Safety Analyses with uncertainty quantification for fusion and fission nuclear power plants. Applications to EU DEMO fusion reactor and BWRs

Extended summary

The EUropean DEMOnstrating fusion power reactor (EU DEMO) Water Cooled Lithium Lead (WCLL) breeding blanket concept is currently in its pre-conceptual design phase by the EUROfusion consortium members. It aims to be the first tokamak fusion reactor to demonstrate the capability for net electricity production, tritium-self-sufficiency, and a lifetime plasma operation of several full-power years. The WCLL breeding blanket is one of the two concepts being studied for implementation in the EU DEMO reactor. This concept relies on the separate-cooled architecture, where the liquid metal is utilized exclusively as tritium breeder and neutron multiplier; the role of coolant is fulfilled by pressurized water.

Following the previous experience of the experimental International Thermonuclear Experimental Reactor (ITER), the design project of the DEMO reactor is constantly supported by safety studies for all the different breeding blanket concepts under investigation so that best performance within safety requirements are achieved. The primary goal of safety is to ensure that a nuclear reactor will not contribute significantly to individual and societal health risks. These risks stem mainly from the radioactive inventory inside the reactor, so this primary goal translates into the prevention of radioactive material releases toward the environment. A secondary but fundamental objective is to prevent fusion power plant main components damage. To deal with safety requirements within the DEMO project, a widely use of passive safety systems (a smart mix with active safety systems) will be made together with established safety principles, such as defense in depth and maintaining doses as low as reasonably achievable (ALARA).

The Ph.D. research activity had the main objective to characterize and quantify the safety and environmental aspects of the EU DEMO WCLL concept design, studying the reactor response to some of the most severe possible accident scenarios. Safety analyses have been performed with the fusion adapted versions of MELCOR code, to investigate the thermal-hydraulic behavior of DEMO main components and radioactive source term mobilization. Moreover, the performed safety analyses, supported by sensitivity studies, could be helpful to provide insight into physics and technology issues that need addressing to develop fusion as an optimal electricity generation alternative in the near future.

In this early development phase of the DEMO design, in the frame of the EUROfusion safety working project (WPSAE), a list of initiating events, which could start an accident sequence, has been identified through a Functional Failure Mode and Effect Analysis

(FFMEA). Many accident sequences enveloping from these initiating events have been comprehensively investigated.

The Ph.D. thesis can be divided into three main parts. The first part concerns a general description of the principal safety issues associated with fusion reactors and provides an overview of main EU DEMO components from a safety perspective. The second part contains preliminary safety studies relating to design basis accidents and hydrogen mitigation systems beyond design basis situations. The progression of design basis accidents has been simulated following a conservative approach considering the plant's passive and active accident mitigation capabilities. Four loss of coolant accident (LOCA) scenarios have been studied: in-vessel LOCA; ex-vessel LOCA; single in-vessel LOCA; and an in-box LOCA.

Concerning the multiple in-vessel LOCA, parametric analyses have been performed to determine the minimum flow area required by the suppression system pipework to limit the vacuum vessel (VV) pressure below the limit of 2 bar imposed as a requirement by safety. Moreover, because limiters could be introduced in the future design of the EU DEMO reactor to prevent the plasma from touching the breeding blankets plasma-facing components (PFC), the same parametric study has been performed to evaluate their accident mitigation effects. In this framework, a new vacuum vessel suppression system (VVPSS) concept has been proposed following the ITER experience. It is based on six separated suppression tanks located in the containment basement, one of them is dedicated to retaining small leakages. The pipework consists mainly of six bleed lines connecting the VV to the small leakage tank and five rupture disks line one for each suppression tank. To avoid steam and radioactive flows inside neutral beam ports, pipework connecting the vacuum vessel to the suppression system has been attached to the upper port. This last choice caused the necessity of a detailed nodalization of in-vessel volumes to model correctly steam flow path from VV to VVPSS and relative convective heat transfer effects between the modules' back supporting structure (BSS) and the steam flowing at high velocity in the interspace volume between the BSS and the VV.

