Verification of the MELCOR Code
Against SCDAP/RELAP5 for Severe Accident Analysis

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The MELCOR code has been developed by the US NRC to analyze hypothetical severe accidents in nuclear power plants and is a lead code for US NRC severe accident calculations. Some advantages that MELCOR has over other codes such as SCDAP/RELAP5 are the capability to evaluate containment behavior and the source term to the environment, and the great modeling flexibility that the control volume approach and control functions afford.

The latest MELCOR input deck for the 4-loop Zion PWR with a 5-ring, 12-level core model was modified to match the geometry and TMLB’ (Station Blackout Event) conditions of the NRC’s SCDAP/RELAP5 input deck. Detailed plots showed that the calculation results of the two codes are very similar in terms of thermal-hydraulic and core degradation response. There are minor discrepancies in various timings of phenomena which are within the uncertainties of the numerics and the physics models. The close agreement of the results is good evidence of MELCOR’s capabilities as a severe accident code and that the MELCOR code is suitable as the US NRC’s “workhorse” of severe accident calculations. The modeling capabilities necessary to meet the severe accident analysis goals and the flexibility to be improved where needed have been demonstrated.

KEYWORDS: MELCOR, SCDAP/RELAP5, TMLB’

I. Introduction
The MELCOR code was originally developed as a probable risk assessment (PRA) code by Sandia National Laboratory. Some advantages that MELCOR has over other codes such as SCDAP/RELAP5 are the capability to evaluate containment behavior and the source term to the environment, and the great modeling flexibility that the control volume approach and control functions afford.

MELCOR input decks often employ coarser nodalizations than SCDAP/RELAP5 input decks and some of the MELCOR modeling is less mechanistic. Despite these simplifications, MELCOR has shown favorable performance in severe accident calculations. Since SCDAP/RELAP5 has been extensively assessed against severe core damage test data, comparing MELCOR with this counterpart code provides an assessment of MELCOR’s qualifications to become a lead severe accident code.

A TMLB’ station blackout event for the Zion nuclear power plant is the reference case for this work. This event provides an appropriate basis for the codes’ assessment due to the fact that the event progression tests a significant portion of the models found in each code. The TMLB’ was also chosen to study the risk of steam generator tube rupture. Conservative analysis conditions were chosen to investigate the integrity of the SG tubes and other components. Specifically, the reactor coolant system was not allowed to depressurize following any pressure boundary failure, pump seal leakage was not considered and no operator or recovery actions were permitted.

The code versions used in this work were MELCOR 1.8.5 and SCDAP/RELAP5 MOD 3.3. For a more level comparison basis, SCDAP/RELAP5 was compiled on a PC with the same compiler and compiler options as for MELCOR and all calculations were run on the same PC. The SCDAP/RELAP5 code is typically distributed only in UNIX versions. The PC version of the SCDAP/RELAP5 executable and guidelines for installation on a PC were developed at Purdue University; in collaboration with D. Barber of ISL.

The scope of the code comparison includes the input geometries, the calculation results and the physics models of each code. This work is being performed at the Laboratory for Nuclear Heat Transfer Systems (NHTS) at Purdue University for the U.S. Nuclear Regulatory Commission.

II. Description of Zion TMLB’ Event
Originally built in the early 1970s, Zion nuclear power plant began operation in 1973. It is a Westinghouse design 4-loop pressurized water reactor (PWR) constructed by Commonwealth Edison. Units I and II have a net capacity of 1040 MWe each for a total of 2080 MWe for the plant. Both units at Zion have been closed since 1998 due to a history of operational problems. It will be decommissioned in 2013 when its license expires. Zion was chosen for this project as it is a representative plant for 4-loop PWRs.

A TMLB’ event is a station blackout where there is a loss of A/C power and steam generator auxiliary feedwater supply. This complex event has many variables that change the progression of the accident. It is assumed herein that there is no operator or outside intervention and

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the pump seal failures are not considered.

The event begins with the loss of A/C power and steam generator feedwater supply without recovery. Then the turbine trips and the reactor SCRAMs. The water in the steam generator secondary side boils away, resulting in a loss of heat sink. As the primary-side pressure increases, the pressurizer PORV’s open. (The PORV’s are assumed herein to operate on battery power.) This LOCA results in a decreasing primary-side inventory.

MELCOR reactor coolant system can be seen in Fig. 1. By Sandia National Laboratory. The nodalization of the MELCOR reactor coolant system is similar to that of the MELCOR deck. Major differences are that all four loops are individually modeled in the SCDAP/RELAP5 deck and several piping components have a finer nodalization. The three intact loops are observed to behave identically, justifying the MELCOR approach of grouping them together. The SCDAP/RELAP5 core has a 5-ring, 10-level nodalization with the lower plenum and lower head being modeled with the finite element approach by the COUPLE code.

