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Thermal-Mixing Analyses for Safety Injection at Partial Loop Stagnation of a Nuclear Power Plant

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When a cold HPSI (High Pressure Safety Injection) fluid associated with an overcooling transient, such as SGTR (Steam Generator Tube Rupture), MSLB (Main Steam Line Break) etc., enters the cold legs of a stagnated primary coolant loop, thermal stratification phenomena will arise due to incomplete mixing. If the stratified flow enters the downcomer of the reactor pressure vessel, severe thermal stresses are created in a radiation embrittled vessel wall by local overcooling. As general thermal-hydraulic system analysis codes cannot properly predict the thermal stratification phenomena, RG 1.154 requires that a detailed thermal-mixing analysis of PTS (Pressurized Thermal Shock) evaluation be performed. Also, previous PTS studies have assumed that the thermal stratification phenomena generated in the stagnated loop side of a partially stagnated primary coolant loop are neutralized in the vessel downcomer by the strong flow from the unstagnated loop. On the basis of these reasons, this paper focuses on the development of a 3-dimensional thermal-mixing analysis model using PHOENICS code which can be applied to both partial and total loop stagnated cases. In addition, this paper verifies the fact that, for partial loop stagnated cases, the cold plume generated in the vessel downcomer due to the thermal stratification phenomena of the stagnated loop is almost neutralized by the strong flow of the unstagnated loop but is not fully eliminated.

Key Words : Thermal Stratification, HPSI (High Pressure Safety Injection), Thermal-Mixing Analysis, Partial Loop Stagnation

1. Introduction

When an accident, such as SGTR (Steam Generator Tube Rupture), MSLB (Main Steam Line Break), etc., arises in a nuclear power plant, a cold HPSI fluid enters the primary coolant loops to protect the core. The HPSI fluid can be well mixed or stratified in the cold legs and the reactor vessel downcomer depending on fluid conditions and density differences between the cold HPSI fluid and the hot coolant in the cold legs. If the stratified flow enters the reactor vessel downcomer, the cold plume quickly cools down a section of the reactor vessel below the bulk average fluid temperature, which increases the failure probability due to severe thermal stresses. Thus, thermal-mixing behavior is a key element to assess the effect of PTS based on the reactor vessel integrity, because it primarily determines the effective driving force for asymmetric thermal transients into the reactor vessel wall. To account for these important effects, much more refined and sophisticated computer programs and/or engineering calculation methods are needed.

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Due to the nature of the calculation and nodal limitation, the general thermal-hydraulic system analysis programs do not properly predict the thermal stratification phenomena in the cold legs caused by the thermal-mixing phenomena in the vessel downcomer (Theofanous et al., 1984 ; Choi et al., 2000). Therefore, if the thermal stratification phenomena are formed in reactor coolant loops, Regulatory Guide (RG) 1.154 in terms of PTS evaluation requires that a detailed thermal-mixing analysis should be performed (RG 1.154, 1987).

When the cold HPSI fluid enters the reactor coolant loops due to overcooling transients, the loops are partially or totally stagnated dependent on transient scenarios. In the partial loop stagnated case, previous PTS studies have used the bulk downcomer temperatures in PFM (Probabilistic Failure Mechanics) analysis which were predicted by thermal-hydraulic system analysis programs (Robinson, 1985 ; CCNPP, 1985b).

However, in case the thermal stratification phenomena are generated in just one loop due to the partial loop stagnation, it is recommended to estimate the downcomer temperature distribution by thermal-mixing analysis because the thermal-hydraulic system analysis programs may not reliably calculate the downcomer temperature. Based on these reasons, a 3-dimensional thermal-mixing analysis model consisting of half of each cold leg and half of the reactor vessel was developed to identify the flow conditions for partial and total loop stagnated cases and to improve the reliability of the thermal-mixing analysis. The thermal-mixing analysis model was developed using PHOENICS code. This paper describes the developed model composition, verification, and analysis results for partial and total loop stagnated cases.

