World Academy of Science, Engineering and Technology
International Journal of Energy and Power Engineering
Vol:10, No:10, 2016

The Model Establishment and Analysis of TRACE/MELCOR for Kuosheng Nuclear Power Plant Spent Fuel Pool

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***Abstract*—**Kuosheng nuclear power plant (NPP) is a BWR/6 plant in Taiwan. There is more concern for the safety of NPPs in Taiwan after Japan Fukushima NPP disaster occurred. Hence, in order to estimate the safety of Kuosheng NPP spent fuel pool (SFP), by using TRACE, MELCOR, and SNAP codes, the safety analysis of Kuosheng NPP SFP was performed. There were two main steps in this research. First, the Kuosheng NPP SFP models were established. Second, the transient analysis of Kuosheng SFP was done by TRACE and MELCOR under the cooling system failure condition (Fukushima-like condition). The results showed that the calculations of MELCOR and TRACE were very similar in this case, and the fuel uncover happened roughly at 4^{th} day after the failure of cooling system. The above results indicated that Kuosheng NPP SFP may be unsafe in the case of long-term SBO situation. In addition, future calculations were needed to be done by the other codes like FRAPTRAN for the cladding calculations.

***Keywords*—**TRACE, MELCOR, SNAP, spent fuel pool.

I. Introduction

KUOSHENG NPP is a type of BWR/6 designed and built by General Electric in Taiwan. There are two units in Kuosheng NPP. There are two loops of recirculation piping and four main steam lines for each unit. After the project of SPU (Stretch Power Uprate), the operating power of Kuosheng NPP is 3001 MWt.

U.S. NRC has developed the advanced thermal-hydraulic code named TRACE for NPP thermal hydraulic analysis [1], [2]. According to the TRACE's user manual [1], TRACE is the product of a long term effort to combine the capabilities of the NRC's four main systems codes (TRAC-P, TRAC-B, RELAP5 and RAMONA) into one modernized computational tool. In addition, the 3-D geometry model of reactor vessel, which is one of the representative features of TRACE, can support a more accurate and detailed safety analysis of NPPs. According to the user and assessment manual [1], [2], TRACE also provides greater simulation capability than the previous codes (TRAC-P, TRAC-B, RELAP5 and RAMONA), especially for events like LOCA. Additionally, TRACE was used to simulate the SFP of Fukushima NPP according to Sandia National Laboratories report [3].

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MELCOR is developed by Sandia National Laboratories [4]. MELCOR can model the progression of severe accidents in NPPs. MELCOR can treat the broad spectrum of severe accident phenomena in NPPs. These include thermal-hydraulic responses in the reactor coolant system, reactor cavity, containment, and confinement buildings; core heatup, degradation, and relocation; core-concrete attack; hydrogen production, transport, and combustion; fission product release and transport behavior. According to Carbajo [5], MELCOR was used to perform the study of SFP of Fukushima Dai-ichi Unit 4. Könönen [6] used MELCOR to establish the SFP model of a Nordic BWR. The loss-of-pool-cooling accidents were simulated and analyzed by this model. Reference [7] also presents the study and simulation of the SFP of MELCOR under the Beyond-Design-Basis Earthquake condition. The above studies indicate that MELCOR is capable of handling the simulation of the SFP.

SNAP (Symbolic Nuclear Analysis Program), a graphic user interface program that processes the inputs and outputs of TRACE and MELCOR is also developed by U.S. NRC.

Two steps were in this research, and the first step was the establishment of Kuosheng NPP SFP TRACE/SNAP and MELCOR/SNAP models. The next step was the transient analysis under the SFP cooling system failure condition (Fukushima-like condition). The above models performed this transient analysis and study.

II. Methodology

The safety analysis methodology of Kuosheng NPP SFP is as follows: First, the data of Kuosheng NPP SFP were collected [8]. Second, the TRACE/SNAP model was established in order to perform the thermal-hydraulic analysis. Third, the MELCOR/SNAP model was built in order to run the severe accident analysis. In this step, the amounts of hydrogen generation can be calculated. The geometry of Kuosheng NPP SFP was 11.16 m × 11 m × 12.19 m, and the initial condition was 311 K (water temperature) / 1.013 × 10^5 Pa. The total power of the fuels was roughly 10.26 MWt initially. Total 4856 fuel bundles were in SFP.