Natural circulation may begin to occur when the primary-side water level has decreased below the hot leg elevation. The anticipation is that a natural circulation path for steam between the vessel and the steam generators will develop and steam condensation in the steam generator tubes, with condensate return to the vessel, will keep the core in a sufficiently cool state.

Due to the very hot steam coming out of the vessel, there is the potential for excessive heatup of structural components such as hot leg nozzles, the surge lines, and steam generator tubes, along with the potential failure of the pressure boundaries. If the pressure boundaries are compromised, there is a possibility of releasing radioactive material to the containment or environment. Core heatup and uncovery result in fuel failure. Molten fuel may then relocate to the lower plenum and damage the lower head and/or penetrate the vessel. Upon failure of the lower head, molten material may be released to the cavity. This molten fuel (corium) will then interact with the concrete basement and any coolant water. The containment will then pressurize.

For the current work, in addition to the code comparison, a second objective is to test the integrity of the steam generator tubes. To be conservative, the steam generator relief valves on the pressurizer loop are assumed to fail upon first challenge. This results in boiloff of the secondary side and a large pressure differential across the steam generator tube walls. Depressurization following any failure is also artificially suppressed, with the knowledge that this is not a physically realistic scenario.

One of the most challenging aspects of the event for the codes to reproduce is natural circulation cooling of the system. In addition to the vessel-steam generator path, natural circulation within the vessel can develop, driven by heat loss from the vessel walls. If the water seal in the cold leg is overcome, steam can also make a full loop from the vessel to the steam generators and returning through the cold leg. Essentially, the steam would take the same path that the water coolant would take during normal operation.

III. Input Description
1. MELCOR

The input deck for a Zion Station Blackout was provided by Sandia National Laboratory. The nodalization of the MELCOR reactor coolant system can be seen in Fig. 1. The loop with the pressurizer is modeled as a single loop while the other three loops are grouped together and modeled as one. To capture the natural circulation flow paths, the hot leg is modeled in two sections, an upper half and a lower half. The halves are connected by flow paths which allow mixing between them. Such a division of the hot leg is necessary to model the flow of steam to and from the steam generators because a single control volume can not handle countercurrent flow of a single fluid. SCDAP/RELAP5 also has the same limitation and the NRC’s SCDAP/RELAP5 input deck includes a similar nodalization scheme.

The thermal hydraulic nodalization of the vessel core region is a 5-ring, 4-level control volume geometry. The core nodalization is represented as a 5-ring, 12-level model with three core control volumes per thermal hydraulic level and 10 heated levels.

MELCOR’s containment is separated into four compartments: the cavity, lower compartment, annular compartment, and the upper compartment. The cavity contains the majority of the reactor pressure vessel, roughly 8 meters, along with an area roughly 6 meters below that. The lower compartment consists of the rest of the RPV along with part of the pressurizer and steam generators. An annular compartment can be found on either side of the lower compartment. The purpose of this compartment is to allow flow from the cavity to the environment or the upper compartment. The upper compartment is a vast space that includes the rest of the reactor coolant system along with the containment sprays.

2. SCDAP/RELAP5

The nodalization of the SCDAP/RELAP5 reactor coolant system is similar to that of the MELCOR deck. Major differences are that all four loops are individually modeled in the SCDAP/RELAP5 deck and several piping components have a finer nodalization. The three intact loops are observed to behave identically, justifying the MELCOR approach of grouping them together. The SCDAP/RELAP5 core has a 5-ring, 10-level nodalization with the lower plenum and lower head being modeled with the finite element approach by the COUPLE code.

IV. Comparison of Input Models

Major differences in the input models are mainly related to the components modeled. MELCOR models the turbines, containment, environment, relief tanks, and others while SCDAP/RELAP5 is limited to the reactor coolant system. As stated earlier, the SCDAP/RELAP5 deck has four separate loop models while the MELCOR deck has only two loops, one single loop with the pressurizer and the other as the other three loops grouped into one. Another area for comparison is the lower head. SCDAP/RELAP5 provides for a very detailed nodalization of the lower plenum and lower head by the COUPLE code, while MELCOR’s modeling uses a small number of lower plenum control volumes and a 4-temperature node conduction model for the bottom head.

The MELCOR input was changed to be as consistent as possible with the SCDAP/RELAP5 deck. Specifically, several components were renodalized and various set points...
and control logic were changed to be the same. In particular, control functions were changed so that breaks were not modeled following any detection of pressure boundary failure. Creep control function checks were modified to check for multiple failures. Core melt ejection was prevented and relief valve and other set points were all made consistent. The relief valve on the steam generator secondary side on the pressurizer loop was assumed to fail open upon first challenge, to place a large pressure difference across the steam generator tubes.

Regarding component renodalization, the original MELCOR nodalization of the steam generator tubes was much coarser than the SCDAP/RELAP5 nodalization. This limited the temperature profiles that could be obtained for steam generator tube rupture evaluation. Therefore, the structures were remodeled using the nodalization as seen in Fig.2.

