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Sensitivity Analysis of Heat Transfer Coefficient in Lower Plenum for In-CV LOCA Severe Accident within i-SMR

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1. Introduction

The innovative Small Modular Reactor (i-SMR) is currently under development in Korea. In order to prove the robustness of the design and components, extensive research is being conducted to verify its safety guidelines [1, 2, 3]. To assess the validation of the design, thorough analysis on every possible scenario needs to be investigated. This study contributes to that ongoing effort by focusing on the scenario of an In-vessel Loss of Coolant Accident (InCV-LOCA). The progression of event and the sensitivity of related variables are examined throughout this paper.

Safety is a critical criterion in designing the i-SMR. To meet the safety standards, i-SMR has incorporated advanced systems, including safety valves and cooling mechanisms. The key safety components are the emergency depressurization valve (EDV) and the emergency recirculation valve (ERV). The locations of the valves are demonstrated on Fig1(a). EDV, located at the top of the pressurizer, equalizes the pressure between the reactor vessel (RV) and the containment vessel (CV) during a Loss of Coolant Accident (LOCA). ERV, positioned above the core but below the steam generator (SG), facilitates the recirculation of coolant from the CV to the Reactor Coolant System (RCS), ensuring continuous cooling in emergency scenarios. InCV-LOCA has two possible scenarios which has a break point in one of the modular makeup purification systems (MMPS) lines. Fig1(b) shows the two pipelines connected to the i-SMR which are charging line (MMPS C/G, the lower pipeline) and let down line (MMPS L/D, the upper pipeline). InCV-LOCA occurs when at least one of these lines break inside the CV [4]. In this study, postulating INCV-LOCA with EDV, ERV all turned off, the scenario is firstly analyzed in base case. Secondly, variation for the sensitivity analysis with the lower plenum thermal hydraulic parameter is evaluated. The purpose of the evaluation is to show the importance of recirculation from the break area of MMPS C/G. With the various thermal hydraulic parameters provided, the RPV does not break due to heat removal achieved from the CV.

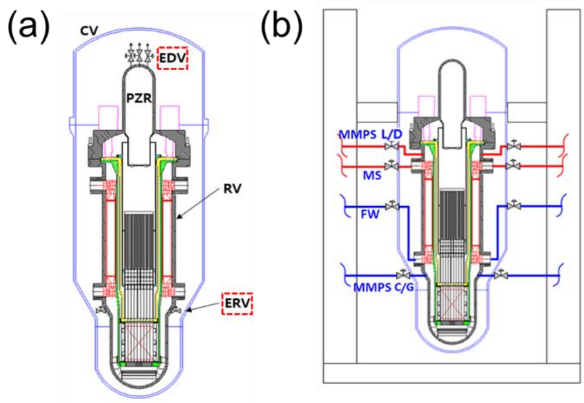


Figure 1 (a) schematic design of i-SMR (b) pipelines connected to i-SMR [5]

2. Methodology

2.1. System Code Description

The progression of the accident was evaluated using CINEMA (Code for INtegrated severe accident Evaluation and MAnagement), one of the severe accident analysis codes developed in Republic of Korea. CINEMA employs three main analysis modules to model the progression: CSPACE, SACAP, and SIRIUS. While SACAP and SIRIUS are utilized for analyzing ex-core phenomena and fission product behavior in large light water reactor nuclear power plants, CSPACE is primarily used for system thermal hydraulics and in-core phenomena analysis. Due to the smaller size of the i-SMR compared to large nuclear power plant, CSPACE is able to cover both RPV and CV thermal hydraulic interpretations. Hence, in this study, CSPACE version 2.0.2.343 is adopted for thorough thermal hydraulic analysis of the accident.

2.2. Base case Scenario Description

The progression of InCV-LOCA (MMPS C/G break size of 2 inch) is shown in Table 1. Due to recovery of the water level within the core, the progression of the severe accident ends without having the core materials damaged.

Table 1. Timeline of the accident

Events	Time
InCV-LOCA	0s
Rx, RCP, MFWP trip	31.7s
Control rod insert	32.23s
Oxidation start	2730s
CET > 923.15K	3182.8s

Figure 2~5 demonstrates the progression of the MMPS C/G line LOCA case. As the simulation of the accident begins, the mass flow through the break area is observed in figure 2.

