Proceedings of the Korean Nuclear Society Spring Meeting
Kori, Korea, May 2000

A Simulation of Forced Cooling of a Reactor Vessel during the Severe Accidents by RELAP5/MOD3 Computer Code

Jin Ho Song, S.B. Kim, H. D. Kim
Korea Atomic Energy Research Institute,
P. O. Box 105, Yusong Post Office, Taejon, 305-600, Korea

Abstract

One of the proposed accident management strategies for severe accident is the In-Vessel Retention (IVR) of molten fuel. As a means of IVR strategy, the reactor vessel is cooled from the outside either by a passive natural circulation or a forced circulation. In the present paper, the integral behavior of two-phase flow dynamics and heat transfer in the flow channel along the reactor vessel for the forced circulation case is analyzed by RELAP5/MOD3 computer code for the Korean Standard Nuclear Power Plant (KSNPP). The case with a flow rate of 315 kg/s, an uniform heat flux of 600 Kw/m^2, and pool water at the saturated condition is selected as a base case. Parametric studies on the flow rate, heat-flux, and the sub-cooling of the pool water are then performed. For the base case, the heat transfer mode in the flow channel is in either sub-cooled boiling or saturated boiling. The void fraction in the flow channel is below 5%. When the flow rate is further reduced below 160 kg/s, the flow regime moves to the film-boiling region and the wall temperature increases above the melting temperature. As the sub-cooling of the pool water increases, the required capacity of forced flow to maintain heat transfer mode in the sub-cooled or saturated nucleate boiling decreases. As the containment can be pressurized during the severe accident, the case with containment pressure of 0.3 MPa is analyzed. Comparisons with a case at 0.1 Mpa indicate that the pressure increase is beneficial for the cooling performance. The results of the present analyses are justified in the sense that the RELAP5 predicts the overall system behavior and the local CHF correlation based on the AECL look-up table is a good first order approximation.

1. Background

One of the proposed accident management strategies for the severe accidents is the In-Vessel Retention (IVR) of molten fuel. The feasibility of the strategy is investigated for the AP600 design [1] and Korean Next Generation Reactor (KNGR) design [2]. The reactor vessel can maintain its structural integrity, if the decay heat generated from the molten fuel pool can be effectively removed. As a means of the IVR strategy, the reactor cavity is flooded by cold water. The reactor vessel cooling is performed by a passive natural circulation flow in a cavity. This passive cooling mechanism is desirable, as it would not call for operator action.

The maximum heat removal capability is determined by both the local phenomena of the critical heat flux on the downward facing walls and the overall system behavior. The local maximum heat flux is limited by the critical heat flux to prevent film boiling. If the heat transfer mechanism becomes film boiling, the reactor vessel wall temperature will increase above the melting temperature. There has been research on the critical heat flux on downward facing walls [3,4,5]. They indicated that the critical heat flux is lowest at the bottom of the reactor vessel and it increases as the inclination angle of the wall increases. However, as the local hydrodynamic and thermodynamic condition along the heated channel is determined by the overall system behavior, the system behavior also needs to be investigated. The overall system behavior of the boiling phenomena outside the full-scale reactor vessel [5] and the boiling phenomena and critical heat flux condition in a simplified geometry [6] were experimentally investigated.

As discussed by Song [7], if the heat load is high enough, natural circulation cooling is not feasible. The vessel wall temperature would exceed the melting temperature of the reactor vessel wall as the heat transfer mechanism becomes transition boiling or film boiling. An additional engineered safety feature is necessary to enable the IVR strategy. One of the proposed concepts of COASISO [8] employs the use of forced convection cooling of the reactor vessel by using the pumps available during the accident. The conceptual design is sketched in Figure 1. Before the molten fuel is relocated to the bottom of the reactor vessel, the forced flow is supplied through the channel that encloses the reactor vessel. The suction can be taken from the bottom of the flooded reactor cavity and the forced flow is supplied to the channel along the reactor vessel wall by the available pumps, such as the containment spray pump. If we have enough source of cold water provided by a water pond or fire hose, sub-cooled water can be continuously supplied. The feasibility of this concept is governed by the competition between the heat load along the heated vessel wall due to the decay heat from the molten fuel pool and the heat removal capability provided by the forced flow supplied by the containment spray pump. The geometry of the flow path is a constraint depending on the shape of the reactor vessel and cavity.

