METHODOLOGY USING MELCOR2.1/SNAP TO ESTABLISH AN SBO MODEL OF CHINSHAN BWR/4 NUCLEAR POWER PLANT

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ABSTRACT

After Fukushima Daiichi accident, the safety analysis of severe accident became one of the safety concerns in Taiwan. In recent days, both MELCOR and MAAP were the main used codes for nuclear reactor severe accident in Taiwan. These two codes can both calculate the phenomenon happen in the late reactor severe accident, like core meltdown, hydrogen generation and fuel debris penetrations, etc.

The main MELCOR used in Taiwan now was an earlier version and in the ASCII code mode (Coding with text file) which is not always easily to be understood by a beginner of the code. In this research, the latest version MELCOR2.1 was used and combined with Symbolic Nuclear Analysis Package (SNAP). In this combination, MELCOR was used with a graphical user interface (GUI) that users can easily modify any detail of the model. It can also combine some other applications like AptPlot for output drawing and DAKOTA for uncertainty analysis. There were three main steps in this research. First one is to establish the MELCOR2.1/SNAP model of Chinshan (BWR/4) nuclear power plant (NPP). This model included 22 control volumes of lower plenum, core, separator, dryer, drywell, wetwell, reactor building and environment, etc. The components of Fuel Dispersal, Cavities and Radionuclide were also built in. Second, a steady-state test was calculated by MELCOR to check the model. After the steady state calculations, the model of MELCOR2.1/SNAP will next used for a transient situation. Finally, the Chinshan NPP model was set to a situation of SBO to simplify the control system. After the model was built, the SBO calculation of this model was done and compared to the results of MAAP5.0 and MELCOR1.8.5. The results of this MELCOR2.1/SNAP model fit the MAAP results well at the main outputs we concerns in the severe accident. In addition, a Chinshan NPP spent fuel pool SBO model was also built by MELCOR2.1/SNAP and the output of cladding temperatures, water level also fit the results from MAAP calculations. The comparisons of MELCOR spent fuel pool and some other thermal-hydraulic codes like TRACE and CFD were also done in this research.

1. INTRODUCTION

On March 11th, 2011, a huge earthquake led to a situation of Station Blackout (SBO) at the Fukushima Daiichi nuclear power plant (NPP). The accident caused core meltdown, hydrogen explosions and the release of radionuclides.

The Fukushima Daiichi Accident let people notice that the severe accident analysis is a very important issue. In recent days, both MELCOR and MAAP are the main used codes for nuclear reactor severe accident in Taiwan. These two codes can both calculate the phenomena happened in the late reactor severe accident, such as core meltdown, hydrogen generation and fuel debris penetrations.

The main version of MELCOR used in Taiwan was an earlier one. The ASCII code mode (Coding with text file), which was used by MELCOR, is not always easily to be understood by a beginner. Therefore, the latest version MELCOR2.1 was used in this research and combined with Symbolic Nuclear Analysis Package (SNAP). In this combination, MELCOR was used with a graphical user interface (GUI) that users can easily modify any detail of the model.

In this study, a MELCOR2.1/SNAP model was built and used to calculate a severe accident case (SBO with high pressure failure) of Chinshan NPP in Taiwan. Base on the analysis of this study, people can understand the phenomena happened in the late severe accident and do the advance preparation to it. The detail information of MELCOR and MAAP code will be shown in the following paragraphs.

1.1. MELCOR2.1/SNAP

MELCOR is a fully integrated, engineering-level computer code that models the progression of severe accidents in light-water reactor nuclear power plants. MELCOR is being developed at Sandia National Laboratories for the U.S. Nuclear Regulatory Commission as a second-generation plant risk assessment tool and the successor to the Source Term Code package.

A broad spectrum of severe accident phenomena in both boiling and pressurized water reactors is treated in MELCOR in a unified framework. These include thermal-hydraulic response 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.

MELCOR applications include estimation of severe accident source terms, and their sensitivities and uncertainties in a variety of applications. MELCOR is also used to analyze design basis accidents for advanced plant applications (ESBWR, EPR, APWR) [1].

In this study, MELCOR2.1 combined to a program called Symbolic Nuclear Analysis Package (SNAP). The model modification of SNAP interface will be easier than the old ASCII file. Figure 1 shows a part of the MELCOR 2.1/SNAP user interface of input file.
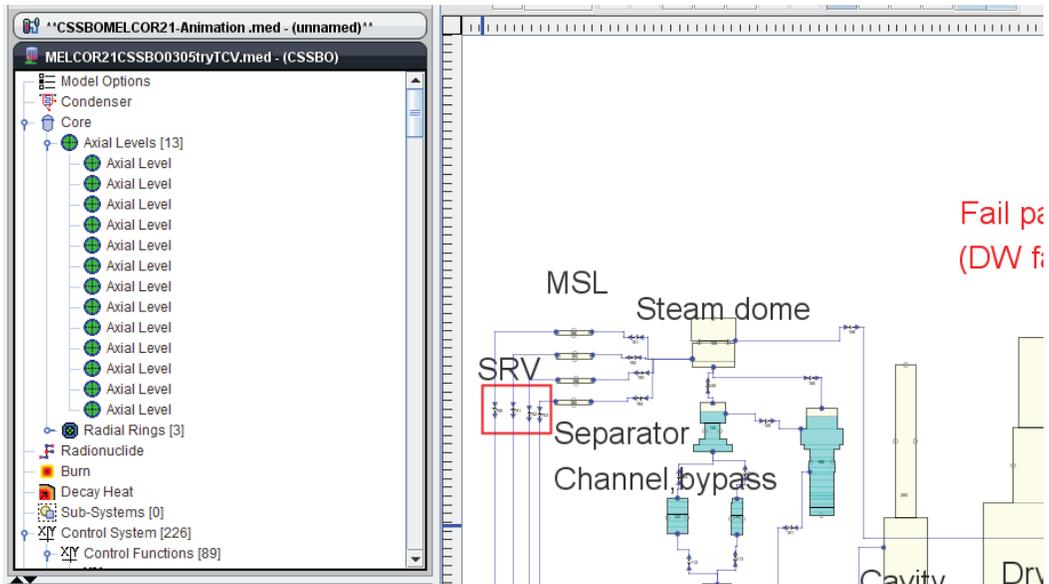
Figure 1. Interface of MELCOR2.1/SNAP

1.2. MAAP5.0

MAAP (Modular Accident Analysis Program) is a code can calculate the phenomenon of core, water cooling systems and containment which happen in a severe accident. The first version MAAP1 was developed by FAI (Fauske & Associates, Inc.).