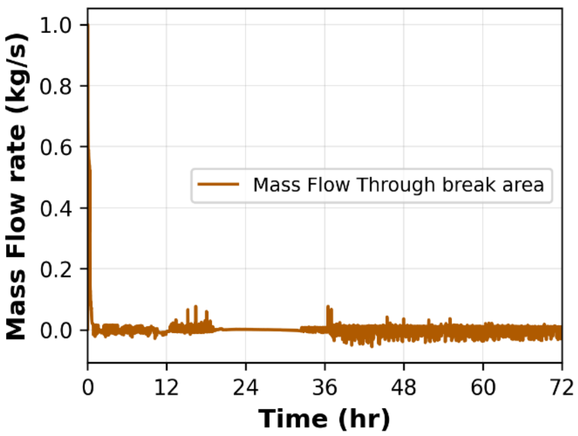


Figure 2 Mass flow rate through break area

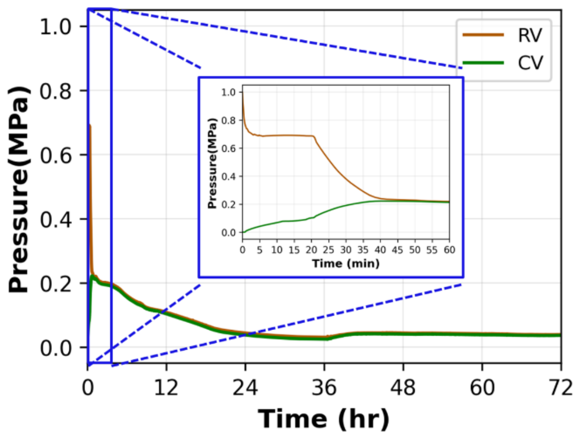


Figure 3 Pressure of RPV and CV

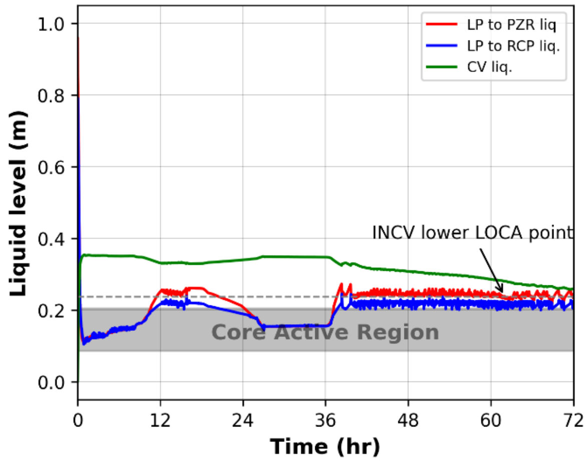


Figure 4 Water level within RPV and CV for 72 hours

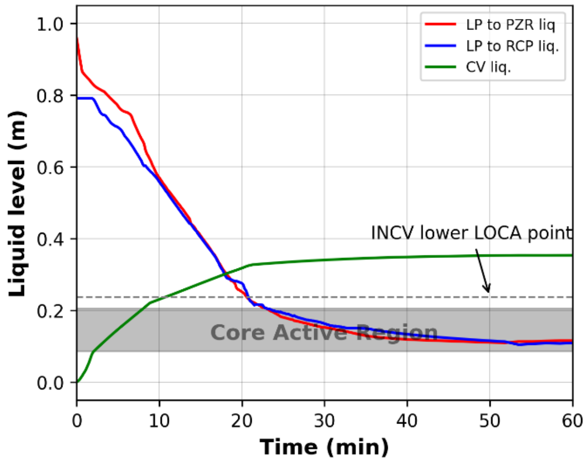


Figure 5 Water level within RPV and CV for 1 hour

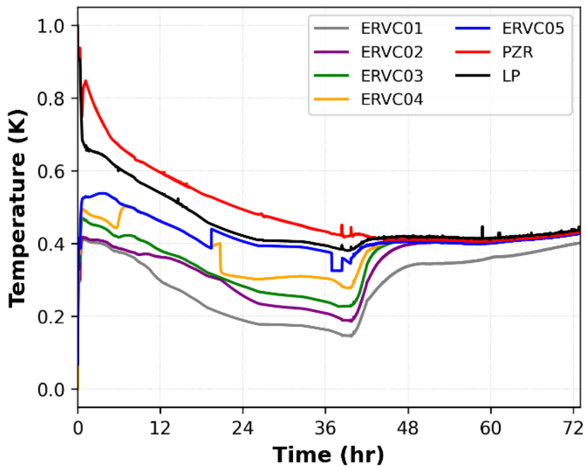


Figure 6 Liquid temperature of RPV and ERVC

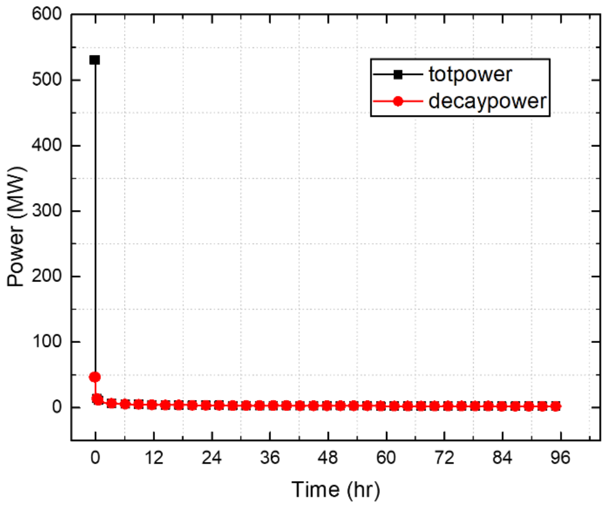


Figure 7 Total power and decay heat through whole progress

In the beginning of the scenario, the excessive amount of water flows from RPV to CV leading the water level inside RPV to fall and CV to rise. During this event, the water inside the RPV leaks significantly, exposing the core and leading to oxidation for having deficiency in heat removal. Also from figure 3, as the accident initiates at t = 0s, the pressure between the RPV and CV begin to equalize. The overall water level of RPV and CV are analyzed in figure 4. The green line in the figure shows the water level within the core region and the red line indicates the water level of the downcomer. As the water level from the CV increase enough to reach the break area, the water flows reversely from CV to RPV. After 20 hours of progression, the water level in RPV decreases due to evaporation from decay heat inside RPV. Such evaporation increases the pressure difference, due to increase in RPV pressure. Due to pressure difference, the water flow from CV to RPV has not been made even though CV had water level high enough to feed in. However, from figure 9, due to drop of pressure inside the RPV, the saturation temperature has also decreased. With RPV liquids staying underneath the saturation temperature, condensation could be achieved leading to increase in water level and pressure drop. Such effect gradually makes a gap for releasing through the break area. With the coolants recirculating heat removal by entering in through the break area, the temperature inside the core keeps decreasing.