The feasibility of forced convection flow to enable IVR can be assessed by experiments on a scaled test facility or by an analytical approach. The analytical approach using computer code has an advantage in the sense that it can simulate a prototypic condition, though the various heat-transfer and hydrodynamic correlation would provide only a good first order approximation of the situation under consideration. In the present study, we analyzed the general concept of forced convection cooling of the reactor vessel including COASISO, during a severe accident by using the state-of-the-art-computer code, RELAP5/MOD3 [8]. As RELAP5 enables parametric studies, it would provide valuable insight on the sensitivity of various system parameters. It is also intended to be used for determining the success criteria of the IVR strategy and the major scaling parameters for the experimental facility.

2. Analytical Models

The RELAP5 computer code employs a one-dimensional, transient, two-fluid model for the flow of a two-phase steam-water mixture. The two-fluid equations of motion are formulated in terms of volume and the time averaged parameters of the flow. The heat transfer model employs a heat transfer correlation in the whole range of the heat transfer regime, including boiling heat transfer. As it is basically a one-dimensional model, it has a limitation in simulating multi-dimensional natural circulation in a pool. However, when the forced convection flow is provided via a flow channel along the reactor vessel, the overall system behavior can be approximated in a one-dimensional fashion. The RELAP5/MOD3 model is constructed for ex-vessel cooling. The inlet-flow provided by the containment spray pump is modeled as a constant flow rate junction. The pool water is modeled as a big pool, the thermodynamic condition of which can be changed by user input. The flow channel is modeled as 21 nodes. The hemispherical portion of the reactor vessel is accounted for by a series of nodes with a variable flow area and inclination angle. The heated channel consists of 13 heated nodes and 8 unheated nodes. The heated wall is a portion of the reactor vessel, which encloses the molten fuel pool. The constant heat flux condition is applied to the inner wall of the vessel. The RELAP5/MOD3 node flow path network is shown in Figure 2. The height of the flow channel is 7.7 m. The gap size is 0.2 m. The length of the heated wall is 3.7m. The node flow path net work is the same as one employed in reference 6.

When the fuels are melted and relocated to the bottom of the reactor vessel during a severe accident, the decay heat generated from the molten fuel pool induces natural convection flow inside the molten fuel pool. The outside of the pool is assumed to be at the constant melting temperature. The heat flux to the outside of the vessel is determined by the natural convection pattern and the heat transfer mechanism outside the reactor vessel wall. Previous research indicated that the heat flux in the upper part of the steel layer is much bigger than the lower part due to the focusing effect [1]. For simplicity, we assumed uniform heat flux from the reactor vessel wall. The heat flux corresponding to this decay heat is about 770 KW/m^2 for KNGR, which is a 1400 MWe advanced Pressurized Water reactor (PWR) under design. It is equivalent to the decay heat of 40 MW, which corresponds to the decay heat of KNGR at 10000 seconds. The value corresponding to that of the Korean Standard Nuclear Power Plant (KSNPP), which is 1000 MWe operating PWR, is 600 KW/m^2. As the containment spray is one of the large capacity pumps used in the power plant, we selected it as a source of forced flow. The capacity of the containment spray pump is 5000 gpm for KSNPP. Depending on the flow rate and the wall heat generated from the molten fuel pool, the heat transfer mode inside the flow channel is determined. The reactor vessel wall temperature can be maintained below the melting temperature, when the heat transfer mode does not evolve to the transition boiling or film-boiling regime. The heat load on the reactor vessel wall due to the decay heat should be smaller than the local critical heat flux.

As the geometry of the reactor vessel wall is multi-dimensional and curved in shape, the existing correlation of CHF in a simple geometry cannot be directly applied. Therefore, there was a lot of effort to investigate the CHF mechanism for downward facing curved walls [3,4,5,6]. The CHF is dependent on the local thermodynamic and hydrodynamic conditions. As the local condition is determined by the overall system behavior, the overall system behavior should be appropriately evaluated. Though the CHF mechanisms are quite different from AECL look-up table suitable for the vertical pipes of a fuel bundle implemented in the RELAP5/MOD3, the RELAP5 correlation would provide a good first order approximation.

