss Of Coolant Accident (LOCA). The steam generators tubes are represented using 19 CVs to capture the natural circulation as a consequence of the temperature gradient. The pump suction (PS) is modelled using 2 CVs, being the reactor cooling pump (RCP) located in the CV close to the cold leg (CL). Finally, the CL is modelled using another 2 CVs which connect to the RPV downcomer (DC). The SG secondary side is modelled using 2 CVs, one for the SG downcomer and the other representing the rest of the SG. All the three main steam lines (MSLs) are represented with 2 CV each.

The accumulators (ACCs) are passive safety system which inject borated water into the reactor in case of an eventually loss of coolant accident (LOCA). In the evaluation model, every ACC is modeled using one CV filled with water and nitrogen at 4.6 bar. A one-way valve is employed for controlling the injection, which will occur when the RCS pressure falls below 4.6 bar.

The Auxiliary Feed Water System (AFWS) supplies water injection from the Condensate Storage Water Tank (CSWT) to the Steam Generators (SGs) to maintain the secondary level above the SGs tubes and below a maximum level, which is assumed to be slightly above 13 m from the SG bottom. This system consists in three circuits, being two of them powered by motor-pumps and the third by a turbo-pump. The steam feeding the AFWS turbo-pump is assumed to be supplied from the SG B and C MSLs (see Figure 2) as it is described in (15,16).