V. Comparison of Calculation Results

There is good agreement between the two codes from the plots of the code results. Figure 3 shows the depressurization for the secondary side on the pressurizer loop. The steam generator secondary side on the pressurizer loop depressurizes in both codes within 2000 seconds. It can be seen that the valve frequencies on the other loops are very close showing the similarities in the codes’ modeling of the boil off rates. The fact that the MELCOR intact loop hovers above the SCDAP/RELAP5 loop is not well understood. The PORV relief valves have identical settings, but the pressure seems to overshoot in MELCOR before decreasing.

The flow rate out of the secondary side in the MELCOR calculation also agrees well with that of the SCDAP/RELAP5 results, as shown in Fig. 4. This shows that the boiloff timing of the secondary side is essentially identical and the heat transfer models of the two codes are consistent with each other.

The pressurizer pressures are shown in Fig. 5 and follow the same pattern. The initial dip within the first 2000 seconds is due to high heat removal through the stuck-open valve on the steam generator secondary side on the pressurizer loop. Enough energy is being moved that the
primary side pressure is reduced. The MELCOR results indicate an initial dip that is slightly larger than that of the SCDAP/RELAP5 results.

Figure 6 reveals similar flow rates and opening times of the pressurizer PORV’s. This is a direct reflection of the heat removal through the steam generators being similar. Even the first opening for both codes occurs at a very similar timing around 1900 seconds. SCDAP/RELAP5 results show fewer valve cyclings after 15,000 seconds. This is a result of differences in the core degradation, which will be discussed below.

![Fig. 5 MELCOR and SCDAP/RELAP5 Pressurizer Pressure](image)

![Fig. 6 MELCOR and SCDAP/RELAP5 Flow through Pressurizer PORV](image)

The pressurizer in the MELCOR calculation tends to hold water longer than the SCDAP/RELAP5 pressurizer does (Fig. 7). The cause of the discrepancy is under investigation and is suspected to be related to the surge line modeling. Flow through the surge line is very complicated as it involves two phase flow with bubbly and other regimes in horizontally and inclined pipe sections. The water level in these sections is highly dependent on the water level tracking and heat transfer models in the codes.

Figure 8 shows the steam generator tube temperatures for the pressurizer agree until roughly 16,000 seconds. The digression is due to differences in the timing of core slumping. The molten fuel crust has failed the periphery and there is a massive slumping at 16,130 seconds in the SCDAP/RELAP5 results. The core reaches the lower plenum after 17,000 seconds in the MELCOR results.

The hydrogen generation rate is a good indicator for metal-water reaction rates (Fig. 9). MELCOR predicts a larger amount of hydrogen produced in comparison to SCDAP/RELAP5. Both codes show that the fuel oxidation begins to occur at roughly the same time. However, the core uncovers faster in SCDAP/RELAP5 and this is the suggested cause for the faster hydrogen generation rate at 15,000 seconds.

![Fig. 7 Collapsed Water Level in Pressurizer for Both MELCOR and SCDAP/RELAP5](image)

![Fig. 8 MELCOR and SCDAP/RELAP5 Steam Generator Tube Temperatures on Pressurizer Loop](image)

![Fig. 9 MELCOR and SCDAP/RELAP5 Hydrogen Production](image)
The runtimes for both codes are comparable (Fig. 10). It should be noted that MELCOR is running faster than SCDAP/RELAP5 after about 25,000 seconds, and that cases going beyond 30,000 sec. (8-1/3 hours) run very quickly. The codes were run on a machine with an 866 MHz processor. With further optimization of the time step sizes and compiler options, the runtimes can be reduced.

VI. Conclusions

The MELCOR input deck was modified for a Station Blackout Event (TMLB') to be as consistent as possible with the SCDAP/RELAP5 geometry and conditions. Major modifications occurred in renodalizing the steam generators and changing control logic and set points.

Detailed plots show that the thermal hydraulic phenomena are in good agreement. There are several discrepancies that could be termed as minor that are possibly due to uncertainties in the numerics and physics models. The overall closeness of the results shows MELCOR’s aptitude as a severe accident code. Future work includes comparing the core degradation progression in detail, looking into the steam generator tube rupture issues, and adding the MAAP4 code to the comparison.

It should be noted that this work was performed without the assistance from the code developers. Working in an environment such as this allows unbiased conclusions to be made from each code’s results. Also, the ability for user’s to write complex input models along with run and debug the code ensures that the code will be a useful tool.

Nomenclature

LOCA             Loss of Coolant Accident
NHTS             Laboratory for Nuclear Heat Transfer Systems
MSIV             Main Steam Isolation Valve
NRC              U. S. Nuclear Regulatory Commission
PORV             power operated relief valve
PRA              probable risk assessment
PRT              Pressurizer Rupture Tank
PWR              pressurized water reactor
RCS              reactor coolant system
SG               steam generator
SR5              SCDAP/RELAP5
TMLB’            Station Blackout Event with no recovery of SG feedwater

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References

Fig. 1 MELCOR Reactor Coolant System