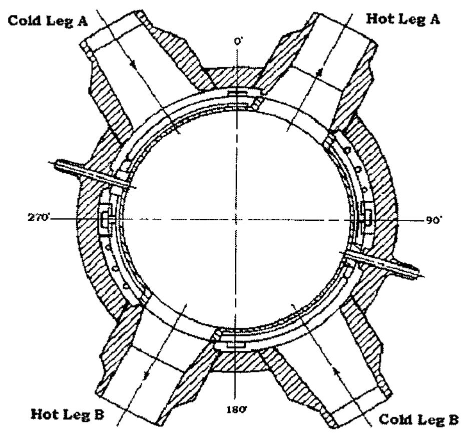

Fig. 1 Top view of RPV

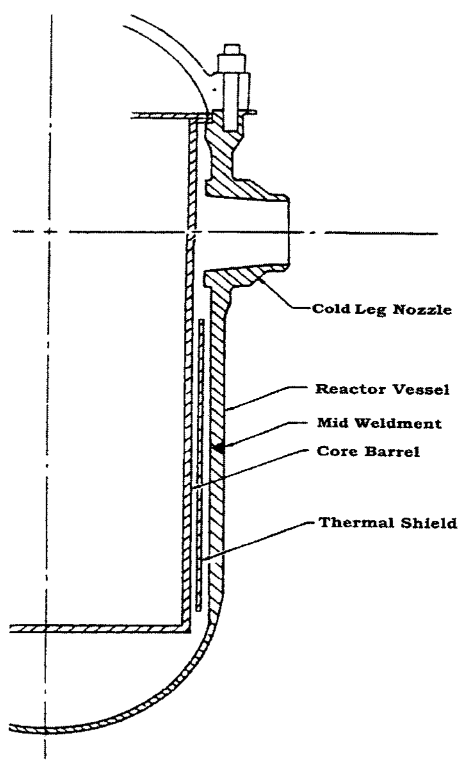

Fig. 2 Elevation view of RPV

2. Model Description and Verification

2.1 Model description

The model plant is a 2-loop PWR (Pressurized Water Reactor) nuclear power plant. Fig. 1 shows a top view of the reactor vessel with the two hot legs and two cold legs. Fig. 2 shows an elevation view of the reactor vessel downcomer region with a cold leg nozzle. To simplify the analysis, a model consisting of half of each cold leg and the reactor vessel was developed. The sections included in this model are the loop seal, cold leg, core barrel, thermal shield, and reactor vessel wall. In order to simplify the modeling of

Table 1 Geometric data

(Unit: cm)

Items	Cold Leg	Vessel/ Downcomer	Lower Plenum	Pump	Loop Seal	Core Barrel	Thermal Shield
Inner Diameter	69.86	335.28	—	—	78.73	276.85	292.80
Length	519.68	610.88	—	—	939.18	772.79	457.20
Base Metal Wall Thickness	6.77	16.52	10.49	0.386	7.56	5.03	9.05
Clad Thickness	0.00	0.30	0.30	—	0.00	—	—
Insulation Thickness	7.62	7.62	7.62	—	7.62	—	—
Fluid Volume, $(cm^3) \times 10^{-6}$	1.99	5.26	10.39	3.18	4.57	—	—

* Inner diameter of safety injection piping : 13.17 cm

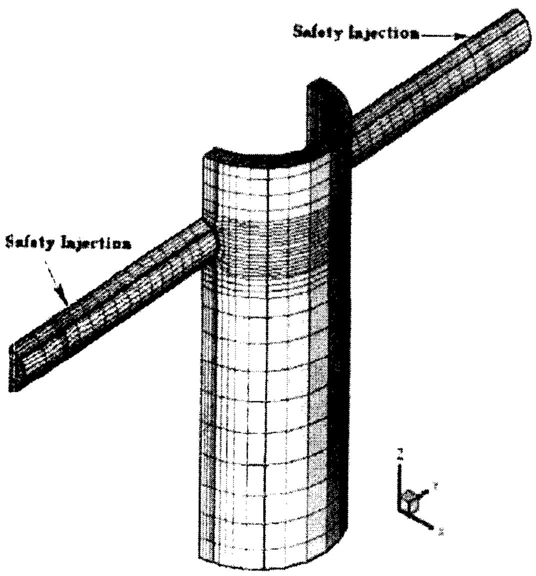

Fig. 3 Cell divisions

the loop seal, the volume was just included in this model. Even though the coolant injected through the cold legs is discharged to hot legs through the lower plenum and core, those sections were not considered in this model because the bottom of downcomer was selected as the outlet.