3. Analysis Results

The flow rate of the two-containment spray pump at 5000 gpm for the KSNPP corresponds to 630 kg/s. The KSNPP case with a 600 KW/m^2 uniform heat-flux from the reactor vessel is considered. The case with one containment spray (315 kg/s) and a saturated pool water condition is selected as a base case. Parametric studies on the flow rate, heat-flux, and the sub-cooling of the pool water are then performed. As an initial condition, the heated channel is filled with sub-cooled liquid. The heat flux input is provided as ramp input, which reaches the full power condition in 300 seconds, which gave better numerical convergence to the steady state.

3.1 Base Case

The fluid temperature in the heated channel and void fraction is shown in figures 3 and 4 respectively. The heated channel has 13 volumes. The temperatures in the first, ninth, and the last volume are shown. The fluid temperature reaches a steady state at about 400 seconds. The vessel wall temperature is maintained below 460K, which indicates that the heat transfer mode is either sub-cooled boiling or saturated boiling. The void fraction in the flow channel is below 5%, and the flow regime in the channel is bubbly flow.

If the forced flow rate is decreased below 300 kg/s, the heat transfer mode of the flow in the flow channel swings between saturated transition boiling and saturated nucleate boiling. When the flow rate is further reduced below 160 kg/s, the flow regime moves to the film-boiling region and the wall temperature increases above the melting temperature. As it deteriorates the structural integrity of the reactor vessel, it would jeopardize the IVR strategy. Figures 5 and 6 shows the fluid temperature in the heated channel and the void fraction for the case of 160 kg/s.

The flow regime at the bottom of the flow channel is either a horizontal stratified flow regime or bubbly flow regime. As the bottom portion of the flow channel is nearly horizontal, it becomes a horizontal stratified flow regime. It changes to slug flow in the upper portion of the flow channel.

3.2 Effect of sub-cooling of the pool water

The pool water can maintain appropriate sub-cooling either by the external supply of water or by external cooling. As shown in Table 1, as the sub-cooling of the pool increases, the required capacity of forced flow to maintain the heat transfer mode in the sub-cooled or saturated nucleate boiling decreases. The wall temperature and the void fraction at the top of the flow channel for each case are compared in Figure 7 and 8 respectively. The trends of the reactor wall temperature with a certain degree of sub-cooling indicate that the heat transfer regime moves directly to film boiling without much oscillation. That of the saturated condition shows quite long oscillatory behavior and then move to the film-boiling region. The void fractions indicate that the flow channel is voided earlier in the transient for the saturated case, as expected.

Table 1 Results of Analyses for the KSNPP Case

Water Temperature	Saturated (373 K)	Sub-cooled (348K)	Sub-cooled (323K)
Required Flow Capacity to maintain saturated boiling (kg/s)	300	170	90
Failure criteria (kg/s)	160	140	80

3.3 Effect of Thermal Load

The cases with a higher heat flux of 770 KW/m^2 representing the KNGR are considered with a variable flow rate and sub-cooling. The case with a saturated pool temperature and the flow rate of 630 kg, which corresponds to the two containment spray pump flow rate, is considered first. The results of the analysis are shown in Figures 9 and 10. It is shown that the wall temperature increases to about 480 K and then returns to 440K in a cyclic manner. This is due to the change in the heat transfer mode from transition boiling to saturated nucleate boiling. The transition-boiling mode is changed from nucleate boiling to film boiling when the wall temperature is higher than the saturation temperature by 100K. The increased heat transfer rate causes the change in the heat transfer mode to the nucleate boiling again. This change repeats its cycle as shown in the Figure. The void fraction in the upper channel reaches about 0.8 and the flow regime is slug flow. As the transition boiling is a rather unstable heat transfer mode, we investigated the minimum required flow capacity to maintain the heat transfer mode within the nucleate boiling mode. The results of the parametric studies are shown in Table 2. Table 2 indicates that the effect of sub-cooling shows the same trends as the previous case. For the saturated pool water condition, the required flow rate to maintain the heat transfer mode in the nucleate boiling is obtained as 800 kg/s. The flow rate of 800 kg/s is very close to the maximum containment spray flow rate of the KNGR, which is 820 kg/s (a two containment spray flow rate of 13000 gpm). These results indicate that the containment spray flow rate is very marginal for the saturated flow case.