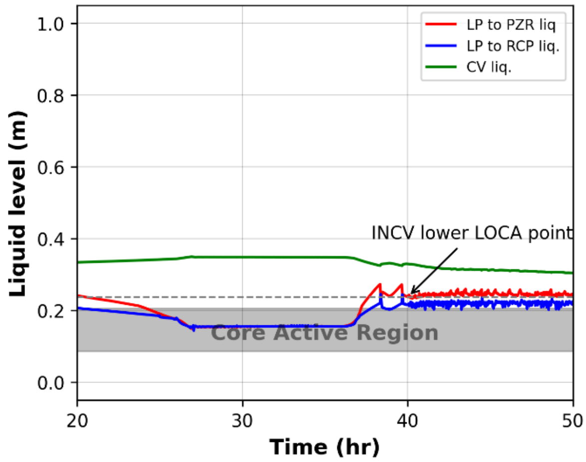


Figure 8 water level in RPV and CV from 20~50 hour

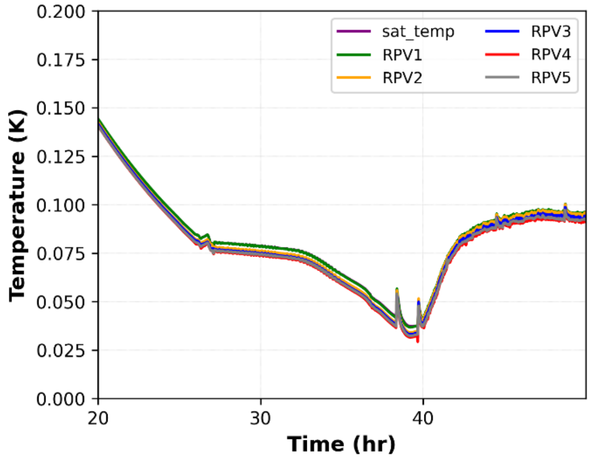


Figure 9 RPV fluid temperature with corresponding saturation temperature

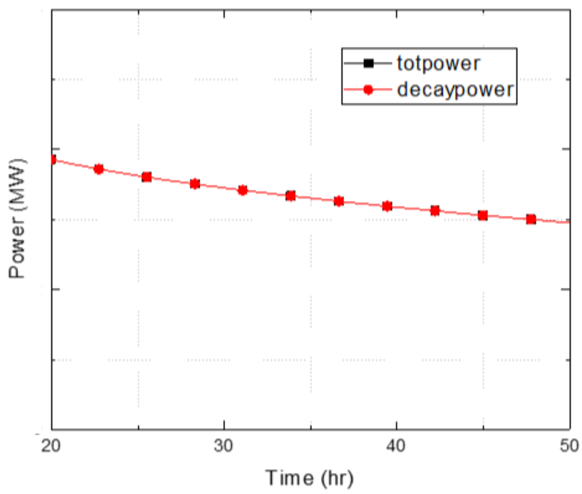


Figure 10 power in 20~50 hour range

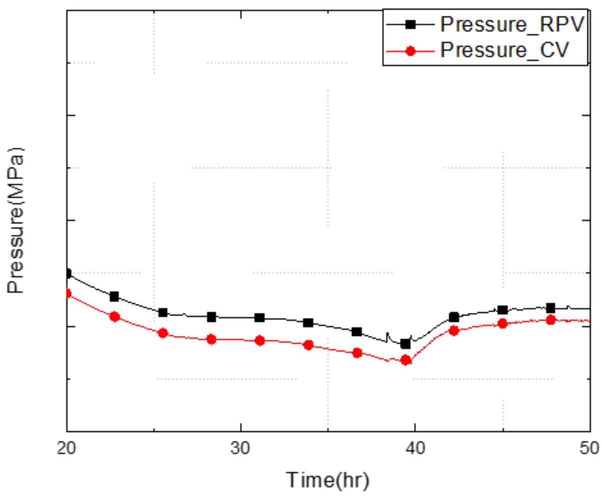


Figure 11 Pressure in 20~50 hour range

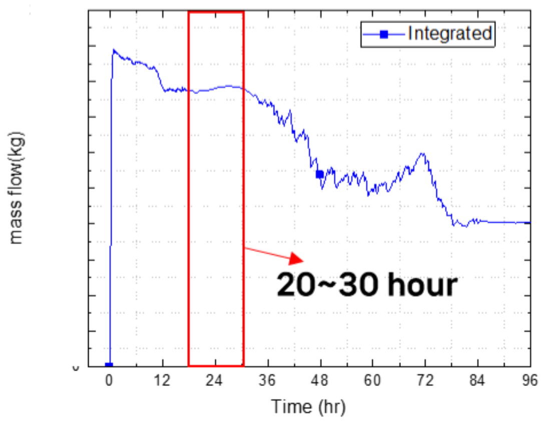


Figure 12 Integrated mass flow through break area

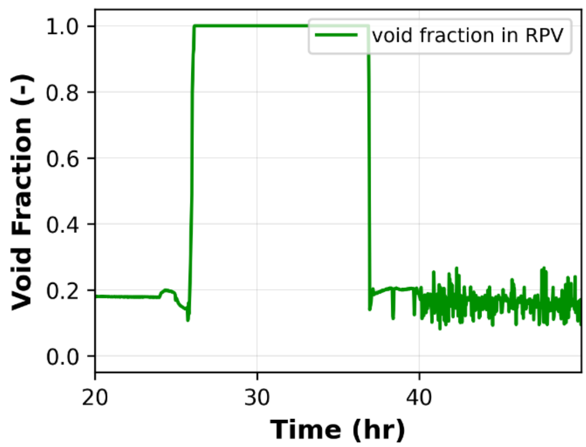


Figure 13 Void fraction within RPV

From figure 8~13, the graphs show specific details of progression in the scenario for the time range of 20 ~ 50 hour after the accident. From figure 8, time zone show needs for the analysis as the water level drops but no oxidation has occurred due to low temperature within this time zone. From figure 12, time range 20~30 hours the mass flow through the break area is no longer available. Within this time range, the temperature of the RPV decreases as demonstrated in figure 9 while the evaporation is still made due to the temperature being above the saturation temperature. Such vapor proportion detail is supported with the figure 13. From figure 6, the temperature difference between the RPV lower plenum (LP) and ERVC can be analyzed. With ERVC temperature being below RPV LP temperature, the heat is removed from RPV LP to ERVC. Such cooling leads to pressure drop within RPV as demonstrated in figure 11.