After that, several updates were done and improved the calculation in the MAAP code. MAAP4 improved some of the physics and chemical calculation models. MAAP4 also added the ability to simulate an ALWR (Advance Light Water Reactor). The version used in this research was MAAP5.0 and the calculations were done by INER (Institute of Nuclear Energy Research), Taiwan [2].

2. DISCIPTION OF CHINSHAN NPP MELCOR2.1/SNAP MODEL

The code versions used in this research were SNAP v2.2.9 and MELCOR2.1. The process of Chinshan NPP MELCOR2.1/SNAP model development is as follows (shown in Figure 2): First, the system and operating data for the Chinshan NPP was collected [2][3][4]. Second, several important control systems such as SRV flow control system, pressure control system and feed water flow control system were established by MELCOR2.1/SNAP. Next, other components such as cavity, main steam line, reactor building rooms, etc. were added into the MELCOR/SNAP model to complete the model for Chinshan NPP. Finally, a steady-state analysis was done to check this model. After all these steps, the MELCOR2.1 /SNAP model of Chinshan NPP was used to calculate a case of SBO with high pressure failure (A SBO case without pressure released and water injection). The results were compared to MAAP5.0 and MELCOR1.8.5. The full view of the Chinshan SBO MELCOR2.1/SNAP model was presented in Figure 3. SNAP also can use the MELCOR results data to make an animation for transient, such as Figure 4. The following sections will describe the details of the methodology of building up the MELCOR2.1/SNAP model in this study.
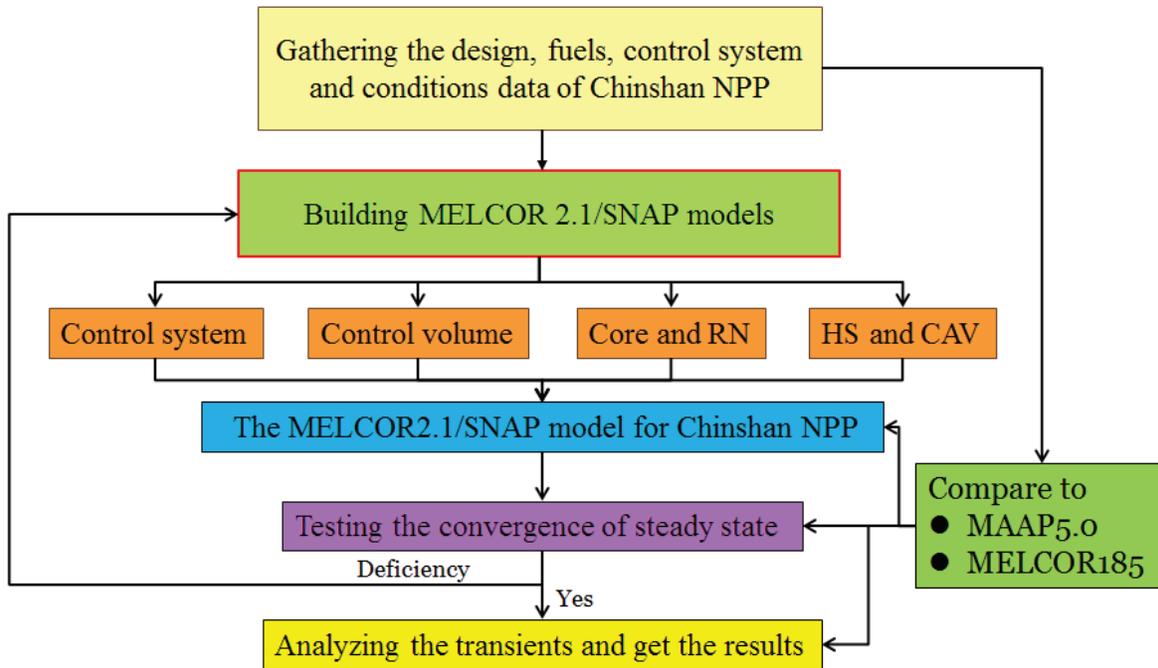
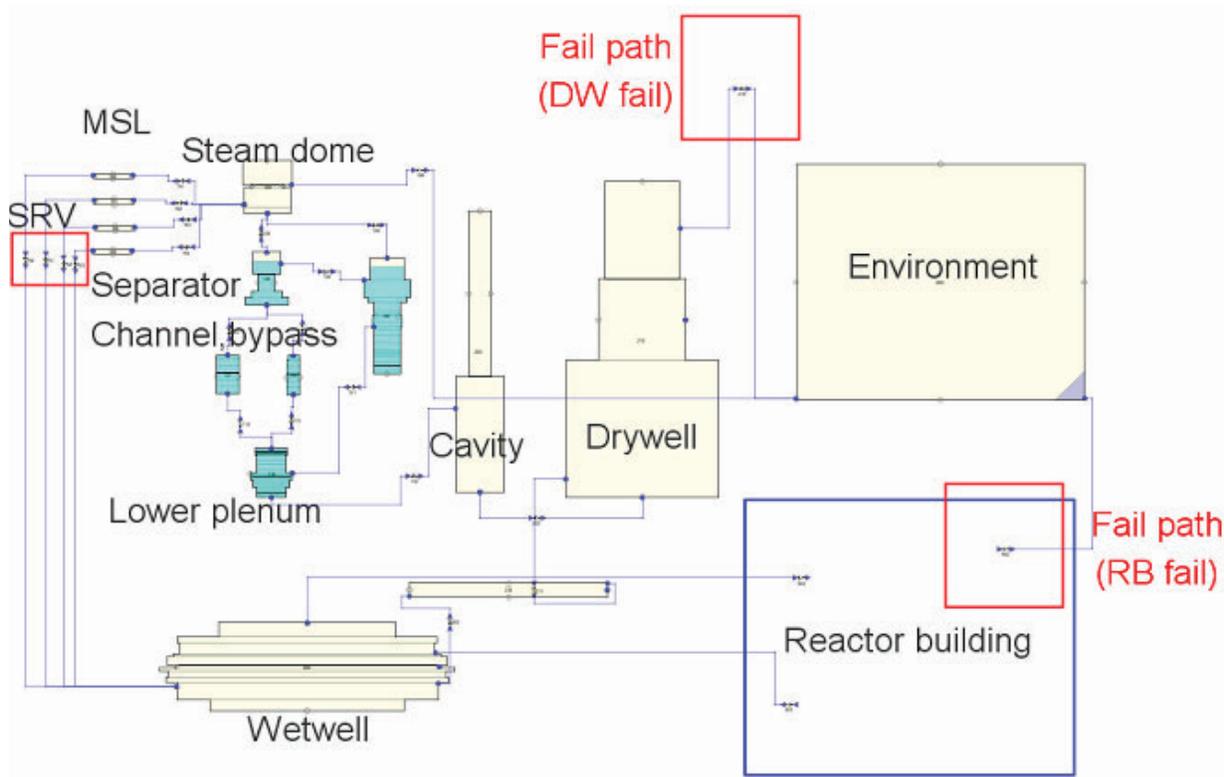**Figure 2. Flow chart to establish Chinshan NPP MELCOR2.1/SNAP model**