Table 2 Results of the Analyses for the KNGR Case

Water Temperature	Saturated (373 K)	Sub-cooled (348K)	Sub-cooled (323K)
Required Flow Capacity to maintain saturated boiling (kg/s)	800	220	140
Failure criteria (kg/s)	270	210	130

3.4 Containment Pressure Effect

During a severe accident, depending on the precursor event, the containment can be pressurized up to a typical post Loss of Coolant Accident (LOCA) or Main Steam Line Break (MSLB) pressure, which is about 4 bar. We assumed that the pool is maintained at atmospheric pressure in the cases discussed above. As the boiling heat transfer rate increases as the containment pressure increases, it is expected that the cooling performance will be enhanced as the containment pressure increases. The typical results of the case with a containment pressure of 0.3 MPa and saturated pool water are shown in the Figures 11 and 12. The flow rate is assumed to be 630 kg/s. If we compare it with the case at 0.1 Mpa, we can clearly see that the pressure increase due to steam generation in the flow channel helps the cooling performance. The wall temperature is about 470 K, which is due to the change in saturation temperature. The void fraction at the exit of the heated flow channel is about 0.6.

3.5 Feasibility of Ex-Vessel Cooling

The results of the analyses discussed above indicate that forced cooling is effective for the KSNPP type reactor. As reference 7 demonstrated that the natural circulation cooling is capable of removing decay heat for the KSNPP type reactor, we can rely on either forced cooling or natural circulation cooling. However, for the KNGR type reactor natural circulation was not feasible, as discussed in the reference 7. The results of the present analyses indicate that the forced circulation by the two containment spray pumps was marginal. If we can maintain appropriate sub-cooling for the water in the reactor cavity by external means, we can cool the reactor vessel with a reasonable flow rate. It is possible that we cannot avoid vessel melting in the KNGR case.

4. Summary and Conclusion

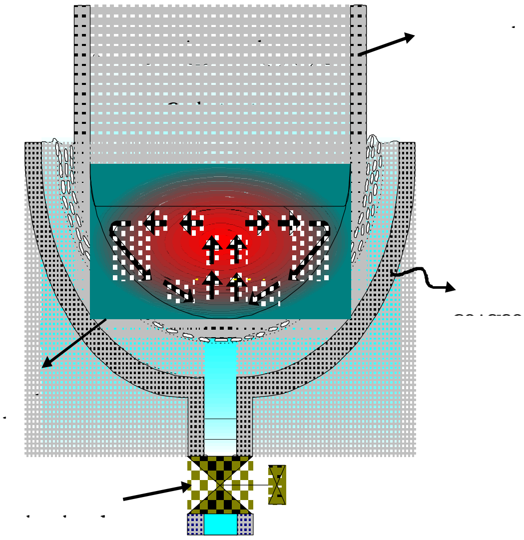
The present paper provides the results of analysis for the integral behavior of ex-vessel cooling during a severe accident by the RELAP5/MOD3 computer code. The case with a flow rate of 315 kg/s, which corresponds to the capacity of one containment spray pump for the KSNPP, with a uniform heat flux of 600 KW/m^2, and pool water at the saturated condition was selected as a base case. Parametric studies on the flow rate, heat-flux, and the sub-cooling of the pool water were then performed.

The analyses results indicate that forced circulation is feasible for the KSNPP type reactor. On the other hand, even forced circulation by the two containment spray pumps is marginal for the KNGR. Additional engineered safety features might be necessary for the KNGR type reactor. The present analysis gives insight to the integral behavior of two-phase flow dynamics and heat transfer, and it is beneficial, because it can simulate prototypic conditions that are hardly achievable in experiments.