And, the safety injection piping was modeled as located directly on top of the cold leg piping to simplify the analysis. The relevant geometric data of the reactor vessel configuration are provided in Table 1. Fig. 3 illustrates the cell divisions based on BFC (Body Fitted Coordinate) methods which have a total 24, 304 computational cells (x x y x z=28×28×31). In the reactor coolant loop, there is a horizontal bend in the cold leg piping. Previous studies have shown that there was little effect on the calculated velocity and temperature profiles adjacent to the vessel wall when a cold leg with a horizontal bend was modeled as straight pipe (EPRI, 1984). Since it makes little difference in the calculations, the horizontal bends in the cold legs were modeled as straight pipes in order to save the input preparation time. Turbulence was incorporated in the code calculations through the use of Yap KE-EP model. The coolant flow into the reactor vessel downcomer may be separated by the thermal shield and recombined in the lower plenum. Even though calculation time is much more than with other turbulence models, the separated and recombined flow behavior can be well predicted by Yap KE-EP model (CHAM, 1992). And, the Boussinesq approximation was used in the thermal-mixing analysis model in order to account for the buoyancy effects.

2.2 Model verification

To verify the developed thermal-mixing analysis model, the CREARE 1/2 scale test facility consisting of one cold leg and half of the reactor vessel was modeled using PHOENICS code. The CREARE test (May 105) was sponsored by the NRC and EPRI (Theofanous and Yan, 1990). For the test, the safety injection piping was connected to the top of the cold leg and the thermal shield was included in the test facility. The geometric data of the facility are given in Table 2 and

Table 2 Geometric data of CREARE 1/2 Scale test facility (Inner diameter of safety injection piping : 11.43 cm)

(Unit : cm)

Items	Cold Leg	Vessel/ Downcomer	Lower Plenum	Pump	Loop Seal	Core Barrel	Thermal Shield
Inner Diameter	36.32	–	–	–	38.10	–	–
Length	377.60	353.15	–	–	272.41	353.15	243.54
Base Metal Wall Thickness	2.10	7.00	0.60	–	2.10	7.00	3.81
Clad Thickness	–	–	–	–	–	–	–
Insulation Thickness	5.10	5.10	5.10	5.10	5.10	5.10	–
Fluid Volume, (cm3) x 10-5	4.07	6.38	6.05	2.72	3.11	–	–

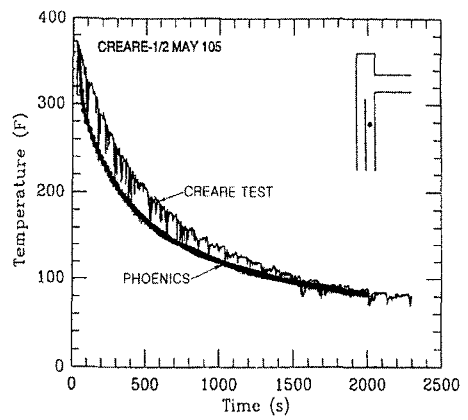

Fig. 4 Comparison of temperature under cold leg

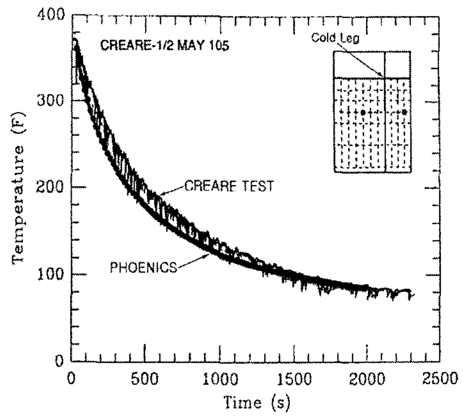

Fig. 5 Comparison of temperature beside and under cold leg

the test conditions are as follows :

- Coolant flow is stagnated
- Density of coolant is 877.8 Kg/m^3
- Cross section area of safety injection piping is 0.01 m^2
- Injection flow rate of safety injection fluid is 5.17 Kg/sec
- Density of safety injection fluid is 1,000 Kg/m^3

Figures 4 and 5 show the comparisons between the CREARE 1/2 scale test results and the calculation results using PHOENICS code. The comparison parts are under the cold leg as shown in Fig. 4 and beside and under the cold leg as shown in Fig. 5, respectively, which are activated by safety injection flow. In both cases, the prediction appears to properly represent the recorded data of the CREARE 1/2 scale test.