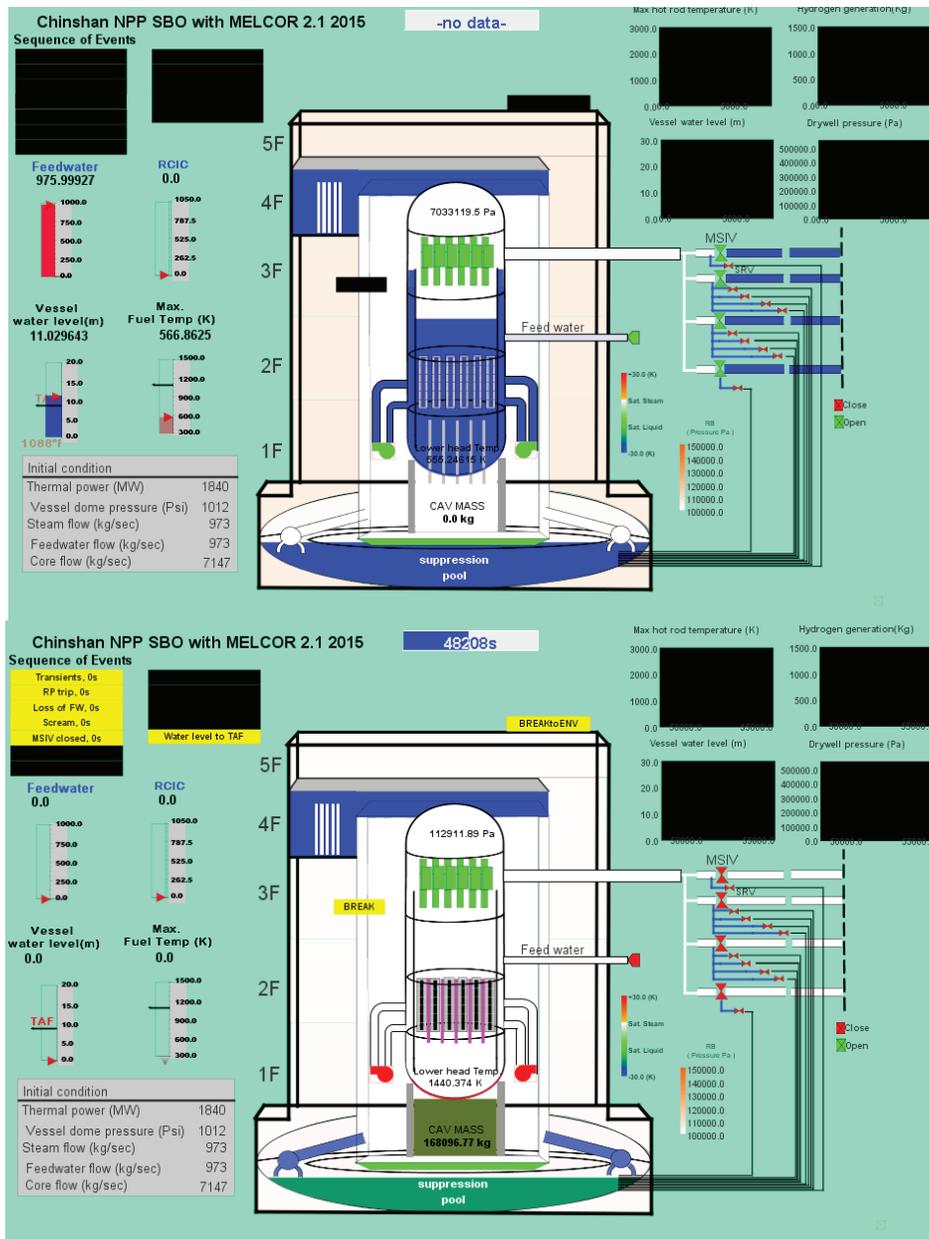**Figure 3. Chinshan NPP MELCOR2.1/SNAP model**

Figure 4. Animation model

2.1. Basic descriptions of Chinshan NPP MELCOR2.1/SNAP model

This model included one core component, 22 control volume components, 73 heat structure components, 110 control/tabular functions, 32 flow path components. The thermal power of this Chinshan NPP model was 1840MW. The Radionuclide, Fuel Dispersal, Burn, Decay Heat and Cavity packages were also included in this model.

Figure 5 shows the basic working flow chart of MELCOR, MELCOR separates the calculation to three parts: Thermal-hydraulic, core mass and RN mass. COR, FDI and CAV components were used to calculate some of the mass in the core, such as fuel relocation, debris mass. RNCOR and RNFDI were used to calculate the radiation nuclide mass and transfer the mass calculations to the control volume.

Finally, the thermal-hydraulic calculation was done by the control volume package. The MELCOR2.1/SNAP model in this study can roughly separate to three parts: reactor pressure vessel (RPV), main steam line and reactor building/containment. The following sections will show the detail of each part.
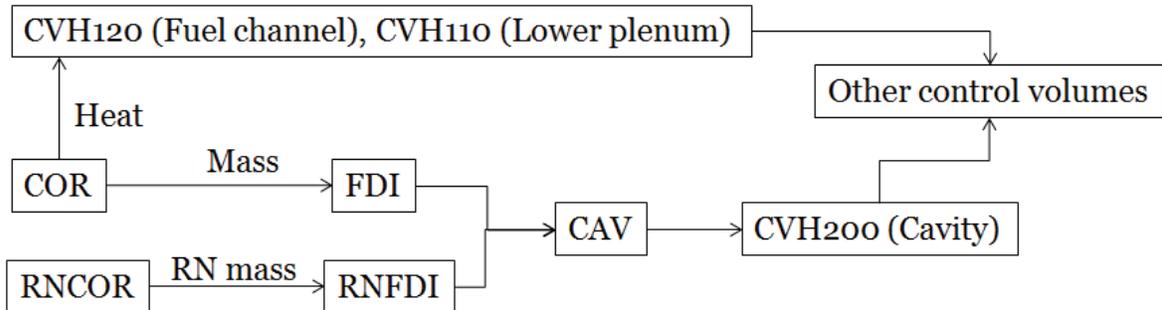

Figure 5. MELCOR calculation method

2.2. Reactor pressure vessel (RPV)

Figure 6 shows the pressure vessel control volumes and COR component of this Chinshan NPP MELCOR model. The lower three control volume: Channel, Bypass and Lower plenum were the control volumes which connected to the COR package. Level 1 to 7 of COR package were lower plenum and level 8 to 13 were reactor core, this connection allowed the control volumes to calculate the thermal-hydraulic phenomena inside the reactor pressure vessel.