3. Result and Analysis

The progression of InCV-LOCA was analyzed. As a sensitivity study, three of the thermal-hydraulic parameters were selected. The parameters are, vessel out surface temperature (temp_lhv_o), ambient temperature of outer surface of lower plenum RPV (T_b), effective heat transfer coefficient of outlier surface of lower plenum RPV (H_b). Table 2 describes each parameter and its default value along with the range selected as a sensitivity analysis. The temperature range is set to be in a specific range. However, since the atmosphere temperature is set to be 20℃ (293K), the minimum temperature is set to be 293K.

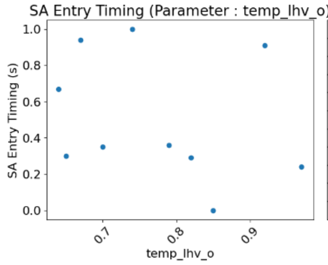
Table 2. parameter for sensitivity analysis

Parameter	Default	Range
Temp_lhv_o	-	293 K ~ 450 K
T_b	-	297 K ~ 477 K
H_b	-	1.0 ~ 100.0

With the Latin Hypercube Sampling (LHS) method, 10 values from each of the parameter range were randomly selected. In total, 30 scenarios were produced and analyzed. For analyzing the sensitivity, four of the figures of merit (FOM), SAMG entrance time, core uncover time, RPV failure and hydrogen production, are observed.

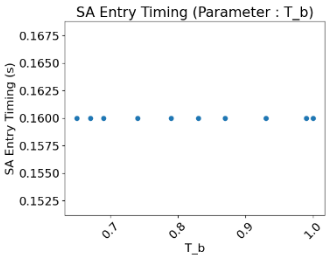
3.1. FOM 1: SAMG entrance time

In traditional severe accident management guidelines (SAMG), with respect to OPR1000, SAMG entrance temperature condition is set as 923.15K (650℃). The same guidelines cannot be the standard however, due to lack of plant-based data, assumption is made for setting the same temperature as a guideline. With setting this temperature as the guideline, each of the parameter value and the time reaching SAMG is demonstrated in the following figures.



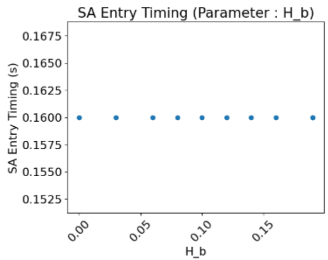
temp_lhv_o	SA Entry Timing (s)
0.64	0.67
0.65	0.3
0.67	0.94
0.7	0.35
0.74	1.0
0.79	0.36
0.82	0.29
0.85	0.0
0.92	0.91
0.97	0.24

Figure 14 FOM 1 with respect to vessel surface temperature



T_b	SA Entry Timing (s)
0.65	0.16
0.67	0.16
0.69	0.16
0.74	0.16
0.79	0.16
0.83	0.16
0.87	0.16
0.93	0.16
0.99	0.16
1.0	0.16

Figure 15 FOM 1 with respect to surrounding temperature.



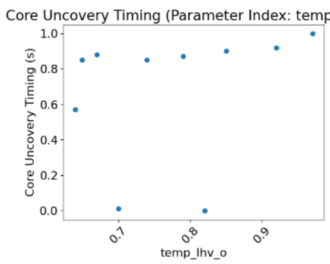
H_b	SA Entry Timing (s)
0.0	0.16
0.03	0.16
0.06	0.16
0.08	0.16
0.1	0.16
0.12	0.16
0.14	0.16
0.16	0.16
0.19	0.16
0.19	0.16

Figure 16 FOM 1 with respect to heat transfer coefficient

From figure 14 through 16, the temp_lhv_o show variation with SAMG entrance timing but T_b and h_b do not show any impacts on SAMG entrance timing.

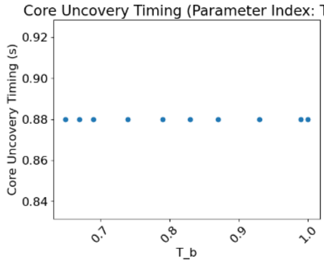
3.2. FOM 2: Cover uncover time.

