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Editorial

In-Vessel and Ex-Vessel Corium Stabilization in Light Water Reactor

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Since the Fukushima accidents, severe accident mitigation as a last protective barrier of nuclear disasters is of a great concern for the public acceptance. A lot of efforts have been made to resolve the severe accident mitigation issues during the several decades. Corium stabilization inside and outside the reactor vessel, one of the issues with a great progress, is a key to securing the safety of Light Water Reactors from the point of view of severe accident mitigation and containment integrity.

Regarding the corium stabilization inside reactor vessel, a concept of IVMR (In-Vessel Melt Retention) is generally used for the Light Water Reactors as a most effective method. The IVMR relies on cooling of the outside surface of the reactor lower vessel by naturally circulating water so-called ERVC (External Reactor Vessel Cooling). In the ERVC, cooling water is supplied to the exterior wall of the reactor vessel in case of severe accident. This strategy was firstly used in the VVER-440 reactor and now it is adopted for a number of higher power reactor designs such as AP1000 and APR1400. For the assessment of successful IVMR, it is important how to accurately estimate the thermal heat load inside the reactor vessel and the heat removal rate outside the reactor vessel. There are several issues for the estimation: corium relocation to lower head, lower head debris bed behaviour, lower head molten pool behaviour, thermal and mechanical loadings and behaviour of structures including the lower head, and external vessel cooling rate.

If the IVMR is not maintained, failure of the reactor lower head starts as the final stage of the in-vessel accident progression. The initial situation is characterized by a molten corium pool in the lower head. Important parameters for the assessment of lower head failure are temperature distribution of vessel wall, internal pressure, dead weight of the vessel wall and the melt pool, thermochemical attack of the corium, and mechanical behaviour of reactor vessel. If it will fail, the location and time of failure have to be evaluated. For the PWRs, ICI (In-Core-Instrumentation) penetration tubes at the reactor lower head are regarded as the most vulnerable parts. BWRs include several venerable parts such as CRGT (Control Rod Guide Tubes), ICMGT (In-Core Monitoring Guide Tubes), and drainage tubes.

Regarding the corium stabilization outside reactor vessel, there are two preferred methods used for the operating Light Water Reactors: pouring water on top of the melt pool which is discharged from the failed vessel into the normally dry cavity and discharging melt into the wet cavity which is flooded with water before the arrival of the melt. The melt should be cooled down lower than the concrete ablation temperature in order to avoid erosion of concrete leading to basemat melt-through and containment failure by generation of gases from concrete ablation. This is caused by so-called MCCI (Molten Core Concrete Interaction). Melt-water interaction is relatively small for the former method, because an insulating crust is formed all around the melt pool after the initial contact with water. For the latter method, the oxidic-ceramic melt breaks into melt particles of small sizes, which results in increase of the contacting surface area between the original melt and the water. This can lead to a steam explosion.

For the advanced reactor, a concept of core catcher is employed to cope with the corium stabilization outside reactor vessel. Ex-vessel melt is retained and cooled in the core catcher. The core catcher is currently installed for EPR and VVER-1000.

The paper "Effect of Subcooling on Pool Boiling of Water from Sintered Copper Microporous Coating at Different Orientations" presents experimental results of a subcooling effect on pool boiling heat transfer using a copper microporous coating water a comparison with those of a plain surface. A HTCMC (High-temperature Thermally Conductive Microporous Coating) was made by sintering copper powder. The nucleate boiling heat transfer did not change much with the degree of subcooling for both the HTCMC and the plain surface. However, the CHF (Critical Heat Flux) linearly increased with the similar rates of 60 kW/m^2 per degree for both surfaces, so the CHF values of the HTCMC stayed about ~1,000 kW/m^2 higher than those of the plain surface throughout the subcooling. The results showed a possibility of using the microporous coating at an outer reactor vessel wall to enhance cooling performance and CHF when in-vessel retention through external reactor vessel wall (IVR-ERVC) strategy is applied as a severe accident mitigation methodology.