The feed water was set inside control volume 100 (RPV ANNULUS), it was an extra data source controlled by a control function which had a feedback calculation of water level. The pressure was also controlled by a similar function. These settings let the water level and RPV pressure became a number we set in the steady-state. The steady-state calculation of feed water was 977 kg/s and the pressure was 7.033E6 Pa.
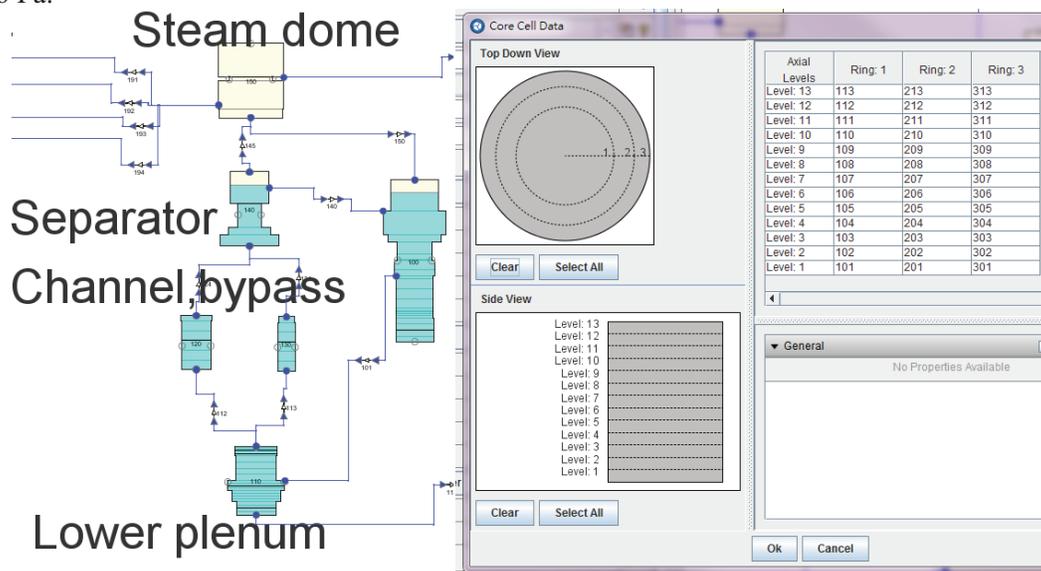
Figure 6. Reactor pressure vessel of the model

2.3. Reactor building and containment

Figure 7 shows the Reactor building and containment of this Chinshan NPP MELCOR model. The reactor building was separated to five floors and a torus room. Control volume 470 (5F) was connected to the environment with a flow path controlled by a pressure difference. When the pressure difference between 5F and environment reach 1.5 atm, the flow path will open and it means the failure of reactor building. Also, there was a flow path between drywell and reactor building 3F, the pressure setting of failure was 5 atm. The failure settings were as same as the settings of MAAP5.0.

In the Chinshan NPP design, the containment was filled with nitrogen to avoid the hydrogen exploration, so all the control volumes except environment were set to be filled with nitrogen in this model.

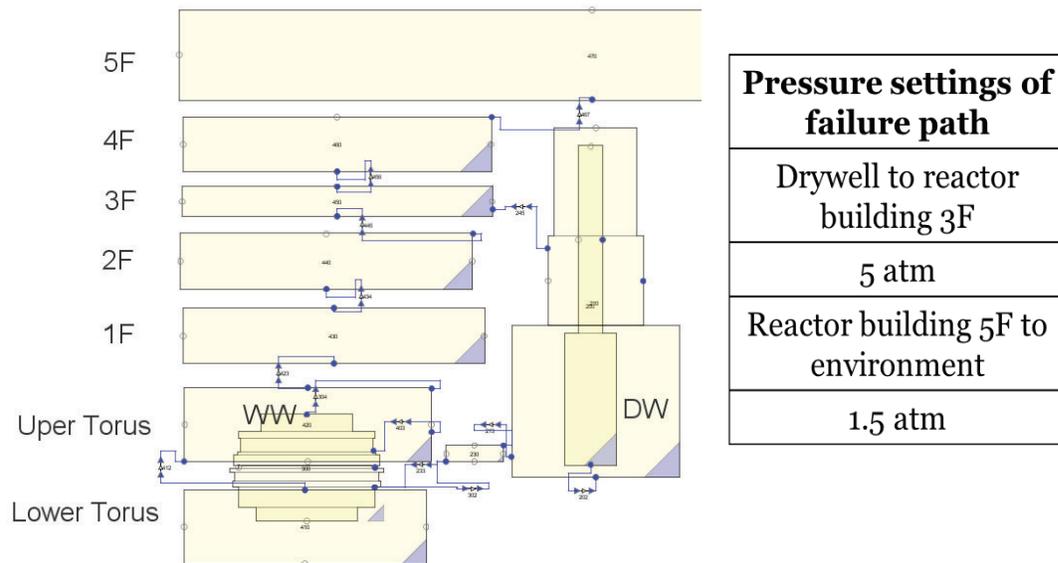| Pressure settings of failure path |
| :---: |
| Drywell to reactor building 3F |
| 5 atm |
| Reactor building 5F to environment |
| 1.5 atm |

Figure 7. Reactor building and containment of the model

2.4. Main steam line and SRVs

Figure 8 shows the main steam line and SRVs. Main steam lines were simulated by four control volumes and connect to wetwell by SRV flow path. The SRV pressure settings are also shown in figure 8. The SRV was separated to four groups. The control of SRV open fraction was the "Hysteresis Control Function" shown in figure 9. By setting a loading function and an unloading function to control how many SRVs should be opened at the setting pressure.