Acknowledgement: This research is supported by the Ministry Of Science and Technology of the Korean Government.

References

1) T. G. Theofanous, C. Liu, S. Additon, S. Angelini, O. Kymalainen, and T. Salmassi, In-vessel coolability and retention of a core melt, DOE/ID-10460, Center for Risk Studies and Safety, University of California, Santa Barbara.
2) S. J. Oh, The role of IVR from the severe accident management perspective, Workshop on In-Vessel Retention for the KNGR Severe Accident Management, 1998, KEPRI.
3) T. G. Theofanous and S. Syri, The coolability limits of a reactor pressure vessel lower head, Nuclear Engineering and Design 169, pp. 59-76, 1997.
4) F. B. Cheung and K. H. Haddad, A hydrodynamic critical heat flux model for saturated pool boiling on a downward facing curved heating surface, Int. J. Heat Mass Transfer, Vol. 40, No. 6, pp.1291-1302, 1997.
5) T. Y. Chu et. al., Ex-vessel boiling experiments: laboratory- and reactor-scale testing of the flooded cavity concept for in-vessel core retention Part II: Reactor-scale boiling experiments of the flooded cavity concept for in-vessel core retention, Nuclear Engineering and Design 169, 1997, pp. 89-99.
6) S. Rouse, SULTAN test facility for large-scale vessel coolability in natural convection at low pressure, Nuclear Engineering and Design 169, 1997, pp. 185-195.
7) J. H. Song, S. B. Kim, H. D. Kim, The analysis of the natural circulation cooling of the reactor vessel during the severe accident by RELAP5/MOD3 computer code, Proceedings of the Korean Nuclear Society Spring Meeting, Kori, Korea, May 2000.
8) H. J. Yoon et. al. In-Vessel Retention Capability of the COASISO Design, The 14th KAIF/KNS Annual Conference. Seoul, Korea, April 8~9, 1999.
9) Scientech, Inc., RELAP5/MOD3 Code Manual, Volume I: Code structure, system models, and solution methods, 1998.



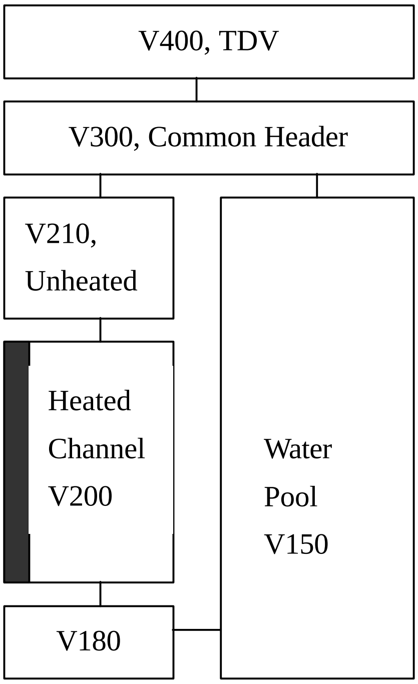
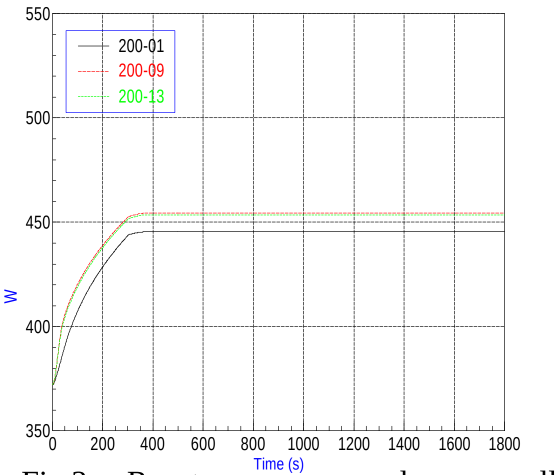


Fig. 1 Conceptual Diagram of COASISO
(Taken from reference 8)