Fig. 1 shows TRACE/SNAP model of SFP. In this model, the 3-D vessel component of TRACE was used to simulate the pool. Channel component of TRACE is a 1-D component and can simulate full length fuel rods, partial length fuel rods and water rods. There were two channel components which were used to simulate the fuel bundles in this paper. The hottest fuel rod was in channel 53. The heat source of the SFP was the decay heat of the fuels and was simulated by a power component of TRACE, which used the power table to simulate the power varying during the transient. The decay heat data are shown in Fig. 2. This model also had the simulation of the heat conduction between the racks of the fuels and the pool. One heat structure component of TRACE was used to simulate the heat exchange from SFP to the fuels' racks.

Fig. 3 shows the MELCOR/SNAP model of SFP. This model includes one core component, ten control volume components, 13 heat structure components, and several control/tabular function components. According to the MELCOR manual [4], two new features to the core component (COR package) that are specific to SFP modeling: (1) a rack component, which permits modeling of a SFP racks, and (2) an enhanced air oxidation kinetics model. The SFP rack component permits separate modeling of the rack and radiative heat transfer between the rack and the existing COR components. The air oxidation kinetics model predicts the transition to breakaway oxidation kinetics in air environments on a node-by-node basis. Therefore, the core component was used to model the material of racks and fuel assembles in this paper. The water of SFP was modeled by using the control volume components (CVH package). The core component was divided into ten axial levels and four radial rings. The fuels were divided into eight axial nodes which were in the level 3~10 of core component and No. 2, 102, 202 CVH components.

III. Results

The heat source of SFP was the decay heat of the fuels. Fig. 2 shows the total decay heat of the fuels. The transient analysis under the SFP cooling system failure condition (Fukushima-like condition) was performed in this study. Due to the cooling system of SFP failed, no water was added into the SFP during the transient. The heat of the fuels was removed by the evaporation of pool water before the uncovered of the fuels occurred. After the uncovered of the fuels occurred, the safety issue of the fuels may generate.