In this study, there was no manual depressurized and all the steam went out by the automatic open of SRVs. In this condition, the RPV pressure will maintain at a higher pressure till the PRV fail.
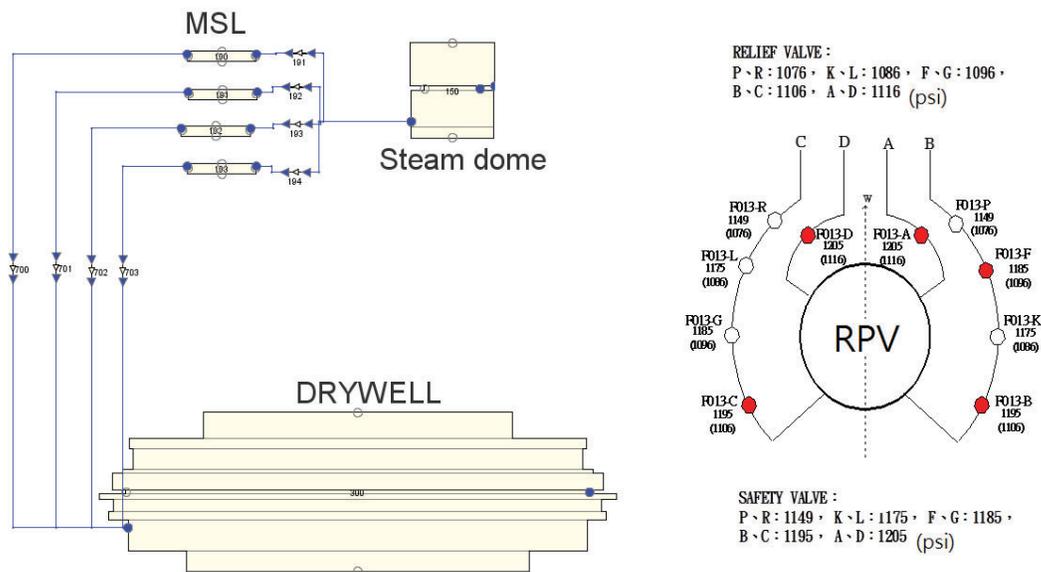
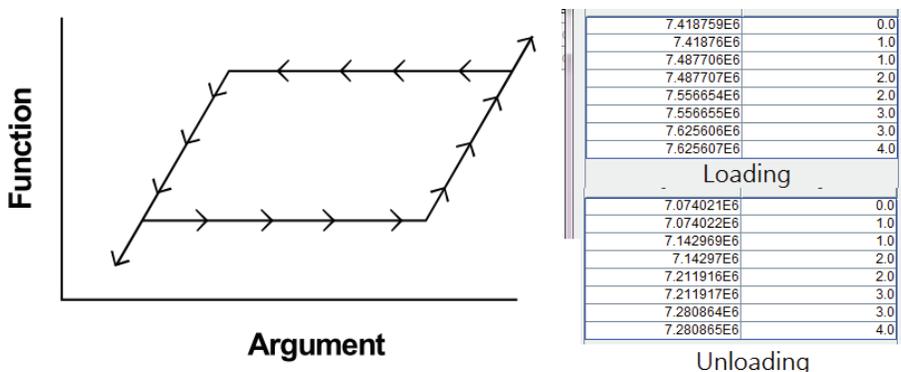**Figure 8. Main steam line and SRVs of the model**

Figure 9. Hysteresis Control Function setting of group B SRV of this model

2.5. MELCOR2.1/SNAP model of Chinshan NPP spent fuel pool

After Fukushima NPP event occurred, the spent fuel pool accident analysis became one of the major jobs in the severe accident researches. So, in order to concern the safety of the Chinshan NPP spent fuel pool, the safety analysis of spent fuel pool was performed by using MELCOR 2.1/SNAP. This model was a stand-alone model just like the spent fuel pool model of MAAP5.0, but it can still show some important behavior of spent fuel pool in the severe accident.

Figure 10 shows the MELCOR2.1/SNAP model of Chinshan NPP spent fuel pool. In this model, the spent fuels were separated to three rings. Ring 1 was the hottest group and the power distribution used in this model was also shown in figure 10.

In this study, the spent fuel pool model was set to be a SBO situation without any water injection. The results of spent fuel pool analysis were also compared to the MAAP5 results in the following paragraphs. In addition, the CFD and TRACE results were also shown in this study [5].
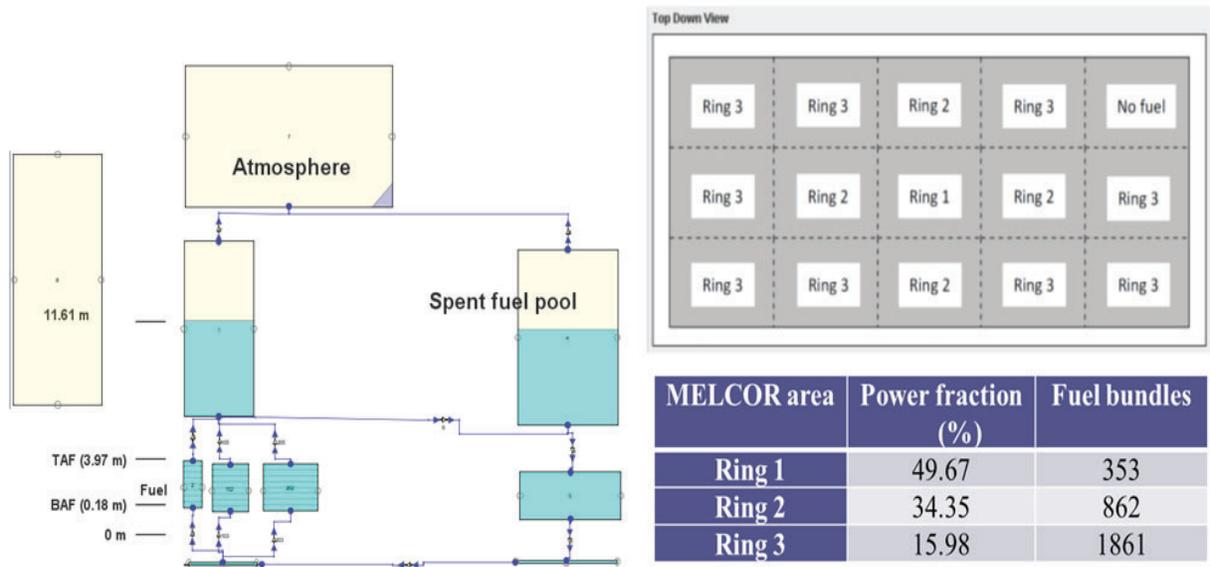
Figure 10. Chinshan NPP spent fuel pool model of MELCOR2.1/SNAP

MELCOR area	Power fraction (%)	Fuel bundles
Ring 1	49.67	353
Ring 2	34.35	862
Ring 3	15.98	1861

3.   RESULTS

The MELCOR2.1/SNAP model of Chinshan NPP was described above. Next step is the results of this model and it was separated to three parts. First, a steady-state calculation was done to check the stability of this model. Second, the transient analysis of Chinshan NPP SBO was calculated by MELCOR. Third, the spent fuel pool results will be shown and compared to other codes.