The paper "Analysis of Steam Explosion under Conditions of Partially Flooded Cavity and Submerged Reactor Vessel" presents the effects of the free-fall of corium on a steam explosion by benchmarking previous experimental cases. Two premixing experiments presenting partially flooded cavity conditions and submerged-RV conditions in the TROI (Test for Real cOrium Interaction with water) facility were modelled by the TEXAS-V code. The impulse of a steam explosion under the condition of a corium jet falling into water without a free-fall height is bigger than that under a free-fall height. Larger fragmented mass of corium in an explosion phase and smaller steam production under the condition of no free-fall height resulted in bigger impulse of steam explosion. The distribution of void fractions was similar in both the experiments and simulations. The effect of a large bubble around a melt jet is an important element to be simulated well.

The paper "Effect of Molten Corium Behavior Uncertainty on the Severe Accident Progress" presents results of an uncertainty and sensitivity analysis of a severe accident progress in a Korean Optimized Power Reactor 1000 MWe (OPR1000) using the MELCOR integrated severe accident code. Uncertainty of severe accident progression phenomena, namely, the time of melt relocation, the time of lower plenum dryout, and the time of the reactor pressure vessel and containment failure in MELCOR simulations, induced by 5 selected uncertainty input parameters has been investigated. In addition, a sensitivity analysis applying a rank regression technique has been performed in order to identify those uncertain parameters, which mainly contribute to the uncertainty of the selected MELCOR results. The results of rank regression analysis showed that 2 out of 5 selected uncertain parameters, i.e., the zircaloy melt breakout temperature and molten clad drainage rate, have the largest contribution to the uncertainty of the time of melt relocation and the time of lower plenum dryout in the MELCOR results.

The paper "A Conceptual Approach to Eliminate Bypass Release of Fission Products by In-Containment Relief Valve under SGTR Accident" presents the conceptual design of the ICRV (In-Containment Relief Valve) as effective means to mitigate the release to the environment of fission products from the SGTR (Steam Generator Tube Rupture) accident. To show the effectiveness of ICRV concept, the simulation of OPR1000 by using MELCOR code was carried out. It was assumed that the steam of the secondary system was released to the upper dome of the containment and the RDT (Ractor Drain Tank) using ICRV. The results show that the radioactive nuclides were not released to the environment even though the containment pressure increases up to 1.2 MPa. To minimize the negative effects of pressurization of the containment, the steam release by the ICRV linked to the RDT and cavity flooding was also simulated. Because the overpressurization of containment is due to heat of ex-vessel corium, it turned out that cavity flooding was effective for depressurization. Finally, authors made a conclusion that the conceptual design of the ICRV is effective to mitigate the SGTR accident.

The paper "Modelling of Severe Accident and In-Vessel Melt Retention Possibilities in BWR Type Reactor" deals with application of in-vessel melt retention, which is in use as a milestone of SAM strategy in several PWR and VVER reactors, in a BWR-5 reactor having quite different design of the lower head, in particular, a forest of control rod and instrumentation penetrations. A computer code RELAP/SCDAPSIM MOD 3.4 and a full plant model of a ~2000 MW thermal power BWR reactor were used to model a large break Loss of Coolant Accident (LOCA) in which total failure of cooling water injection was postulated. A full accident sequence, up to corium relocation into the lower head and molten pool formation, is calculated with a particular attention to vessel external cooling by water. In the last part, different ex-vessel heat transfer models were used and compared, and it was concluded that the implemented heat transfer correlations of the COUPLE module are more accurate than that of the RELAP5 based module. As the used code does not take into account stratified molten pool with the top metal layer, the analytical study was conducted to estimate possible focusing effect in the location of top metal layer. The results show that at prototypic corium masses and volumetric heat in the RPV lower head a significant amount of steel should be in the molten pool in order to keep the heat fluxes on the external vessel surface below CHF (1.06 MW/m^2). It was concluded that detailed evaluation of the steel mass relocated to the lower head during severe accident is not possible with the existing RELAP/SCDAPSIM models, which are recommended to be further developed in order to evaluate complex phenomena in the debris bed and to model heat and mass transfer within a molten pool having 2 or 3 immiscible layers.