From figure 4, the water level within the core has gone underneath the core region. The second FOM is the core uncover time. The core uncover time is defined as



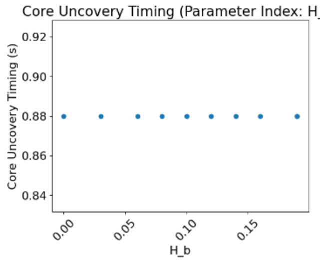
temp_lhv_o	Core Uncovery Timing (s)
0.64	0.57
0.65	0.85
0.67	0.88
0.7	0.01
0.74	0.85
0.79	0.87
0.82	0.0
0.85	0.9
0.92	0.92
0.97	1.0

Figure 17 FOM 2 with respect to vessel surface temperature



T_b	Core Uncovery Timing (s)
0.65	0.88
0.67	0.88
0.69	0.88
0.74	0.88
0.79	0.88
0.83	0.88
0.87	0.88
0.93	0.88
0.99	0.88
1.0	0.88

Figure 18 FOM 2 with respect to surrounding temperature



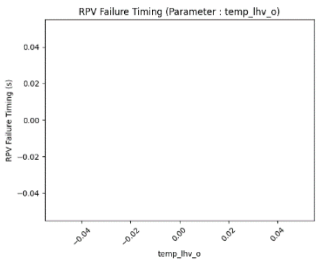
H_b	Core Uncovery Timing (s)
0.0	0.88
0.03	0.88
0.06	0.88
0.08	0.88
0.1	0.88
0.12	0.88
0.14	0.88
0.16	0.88
0.19	0.88
0.19	0.88

Figure 19 FOM 2 with respect to heat transfer coefficient

From figure 17 through 19, the temp_lhv_o show variation with core uncover timing but T_b and h_b do not show any impacts on SAMG entrance timing.

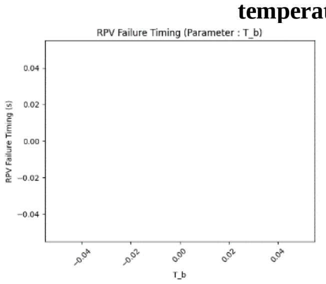
3.3. FOM 3: RPV failure time

The sensitivity analysis of two FOMs show that the h_b and T_b have no impact. Additionally, for this case scenario, no valid data could be found for fuel melt due to coolability reached inside the core. Hence, two of the main FOMs, accumulated corium mass and RPV failure time could not be investigated from the data.



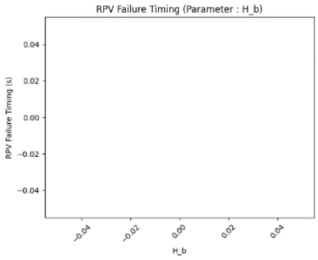
temp_lhv_o	RPV Failure Timing (s)
294.364	nan
299.455	nan
306.437	nan
321.811	nan
340.123	nan
363.073	nan
375.533	nan
391.485	nan
423.526	nan
447.474	nan

Figure 20 FOM 3 with respect to vessel surface temperature



T_b	RPV Failure Timing (s)
299.157	nan
308.746	nan
318.538	nan
342.279	nan
363.667	nan
383.876	nan
398.908	nan
429.509	nan
455.204	nan
459.625	nan

Figure 21 FOM 3 with respect to surrounding temperature



H_b	RPV Failure Timing (s)
1.617	nan
15.542	nan
27.518	nan
40.146	nan
48.275	nan
54.873	nan
65.904	nan
76.403	nan
88.235	nan
90.531	nan

Figure 22 FOM 3 with respect to heat transfer coefficient

3.4. FOM 4: Accumulated Hydrogen Mass

Hydrogen mass is analyzed due to the behavior of detonation and deflagration during the accident. For hydrogen, there exist detonation region, complete combustion region, and ignition region which all are critical for nuclear power plant.

Figure 23 FOM 4 with respect to all coefficient

From figure 23, the temp_lhv_o show variation with hydrogen production but T_b and h_b do not show any impacts on hydrogen production mass.

4. Conclusion

In this study, one of the severe accidents was selected and analyzed with three of the parameter's sensitivity. Due to recirculation of the coolants through the break area, the core temperature was able to maintain the safe condition which enables the reactor to be in adequate condition for cool shutdown. For the future study, the additional analysis for different case scenario is to be made for supporting robustness in the design of i-SMR. Investigation is expected to be made with the wider range of FOMs and parameters to analyze the uncertainty.

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