3.1.   Initial condition and assumptions/Steady state test

The model was set to start calculation at -300 sec and start SBO event at 0 sec. So the time between -300 sec and 0 sec was the steady-state test region. The results of steady-state test of this Chinshan NPP model are shown in figure 11-12.

Figure 11 shows the feed water flow calculated by the control function to control the water level and pressure. The water level was set to be 14.08m at steady-state and the RPV pressure was 7.033E6 Pa. By these settings, the feed water flow became steady as 977 kg/s and the value of this steady state feed water flow was as same as the result calculated by TRACE code in the Chinshan NPP model [4]. The water mass inside RPV was 140000 kg. The detail of the initial conditions and assumptions are shown in table I.
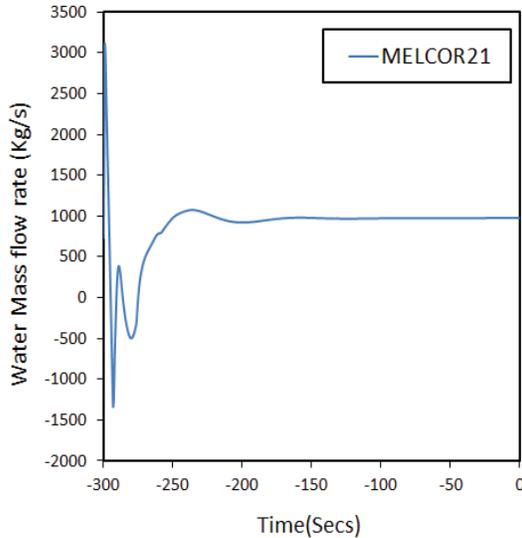
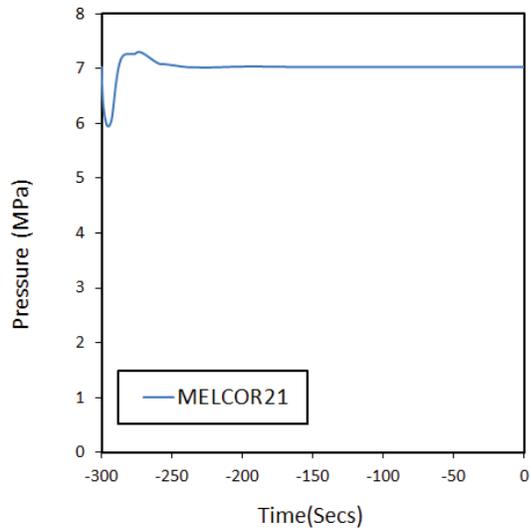**Figure 11. Feed water flow rate**

Figure 12. RPV pressure

Table I. Initial conditions and assumptions

Feed water flow	977 kg/s	Thermal power	1840 MWt
Steam flow	977 kg/s	Water level	14.08 m
RPV pressure	7.033 MPa	Cladding temperature (Max)	566.86 K
Core flow	6678 kg/s	Core water temperature	560 K

3.2. Transient analysis (SBO with high pressure failure)

The SBO event started at 0 sec. After the event of SBO, NPP lost all the water injection and water level went down because of water evaporation. Figure 13 shows the water level results of MELCOR and MAAP. The time to TAF (Top of Active Fuel) calculated by MELCOR2.1 was 42 minutes and the MAAP5 calculation was 48 minutes. The water level keeps going down and the lower plenum dry out at 5.6 hrs. and 4.06 hrs. calculated by each code. The time difference of lower plenum dry out was because of the different definition of two codes. There was still 20000kg water in the RPV when MAAP5.0 showed the message of lower plenum dry out.

The pressure rose due to the decay heat and the steam went out to the wetwell through SRVs. The RPV pressure is shown in figure 14. The drop of RPV pressure was because of the failure of RPV. MAAP calculation shows that the RPV failed at 4.67 hrs. and it was earlier than the MELCOR2.1 calculation 6.83 hrs. This difference maybe cause by the model of penetration and fuel relocation were not the same. The steam kept going to the wetwell and the pressure of wetwell rose and finally broke at 6.8 hrs. The trend of wetwell pressure was very close of MELCOR2.1 and MAAP5.0 (shown in figure 16).

After the water level was lower than TAF, fuel temperature started going up and reached 1500°F (1088K) at 1.17 hrs. and 1.23 hrs. for MELCOR2.1 and MAAP5. Figure 15 shows the max hot rod temperature of each code. The temperature of fuel went to zero means the rod relocation happened and dropped into lower plenum. The core relocation of MELCOR2.1 was separated to three rings so the first ring will drop to lower plenum little earlier than MAAP5, which the relocation was calculated as a whole core.

Figure 17 shows the mass of core, lower plenum and cavity of MELCOR2.1. The red line shows the core mass and the first fuel relocation happened at 2.35 hrs. The fuel all drop to lower plenum at 5.5 hrs. and started to penetrate the the bottom of RPV. MELCOR2.1 calculated that the RPV failed at 6.83 hrs. and MAAP5.0 was 4.67 hrs. The large difference of these calculations is because of the difference of fuel relocation model and penetration model between two codes. The results of RPV failure time were very close between MELCOR2.1 and MELCOR185. The debris of MELCOR dropped separately and easier to be cooled by water in the lower plenum while the debris of MAAP5.0 just dropped into lower plenum together and evaporated large amount of water immediately. It could be the main reason caused the difference of RPV failure time.

Figure 18 shows the hydrogen generation of each code. MELCOR calculated that in this case there was about 800 kg hydrogen generation and MAAP5 only calculated as 100 kg. In the previous work of the comparison of MELCOR and MAAP, MAAP always lower predicted the hydrogen generation. So the MAAP5.0 code settings caused the large different of hydrogen generation.

