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Editorial

Accident Modeling and Analysis of Nuclear Reactors

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Nuclear energy has played and will continue playing a key role in the fight against climate change due to its extremely low green-house gas emissions, and high energy density and reliability. Up to August 2022, there were 439 reactor units in operation worldwide providing around 30% of the world's low carbon electricity. Now, many countries including China, France and the USA have taken nuclear energy as one important approach to achieve carbon neutrality by 2050–2060. Nuclear safety issues, however, are a great challenge for the development of nuclear power plants (NPPs). Three severe nuclear power plant accidents, i.e., the Fukushima nuclear disaster (2011), Chernobyl disaster (1986), and Three Mile Island accident (1979), have caused great loss of life and large monetary costs for remediation work due to the release of high-level radioactivity. Accident modeling and analysis of nuclear reactors has become paramount for reactor system design and license application. The probability of an accident and the magnitude of its consequences are two dimensions of nuclear accident study. They supplement each other, codetermining the importance degree of the accident in safety design. As computing technologies constantly improve, accident modeling and analysis of nuclear reactors approach real accident conditions with little approximation and can provide more reliable accident assessments. In this Special Issue entitled, "Accident modeling and analysis of nuclear reactors", methodologies and models developed for determining accident probability and consequence analysis are comprehensively reviewed from different perspectives to outline the development trends.

This Special Issue contains eight papers and covers different technological aspects, ranging from external accident analysis of earthquakes [1] to internal accident analysis including station blackouts [2], steam explosion [3] and hydrogen explosion [4]; from a fuzzy fault tree method [5] to a dynamic probabilistic risk assessment model [6] applied for accident probability analysis; and from thermal-hydraulic modeling [7] to fuel failure modeling for accident simulation [8]. The contents of the selected papers are summarized in the following paragraphs.

Earthquakes can damage the structural components of NPPs, inducing a severe accident externally. They are regarded as one of the most severe natural disasters that threaten the safety of NPPs. To investigate the influence of different vertical equivalent damping ratios of a 3-dimensional combined isolation bearing (3D-CIB) on seismic response and isolation effectiveness, Zhu et al. [1] proposed an earthquake simulation model by coupling the commercial finite element software ANSYS and bearing element-type model COMBIN40. ANSYS was employed to establish the model of a 3D base-isolated nuclear reactor building, and COMBIN40 was adopted to simulate the horizontal behavior of 3D-CIB. The authors found that increasing the vertical equivalent damping ratio would benefit isolation effectiveness, and a 3D-CIB with a vertical equivalent damping ratio of 15~20% is appropriate and acceptable from the viewpoint of displacement control.

In the work of Frano et al. [2], an internal severe nuclear accident caused by a station blackout (SBO) was modeled by assuming that all the emergency diesel generators were not available. Severe consequences were caused since active injection systems, including low- and high-head safety injections, failed to activate due to the lack of power. To evaluate the performance of an aged reactor pressure vessel (RPV) during this accident, MELCOR and MARC codes were coupled in simulation with the former, providing the 3D temperature and pressure distribution to the latter as the boundary conditions. MARC was then applied to perform thermo-mechanical analysis of the material considering the effects of elastic deformation, plasticity, creep and thermal degradation/aging. This analysis demonstrated that time-dependent phenomena play an important role in the structural integrity of RPVs, and an aged PRV system may collapse earlier and in less time under an SBO accident.

Steam explosion and hydrogen explosion are two other internal severe nuclear accidents that may destroy the reactor pressure vessel and cause radiation leakage. Both of them result from core fuel melt and its interaction with water or steam, but occur in different mechanisms. Steam explosion is primarily caused by the rapid energy transfer from the melt debris of fuel to the coolant as they make contact with each other. A large amount of vapor is produced within a very short time, subsequently leading to high pressure and fast expansion of vapor. Based on the NPP of the APR-1400 in operation in Korea, Park et al. [3] conducted an analysis on a steam explosion that happened in the reactor cavity as the melt was discharged from the core to the outside cooling water. The authors coupled the steam explosion code, transient analysis code for explosive reactions (TRACER-II) and finite element (FE) model to perform the structural integrity evaluation of the reactor cavity. TRACER-II has the ability to simulate the processes of steam explosion by solving the equations of continuity, momentum and energy involved in multiphase flow. It provides the data of the pressure wave to the FE model, in which the stress and ductile failure strain on the reactor cavity, liner plate and rebar are evaluated for integrity. The internal severe nuclear accident of a hydrogen explosion arises from chemical oxidization of zirconium (contained in the fuel rod cladding) with steam or water at a high temperature ranging from around 1000 °C to 1500 °C. Kang et al. [4] established a multidimensional hydrogen explosion analysis system based on the codes of MAAP, GASFLOW and COM3D, in which GASFLOW calculated the hydrogen distribution in the containment with a hydrogen source evaluated by the MAAP during severe accidents. Combining the data from ENACCEF and THAI tests, hydrogen flame acceleration was simulated by COM3D. With this developed model, the integrity of APR-1400's containment was evaluated under a hydrogen explosion accident triggered by an SBO accident.

The probability of accident reflects the reliability of a nuclear energy system and determines the design of the safety level for the system. The fuzzy fault tree analysis method is one quantitative method that is widely used for probabilistic safety assessment (PSA). It can mathematically model qualitative data and has the ability to perform the PSA in a quantitative way even if component failure data are insufficient or unavailable. Based on this method, Hermansyah et al. [5] conducted a PSA for a loss-of-coolant accident and loss-of-coolant flow accident in the primary cooling system of a research reactor, the G.A. Siwabessy Multipurpose Reactor in Indonesia. To address the problem of lacking sufficient historical failure data to statistically assess the component reliability, the qualitative judgments of experts were collected and converted to quantitative values of basic event failure probability by using the fuzzy fault tree analysis method. In a review of integral PWR-type small modular reactors, Zeliang et al. [6] highlighted that several issues and challenges remain in the numerical modeling of reliability assessments of passive safety systems. The dynamic PRA (probabilistic risk assessment) method was recommended to overcome the shortcomings of existing methodologies; for example, system dynamics, time, dynamic interactions and failure event ordering cannot be explicitly accounted for.

Thermal-hydraulic and fuel failure modeling are key technologies in the analysis of a nuclear energy system accident. To simulate the transient behavior of natural convection in a cylindrical containment vessel (CIGMA) under a severe accident, Hamdani et al. [7] proposed a transient 3D model based on OpenFOAM. With unsteady thermal boundary conditions, four turbulence models, including the standard k-ε, standard k-ω, k-ω shear stress transport (SST), and low-Reynolds k-ε Launder–Sharma models, were studied and compared to the experimental data. The authors observed that the k-ω SST model showed a better prediction compared to the other turbulence models. They also pointed out that the conjugate heat transfer in the internal structure inside the containment vessel must be modeled accurately to improve the prediction of temperature and pressure. In the work of Shah et al. [8], a dynamic risk assessment method for fuel failure was put forward by coupling the RAVEN (dynamic probabilistic risk assessment tool), RELAP5 (thermal-hydraulic code) and FAMs (fuel failure analysis modules). With the developed method, an evaluation on the risks of accident-tolerant fuel (ATF) was conducted with and without flexible coping strategies (FLEX) under different accident scenarios, including high-pressure accident scenarios and low-pressure accident scenarios. The results indicated that the SiC-type ATF with FLEX has the highest safety margin.

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