Relatively to a simple in-vessel event involving the rupture of 10 first wall cooling pipes, two different simulations have been performed to evaluate downstream isolation valves' effects in terms of radioactive releases and thermal-hydraulic behavior of main DEMO components. In fact, the large number of downstream valves (isolation and Safety Relief Valves (SRV) to be installed could give rise to safety and reliability constraints.

Concerning ex-vessel LOCA events, a very unlikely double-ended pipe rupture is postulated in a coolant distributor ring of the EU DEMO reactor. The fusion power termination system is assumed to terminate the plasma burn with a mitigated disruption. Two different simulations have been performed related to failure in FW-PHTS and BZ-PHTS, respectively. However, due to its similarity, only the former results are described. These analyses aim to show that the accident consequences are within the safety requirements for tokamak building structures that must withstand large internal pressures and avoid significant leak rates into the environment. Because the tokamak building layout of the EU DEMO is currently in a preliminary design phase, parametric studies have been performed to support design activities. A preliminary but quite detailed model of the TCR was made to take into account steam condensation phenomena on TCR walls, being the only available effect for mitigating the overpressure in the TCR. In fact, no active systems for containment cooling are currently foreseen for DEMO.

A preliminary analysis of an in-box LOCA has been carried out to complete the wide range of DBA performed for the EU DEMO WCLL concept. This kind of accident has not been yet deeply investigated for fusion reactors because of the lack of multi-phase safety-related system codes able to deal with water and liquid metals. A Python script has been developed to overcome this code limitation for an external coupling of two MELCOR input decks working with different fluids. At each user-imposed time step, the information of one MELCOR run is extracted and used as feedback for the other input deck.

In the framework of BDBA an ex-vessel LOCA and a Loss of Flow Accident (LOFA) have been studied to show the robustness of the defense in depth approach and demonstrate that no cliff-edge effects occur in the safety analysis. In both the simulations, the failure of an active plasma shutdown system has been assumed as aggravating event. Differently from DBA, these accident analyses should be performed using best estimate assumptions, not conservative ones. At this purpose, the failure temperature of FW structure is increased from 1273 K to 1598 K. However, these parameter results very correlated, particularly for the ex-vessel LOCA simulation, with the amount of tritiated water other radioactive aerosols that could be released toward the external environment. Instead, preliminary safety analyses for the LOFA beyond the event highlighted that the major safety concern is not related to radiological releases, but to the huge pressurization of in-vessel components. For this reason, this accident has been simulated using a lower pressure setpoint for safety relief valves. Three different simulations have been performed by changing the number of FW channels affected by the rupture.

In-box LOCAs and other accidents involving a chemical reaction between hot steam and lead lithium could lead to the production of large amounts of hydrogen inside the tokamak vacuum chamber. To avoid flammable concentrations could be achieved, the production of hydrogen must be limited and adequately monitored. In particular, the simultaneous presence of hydrogen and dust in the VV volume enhances the risk of explosion. After a short description of possible technical solutions suitable for EU DEMO to mitigate hydrogen concentration, a preliminary accident study involving passive autocatalytic recombiners (PARs) is reported. The MELCOR ESF package has been activated to simulate the presence of PARs directly installed in the atmosphere of the VVPSS suppression tanks.

Successively, in the third part, sensitivity and uncertainty analyses are reported. Because severe accidents in both fission and fusion power plants involve a wide range of uncertain phenomena and parameters, sensitivity and uncertainty analysis have to be performed to evaluate the influence of input parameters on selected figures-of-merit (FoMs). For this purpose, a Python interface has been developed to allow the interaction between RAVEN and MELCOR. The Python interface allows to perturb all the parameters accessible through the MELCOR input deck. In such a way, RAVEN is capable of investigating the system response and the input space using sampling schemes. Two sensitivity and uncertainty analyses have been performed, with applications to the EU DEMO reactor and Fukushima Daiichi unit 3 power plant.