The main purpose of this special issue is to provide high quality articles containing latest research achievements on in-vessel and ex-vessel corium stabilization in Light Water Reactors. We believe that interesting information on this topic will be supplied to the readers.

Conflicts of Interest

The authors declare that there are no conflicts of interest regarding the publication of this paper.

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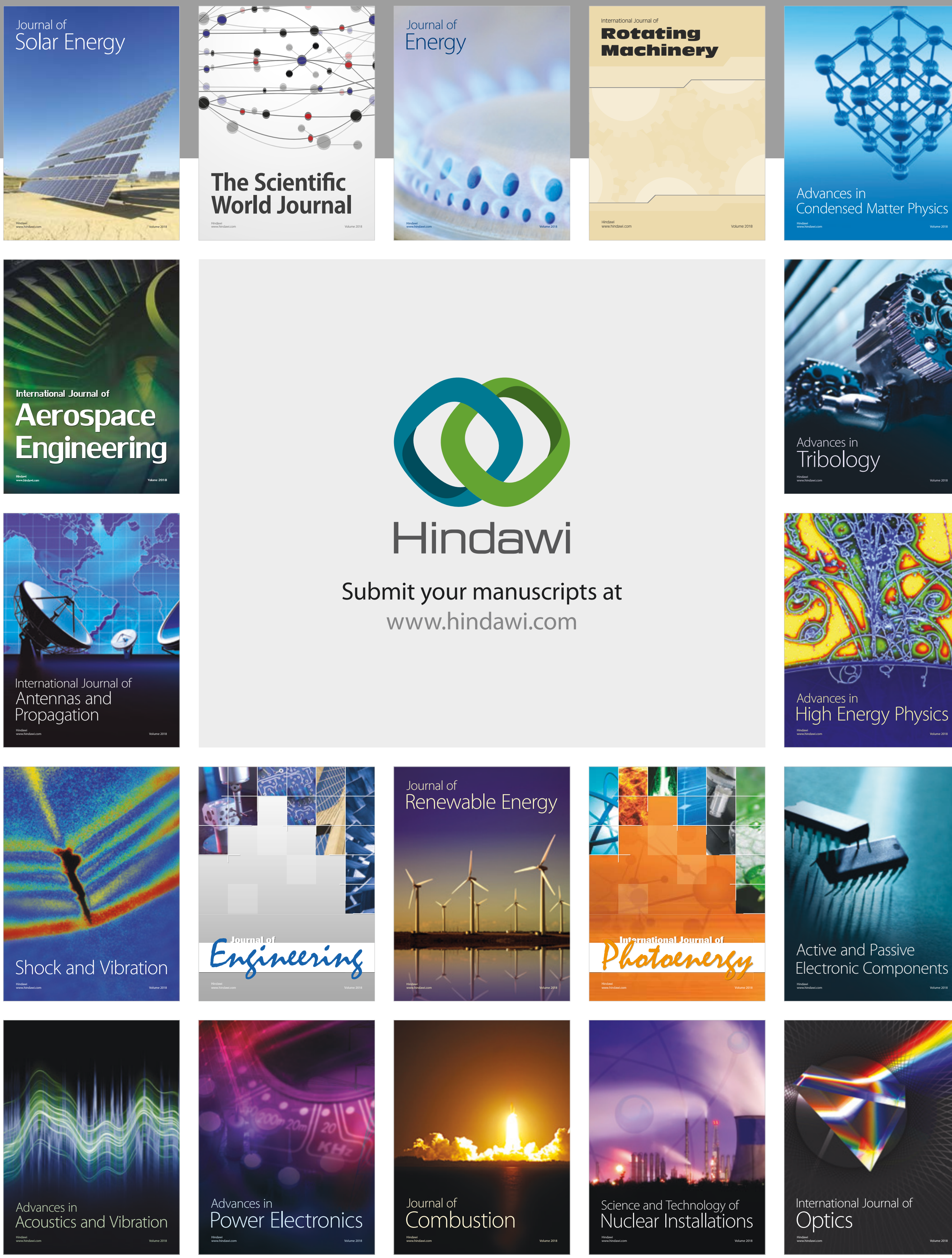
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