Fig. 2 RELAP5 Node-Flow Path Network



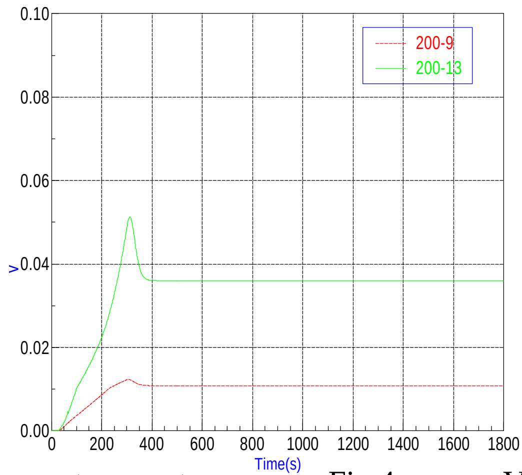


Fig.3 Reactor vessel wall temperature

Fig.4 Void fraction in the flow channel

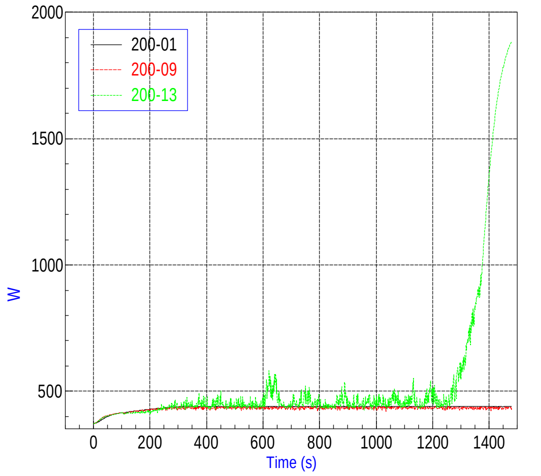


Fig.5 Reactor vessel wall temperature

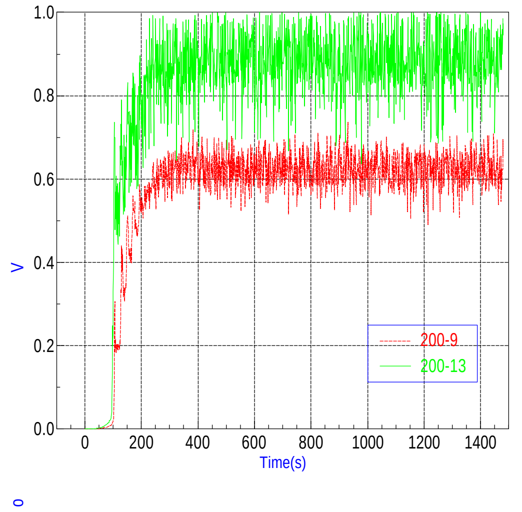


Fig.6Void fraction in the flow channel

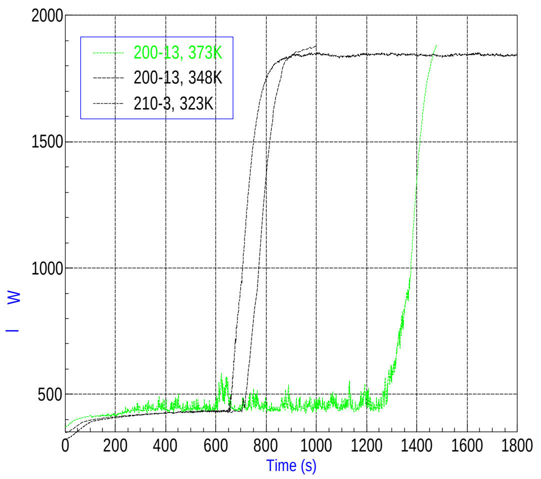


Fig.7 Reactor vessel wall temperature