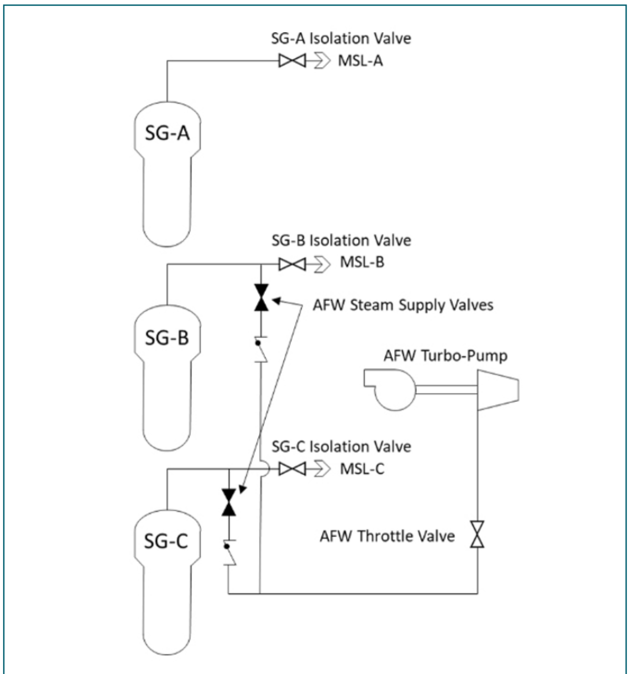

Figure 2. AFW Turbo-Pump Schem

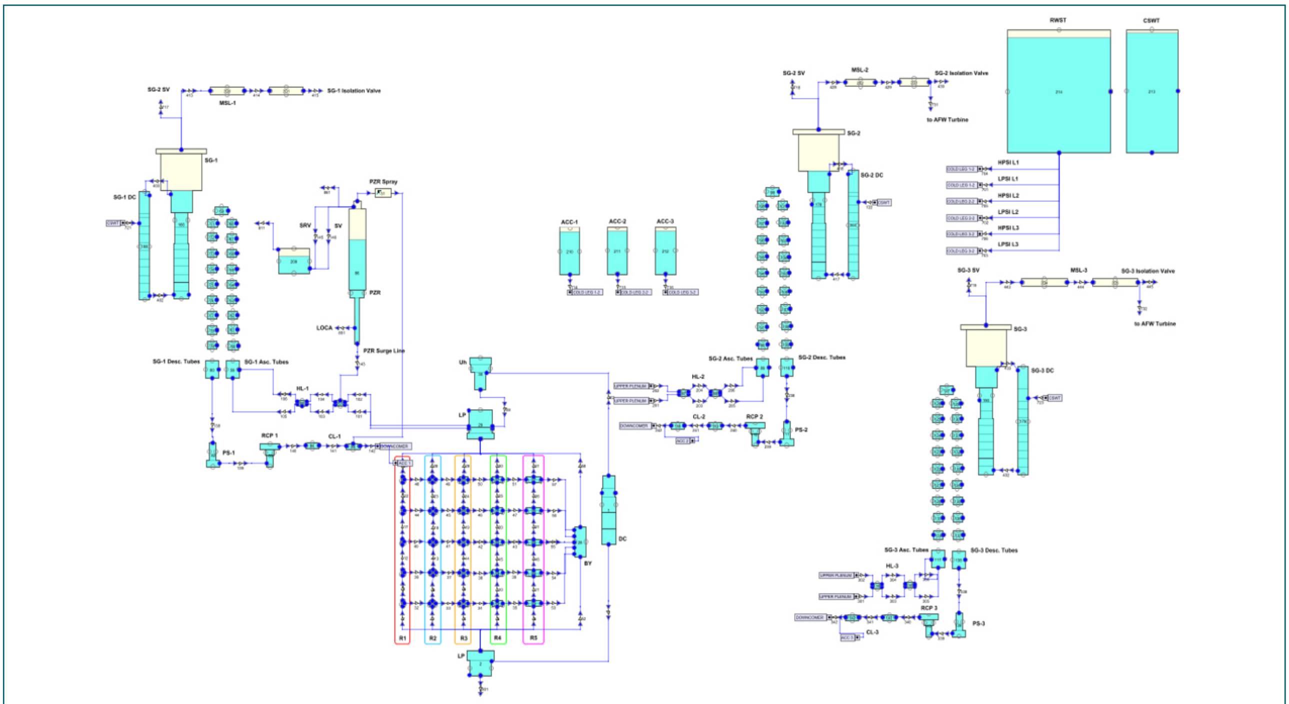

Figure 1. Nuclear Steam Supply System scheme (CVH) with MELCOR.

SCENARIO & RESULTS

The scenario analyzed is an unmitigated SBO with a duration of 24 h. Therefore, the only system assumed available for the accident mitigation are the accumulators (ACCs), and the Auxiliary Feed Water System (AFWS) by means of the turbine driven pump. It is also assumed that the AFWS will maintain the water level in the SGs until the batteries are depleted. At that time, the steam throttle valve is assumed fails at an open position, but the AFW steam supply valves are assumed to fail at a close position (see Figure 2).

Usually, Gen. 2 PWRs include at least two battery banks (12), one for each emergency trains (Safety injection, emergency diesel generators, etc.). Based on (17), it is assumed a capacity of 2600 Ah each. In addition, a third battery bank is assumed for powering all the AFW-related equipment, with an estimated capacity of 107 Ah and a constant consumption of around 20 A. For the selected scenario, the latest will be the focus of the analysis. In Table 1 can be seen the assumed consumption for each AFW-related equipment included in the analysis.

Battery Bank C			
System	**Consumption [A]**	**Capacity [Ah]**	**Duration [h]**
AFWS: Control	0.50	107	5.35
AFWS: Valves	12.00		
AFWS: Emergency lights	3.00		
AFWS: Relay Power Circuit	0.50		
AFWS: Emergency Signals	0.50		
AFWS: Instrumentation	0.50		
Circuit self-consumption	0.50		
125 VCC Battery Charger	2.50		
TOTAL	**20.0**		

Table 1. DC power assumed consumption in battery bank C

In Table 2 it is shown all the cases included in the analysis. In all of them, the DC power consumption is assumed to be 20.0 A, except for the case C, where some of the plant functions, such as the emergency lights and the battery charger, are assumed to be shut down resulting in a DC consumption of 14.5 A. In the case A, it is assumed a battery capacity 85.6 Ah, equivalent to an 80% of its nominal capacity in a way of accounting the battery aging. Case B considers the battery bank at nominal capacity (100 %). Case C is similar to the Case B except for the consumption rate, as has been explained above. In case D, it is assumed that one of the other two battery banks (bank A or B) is interconnected with the bank C to extend its capacity to 2707 Ah (2600 Ah + 107 Ah). Finally, the case E is similar to the case A, but reducing the battery capacity to 75 % instead 80 % and assuming that the AFW steam supply valves will remain “as is” at the moment the DC power fails (18).