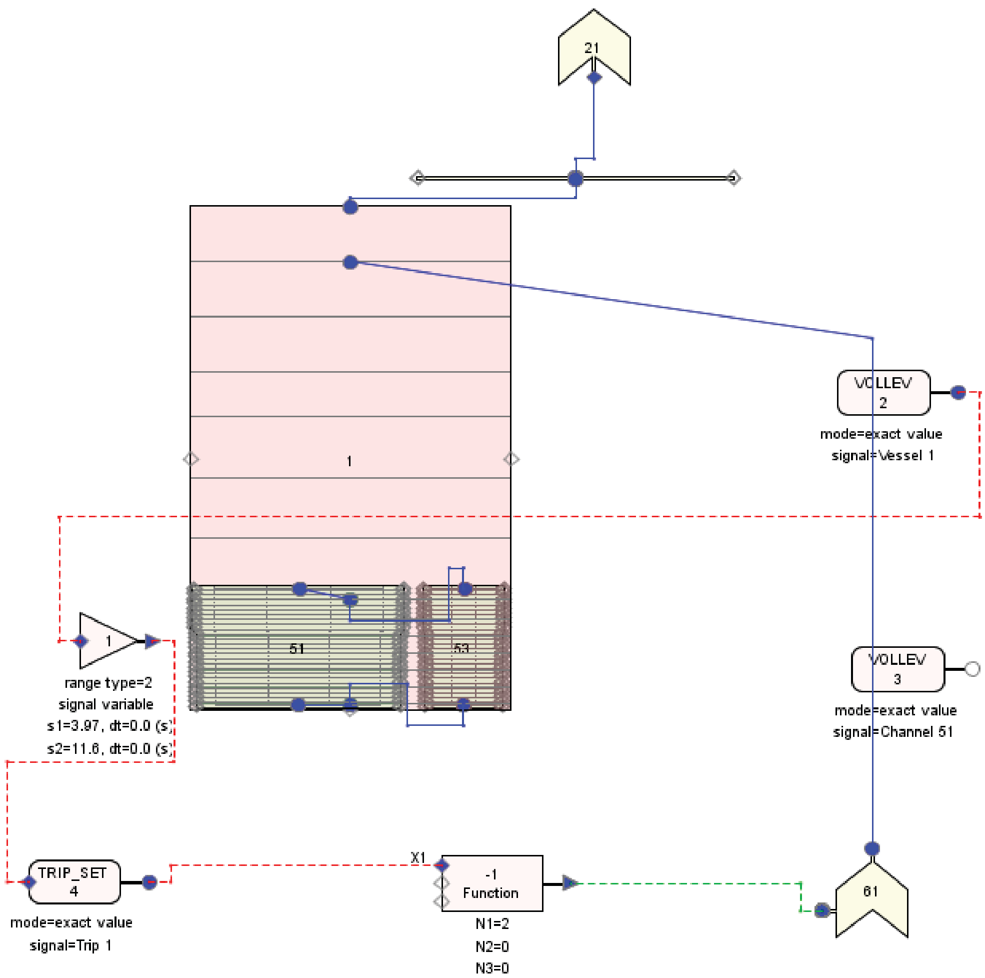

Fig. 1 TRACE/SNAP model

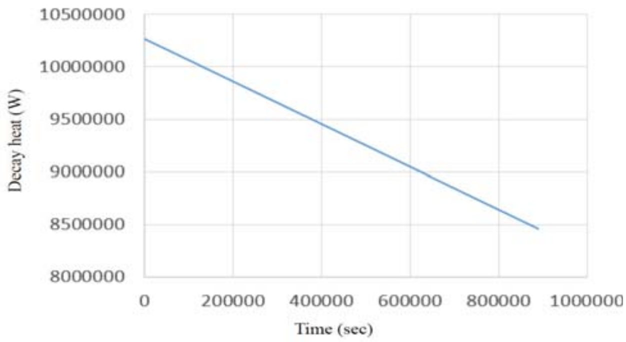

Fig. 2 Decay heat

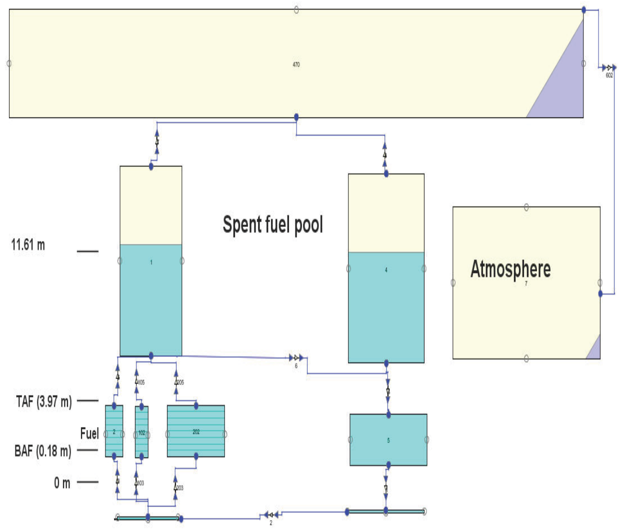

Fig. 3 MELCOR/SNAP model

Fig. 4 shows the water level results of TRACE and MELCOR. TRACE and MELCOR predictions were similar. According to Fig. 4, the water level was lower than TAF (top of active fuel) at 4th day roughly. The uncovered of the fuels caused to the cladding temperature increase. Fig. 5 presents the max cladding temperature results of TRACE and MELCOR. Their results were also similar. The initial water temperature of the pool was 311 K. After the cooling system of SFP failed, the time of the cladding temperature which reached 373 K (100 °C) was roughly 0.4 day. Subsequently, because no water was added into SFP, the water dried out which led to the water level be lower than the TAF. Finally, the cladding temperature reached 1088.7 K at 5th day roughly. According to the URG [9], the max cladding temperature should be lower than 1088.7 K. When the max cladding temperature reached 1088.7 K, it indicates that the zirconium-water reaction of the fuels occurs. The zirconium-water reaction may make the cladding temperature increase sharply and may generate the burst of the cladding of the fuel rods. The above phenomenon may cause the safety issue of the fuels.

Fig. 6 shows MELCOR results about zirconium-water reaction. When the zirconium-water reaction occurred, the mass of Zr decreased and ZrO_2 increased. In addition, the mass of H_2 also went up after five days. Because the zirconium-water reaction occurred, the larger heat amount was observed after five days.