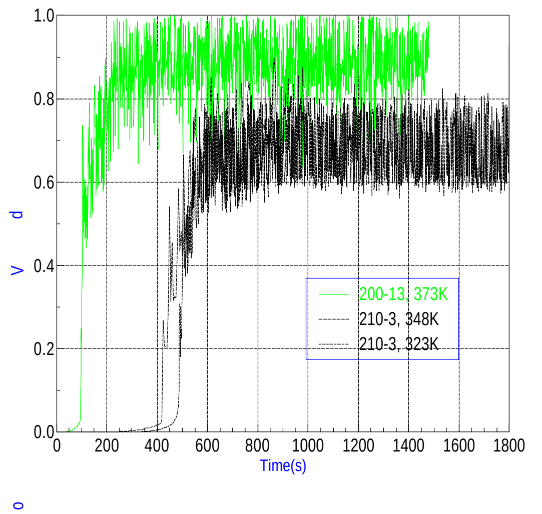


Fig.8 Void fraction in the flow channel

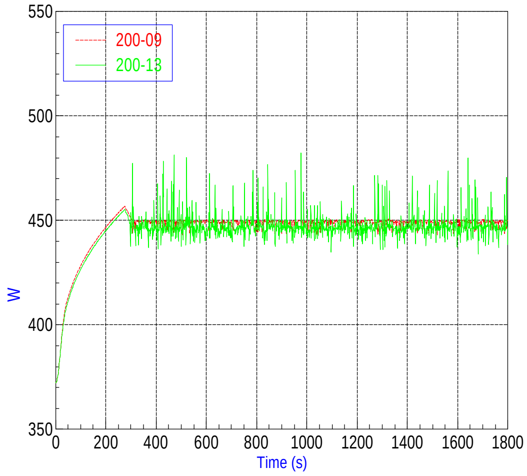


Fig.9 Reactor vessel wall temperature

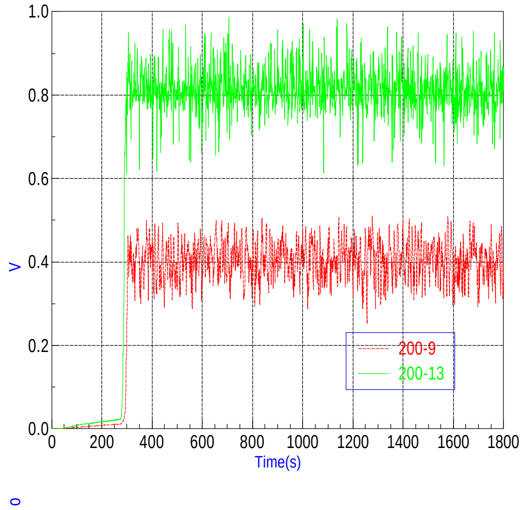


Fig.10 Void fraction in the flow channel

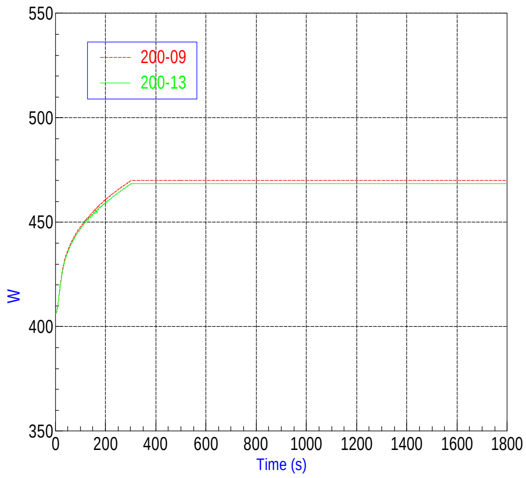


Fig.11 Reactor vessel wall temperature

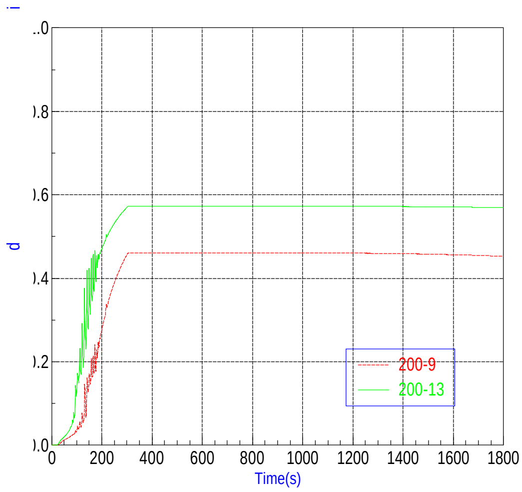


Fig.12 Void fraction in the flow channel