Table II shows the important event time line in this research. It shows a large difference after the fuel relocation happened. The reason was already told previously. According to the MELCOR calculation, after RPV failure, the high pressure almost immediately causes the failure of drywell and reactor building 5F. But the MAAP5 calculation had a delay between RPV failure and reactor building failure.
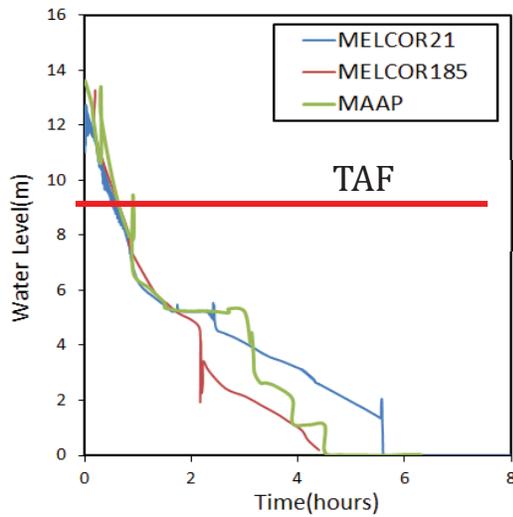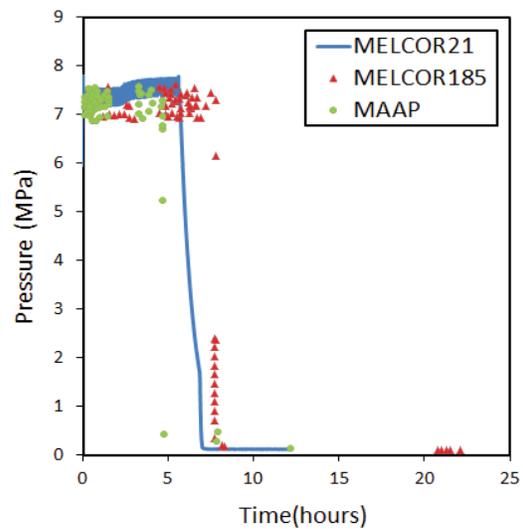

Figure 13. Core water level
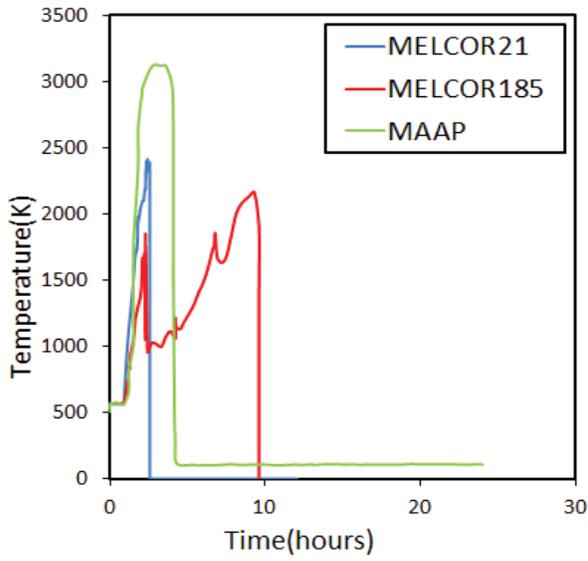

Figure 14. RPV pressure
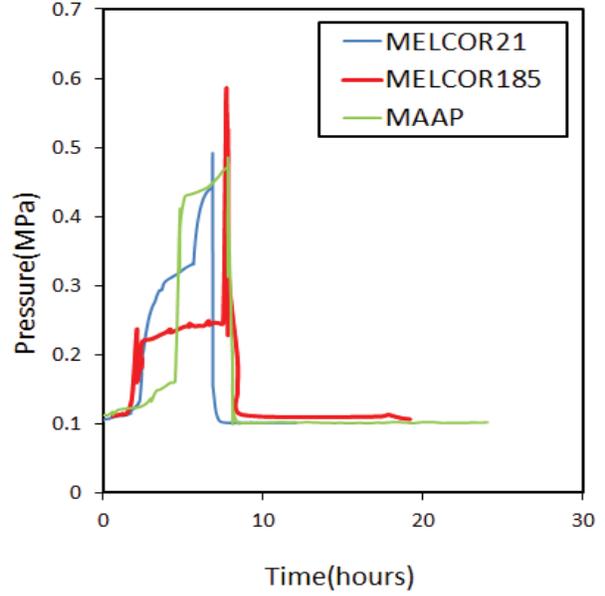
Figure 15. Max hot rod temperature
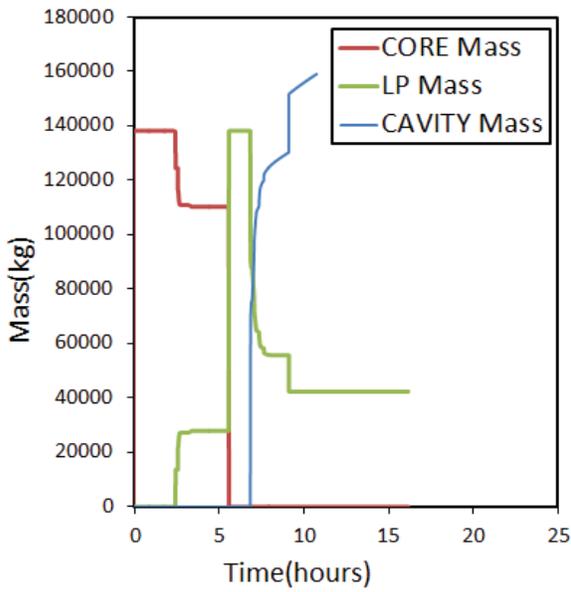

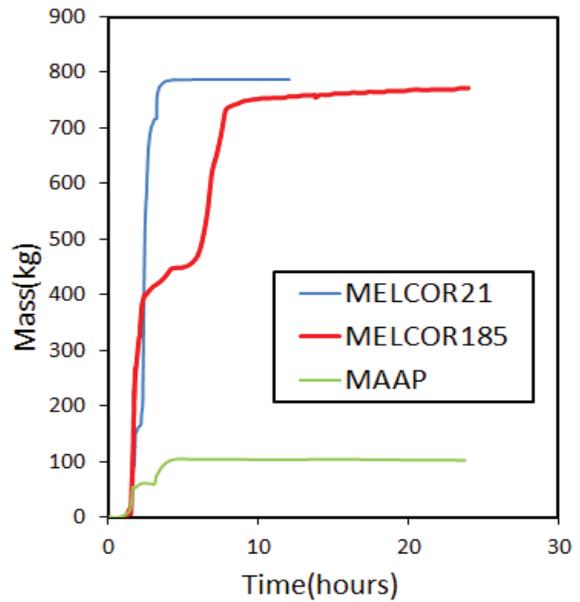Figure 16. Wetwell pressure

Figure 17. Mass in different locations

Figure 18. Hydrogen generation

Table II. The time line comparison of the important events

| Time (hrs.)	MELCOR21	MELCOR185	MAAP5
SBO	0	0	0
TAF	0.7	0.76	0.8
Fuel > 1088K	1.17	1.33	1.23
Fuel > 1477K (2200F)	1.48	1.49	1.4
BAF	1.62	1.9	2.53
Fuel relocation	2.35	2.22	3.21
LP dry out*	5.6	4.74	4.06
RPV failure	6.83	7.71	4.67
RB failure	6.86	7.71	7.9