Case	Battery capacity [Ah]	Consumption (A)	Time for depletion [h]
A	85.6	20.0	4.28
B	107.0	20.0	5.35
C	107.0	14.5	7.37
D	2707.0	20.0	135.35
E*	80.25	20.0	4.01

Table 2. Cases included in the analysis

The analysis has been focused on the thermal-hydraulics, core degradation and combustible gases release. Therefore, the figures of merit selected for the analysis are the RCS pressure (measured at the PZR), the RPV liquid level, the SGs liquid level and the in-vessel hydrogen release.

In Figure 1, it is shown the results which includes both the RCS and the Secondary Cooling System (SCS) figures of merit. In the right side, it can be seen the PZR pressure (in black) and the RPV liquid level (in red) ranging from the bottom of the lower plenum to the top of the upper plenum.

Setting the focus on the case A, the RCS pressure drops from its nominal value (15.5 MPa) to approximately 10 MPa as a consequence of the reactor SCRAM. Soon after, the pressure is further increased because of the SGs isolation and the Reactor Coolant Pumps (RCPs) shut down, which means the swap from forced to natural circulation. With the AFW start up, the RCS pressure is slowly decreased to 7.5 MPa approximately. As can be seen in Table 2, the batteries are assumed depleted at 4.28 hours. At this point, the water level in the SGs begin to drop (as can be seen in the right side of Figure 1), being totally dry at around 7 hours. As a consequence, the coolant reaches the saturation point, and the pressure increases until reaching the Safety Valve set point, which is assumed to be at 16 MPa. At this moment, the RCS inventory starts to be released to the PZR expansion tank. This tank includes a pressure disk designed to open when the pressure reaches around 0.8 MPa. This happens at 7.5 hours producing a release to the containment lower part, and therefore, a constant reduction of the RCS inventory which leads to the core uncovering at around 9 hours. Soon after, the cladding oxidation begins releasing hydrogen to the system and degrading the core integrity. The high pressure-temperature conditions inside the RCS produces the hot leg creep rupture at around 11 hours. At this moment, all the remaining inventory in the RCS is released to the containment producing a sharp decrease in the RPS pressure. This pressure drop allow the ACCs to inject borated water into the RPV, increasing the water level above the Top of Active Fuel (TAF). However, the water level drops again due to both, the residual and the cladding oxidation heat. Finally, the RPV fails at around 12.5 hours due to the core relocation on the RPV lower head, producing firstly the penetration failure followed by the RPV lower head major failure. After 24 hours, most of the reactor core has been relocated into the containment cavity and almost 700 kg of hydrogen has been produced because of the cladding oxidation.

Case B and C show the same trend but with different timing of events, as it was expected. In case B, where it was assumed a battery depletion at 5.35 hours, the core uncovery begun at 10 hours and the hot leg creep rupture at 12.5 hours. However, the amount of hydrogen generated during the cladding oxidation resulted slightly larger than that in the case A. Case C follows the same pattern, being the core uncovering at 12.5 hours and the creep rupture at 15 hours. The amount of in-vessel hydrogen generated was slightly smaller than the case A (~650 kg).