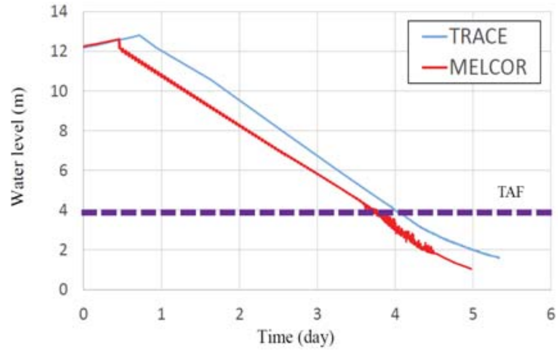

Fig. 4 Water level results

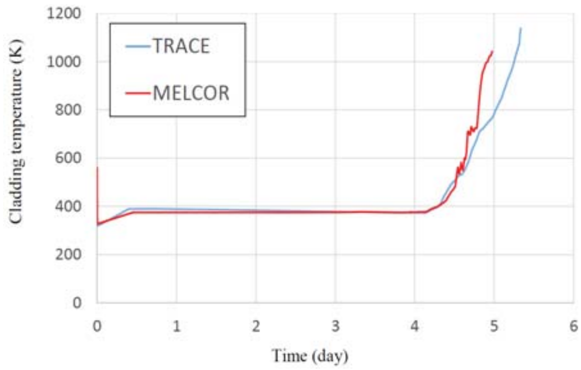

Fig. 5 Cladding temperature results

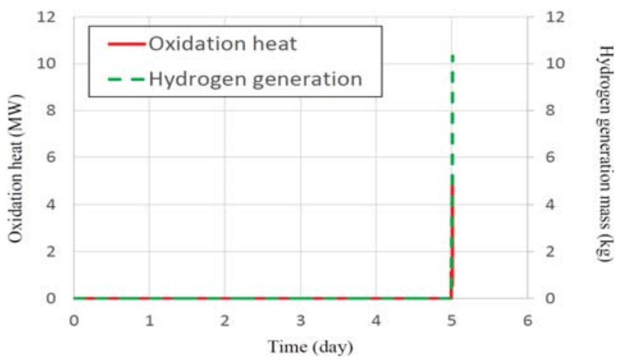

Fig. 6 Oxidation heat and hydrogen generation results

Additionally, in order to estimate the safety of SFP in more detail, we will modify the methodology of Kuosheng NPP SFP in the future. The new methodology is presented in Fig. 7. The CFD and FRAPTRAN analysis will be added in this methodology. The CFD model will be developed in order to estimate the thermal-hydraulic phenomenon of the local region of SFP in detail. By using TRACE, or MELCOR, or CFD results (ex: power and coolant conditions), the fuel rod of FRAPTRAN model will be established. And the fuel rod performance analysis of SFP will be performed by the FRAPTRAN model.

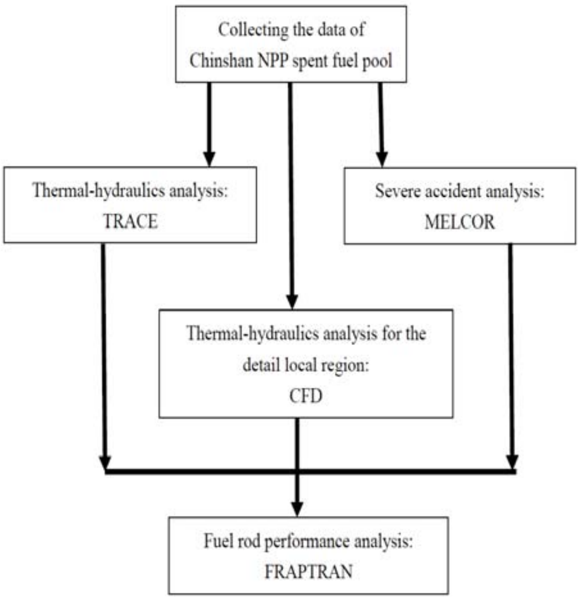

Fig. 7 New methodology for Kuosheng NPP SFP

IV. Conclusion

This study has developed MELCOR and TRACE models of Kuosheng NPP SFP with SNAP interface code successfully. By using the above models, the safety analysis of SFP was performed under the cooling system of SFP failure condition (Fukushima-like condition). The predictions of MELCOR and TRACE were similar in this case. Their results show that the uncovered of the fuels occurred at 4^{th} day roughly. The cladding temperature reached 1088.7 K at 5^{th} day roughly. The above results indicated that Kuosheng NPP SFP may be in an unsafe situation. In addition, these results will be the references for the CFD and FRAPTRAN analysis in the future.

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