*The definition of lower plenum dry out was different between MAAP and MELCOR

### 3.3. Chinshan NPP spent fuel pool SBO analysis

In this study, all the cooling systems of the spent fuel pool were set to be failed, so no water added into the spent fuel pool during the transient. After the cooling system of the spent fuel pool failed (0 sec), this transient began. Figure 19 and figure 20 show the water level and the cladding temperature of CFD, TRACE and MELCOR2.1 [5]. This calculation shows the thermal-hydraulic calculation of MELCOR in the earlier accident time line also feet the other thermal-hydraulic codes. Figure 21 and figure 22 shows the comparisons of MELCOR2.1 and MAAP5.0 in this spent fuel pool calculations. The spent fuel pool model was very simple compare to the NPP model, so the thermal-hydraulic results of MELCOR2.1 and MAAP were closer than the NPP calculation. The cladding temperature shows the two codes calculate the fuel relocation happened at the same time. Figure 22 shows the hydrogen calculation of two codes and the large difference of hydrogen were also shown in the NPP calculation in the previous sections.
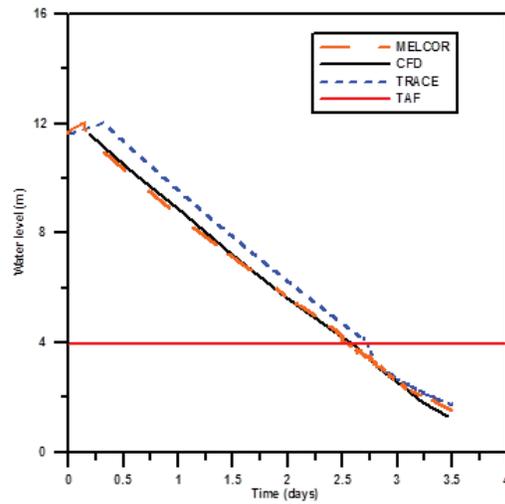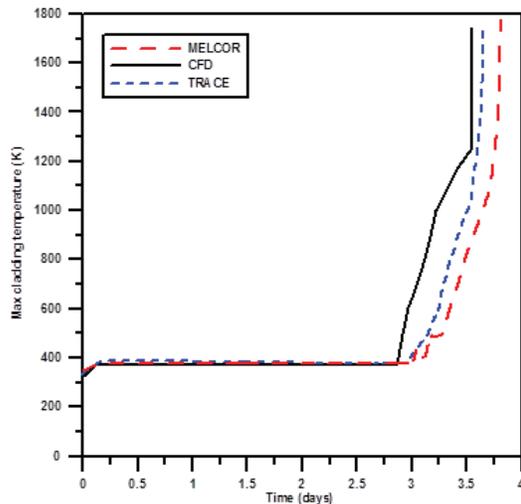
Figure 19. Peak cladding temperature of spent fuel pool   **Figure 20. Water level of spent fuel pool**
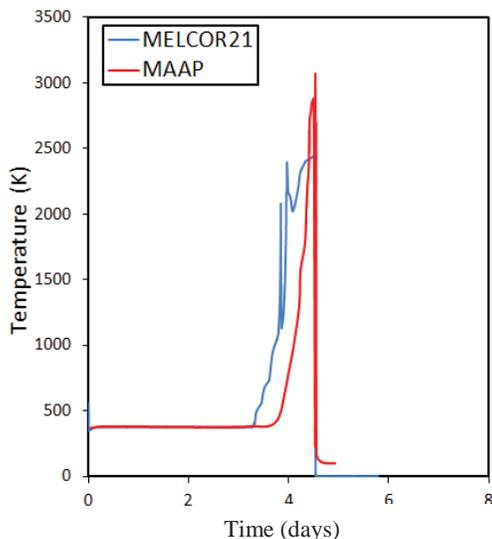
Figure 21. Peak cladding temperature of spent fuel pool
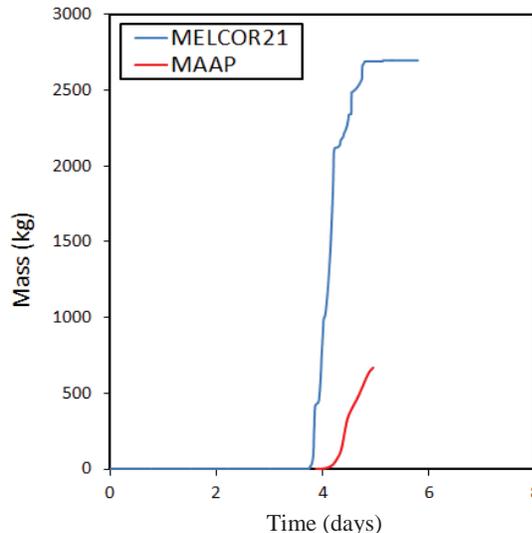
Figure 22. Hydrogen generation

4.   CONCLUSIONS

By the calculation of MELCOR2.1/SNAP, this study gives several conclusions:

1.   This study has developed the MELCOR2.1/SNAP models of the Chinshan NPP successfully.
2.   By using the above models, the safety analysis of the Chinshan NPP SBO severe accident was performed under the condition that all water injection system failed. The analysis results of MELCOR2.1, MELCOR1.8.5, MAAP5.0, are similar in this case. It indicates that the MELCOR2.1 results are consistent with other codes/versions.
3.   The analysis results depicted that the uncovered of the fuels occurred in 42 minutes and the fuel debris drop to lower plenum at 2.35 hrs.
4.   According to this study, the case of SBO caused the RPV fail at 6.83 hrs. and the reactor building fail at 6.86 hrs.
5.   MELCOR2.1/SNAP can predict the water level and cladding temperature well in a spent fuel pool SBO case.
6.   This study's results can help to evaluate the safety issue of the Chinshan NPP severe accident.

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