Case D shows a different pattern since the battery capacity assumed allowed maintain the DC power for more than 135 hours. However, the CSWT capacity, which feed the AFWS, is limited (~485 m^3). Therefore, the AFWS controlled operation is guaranteed for around 15 hours. After that, the sequence of events is similar to that in cases A, B and C with some exceptions. The core uncovery begun at ~20.6 hours but no creep rupture is produced during the first 24 hours. The amount of in-vessel hydrogen released at 24 hours was around 550 kg. No vessel failure was also registered during this time lapse.

The case E also shows a different trend. On the one hand, the AFWS steam admission valves are assumed to fail "as is". Therefore, when the batteries are depleted, the AFWS is kept injecting water into the SGs at its maximum capacity (only by means of the turbo-powered train). On the other hand, the steam feeding the AFWS turbine is extracted from the MSL-B and C, initiating an asymmetric transient when the AFWS control is loss as a consequence of the battery depletion.

The RCS pressure evolution is initially similar to the other cases previously commented. The differences begin when the batteries are depleted, moment where the pressure rapidly drops as a result of the higher injection rate of cold water into the SGs. As can be seen at the right-bottom of Figure 3, the SG-A water level starts to diverge from that the SG-B and C at 4 hours approx. The pressure drops until the CWST is depleted at 7.3 hours, moment at which the pressure starts to progressively rise. However, the SG-A becomes fully filled at around 10.5 hours, more than 3 after the AFWS stop injecting water on it. This happens as a result of the water expansion with the temperature increase and because of the water swelling with the steam bubbles. When the water level reached the main steam line, it is assumed to fail due to the weight of the water, releasing almost all the SG-A inventory into the containment.

Since almost all the inventory in the SG A is flashed out, there is a small pressure drop caused by the latent heat transferred

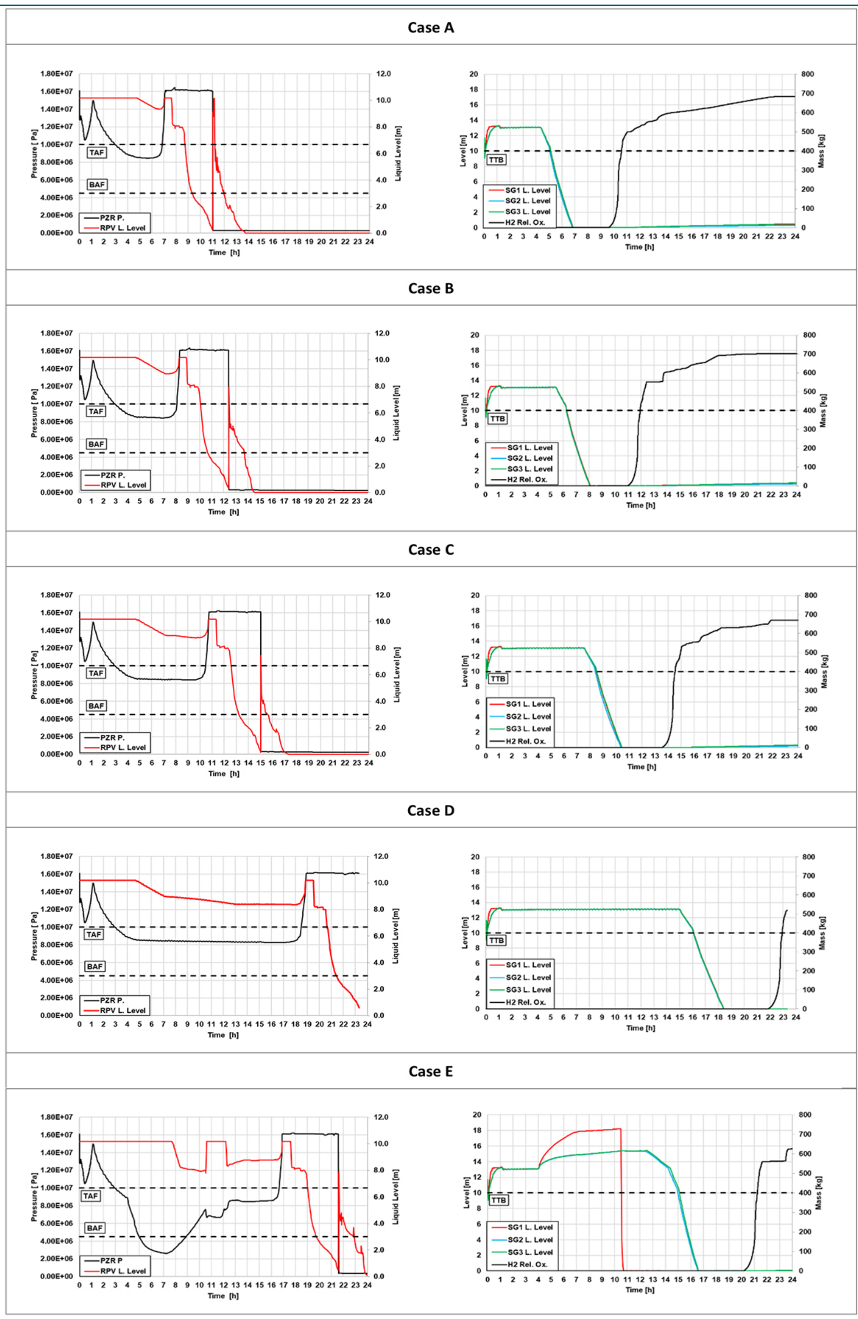

Figure 3. NSSS Results

from the RCS to the secondary. Both, SG B and C, also increase its water level but, since there is steam being directed to the AFWS turbine, level rise resulted much slower than that in the SG A. At around 12.5 hours, the CSWT is depleted, stopping the water injection into the SG B and C. The RCS pressure stabilizes at around 8.5 MPa until 16.5 hours, when the SG B and C become totally empty. At this moment, the pressure sharply increases until reaching the PZR safety valves pressure set point, and therefore, starting to discharge the inventory into the PZR expansion tank, and soon after to the containment. The water level in the RPV is constantly lowed, starting the core uncovery at 19 hours, and the degradation at 20 hours approx. The pressure is kept at this point until the hot leg creep rupture, releasing all the remaining inventory to the containment. The consequent pressure drop allows the ACCs to discharge into the RPV increasing the water level above the TOF. However, since there is no heat sink available to evacuate the residual heat from the core, the RPV water level drops again until it is totally dry. At 24 hours, a total of 630 kg of hydrogen has been generated as a consequence of the water-zirconium reaction. The RPV failed at 23.5 hours ejecting around 11 tons of corium to the reactor cavity at 24 hours of transient.

CONCLUSIONS

An evaluation model of a generic 3-loop PWR-W has been developed with the MELCOR code following the state-of-the-art best practices. The evaluation model has been employed for studying the impact of the DC power durability on the plant progression during an eventually station blackout with no mitigation during the first 24 hours. A total of 5 cases has been analyzed assuming different battery capacity and assumptions related with the availability of the auxiliary feed water system.

As a result, it can be said that the duration of the batteries is relevant to the timing of the main events during the transient. The larger the batteries capacity the longer it takes to get the core damaged. However, since no mitigation measures have been assumed, this statement is limited by other factors such as the CWST capacity. There is a threshold where the DC power becomes irrelevant since there is no water to inject into the steam generators, leading to the inevitable core damage.

In addition, although the cooling capability by means of the turbo powered AFWS extends the time available for acting with FLEX measures, it has to be accounted that an uncontrolled water injection in the secondary could lead to the failure of one of the SGs, initiating an MSLB event in the containment, converting the scenario in an asymmetric transient in the RCS, and loosing 1/3 of the core cooling capability. This MSLB is followed by a SB-LOCA produced through the PZR safety valves when the heat sump is lost, releasing the RCS inventory in the lower part of the containment, and finishing with a LB-LOCA as a consequence of the hot leg creep rupture. In those conditions, the containment pressure could surpass the design limits producing an eventually radionuclide release to the environment.

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