2.3 Boundary conditions

Boundary conditions for the analyses were taken from the analysis results using RETRAN code for a steam generator tube rupture at full power operation of a nuclear power plant. The boundary conditions for each case are given in Table 3. The Case 1 is total loop stagnated condition. The Case 2, 3, and 4 are partial loop stagnated conditions. The boundary condition input data of case 4 were obtained at a time of 3,000 seconds when the loops were partially stagnated (KEPRI, 1999). The other cases were selected to compare the results of each case. The HPSI flow rates, cold leg flow temperatures, and HPSI flow temperatures were taken as the same in all cases. The

Table 3 Input boundary conditions

Items		Loop A	Loop B
Cold leg flow rate, Kg/sec	Total loop stagnation (Case 1)	0.00	0.00
	Partial loop stagnation (Case 2)	99.80	0.00
	Partial loop stagnation (Case 3)	149.70	0.00
	Partial loop stagnation (Case 4)	199.60	0.00
HPSI flow rate, Kg/sec		20.16	20.16
Cold leg flow temperature, ℃		224.44	224.44
HPSI flow temperature, ℃		26.70	26.70

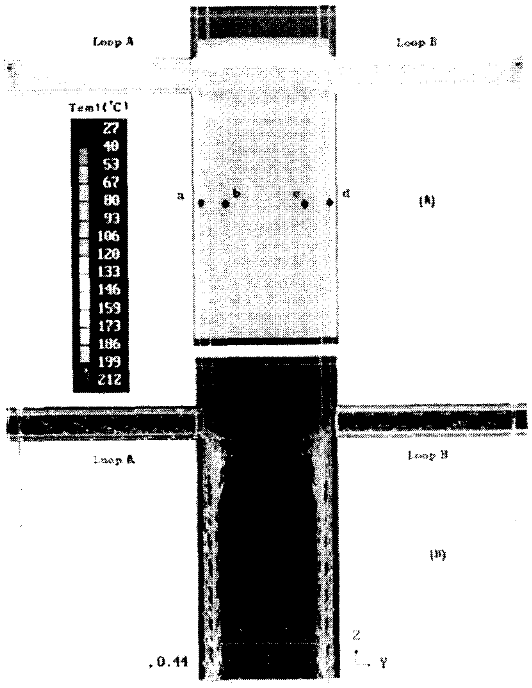

Fig. 6 Temperature (A) and vector (B) distribution of total loop stagnation (Case 1)
* a, b, c, and d : thermal shield-reactor vessel wall

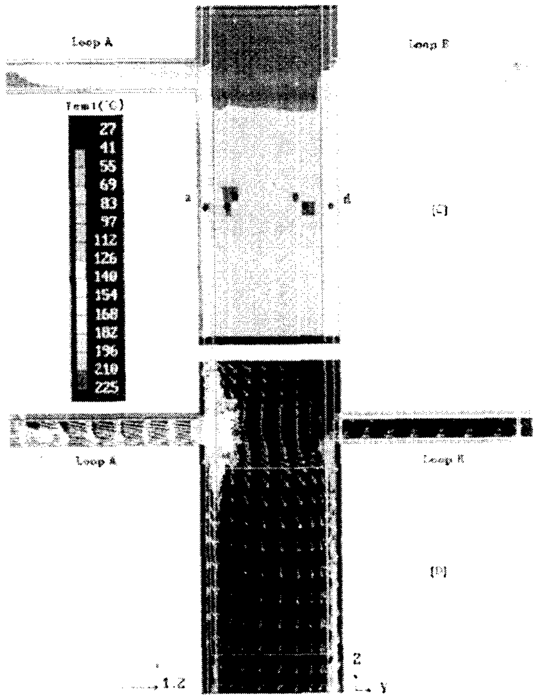

Fig. 7 Temperature (C) and vector (D) distribution of partial loop stagnation (Case 4)
* a, b, c, and d : thermal shield-reactor vessel wall

inner wall of the core barrel and the outer wall of the reactor vessel were assumed adiabatic condition.

3. Results and Discussion

During an accident in a nuclear power plant, the reactor coolant loops can be stagnated partially or totally because natural circulation may be destroyed by the loss of secondary heat removal source or the generation of steam in the reactor coolant system. In case of partial loop stagnation, previous studies have assumed that the cold plume moving into the reactor downcomer of the stagnated loop side, which is generated by thermal stratification, is neutralized by the strong flow from the unstagnated loop (Selby, 1985a and 1985b). In order to identify those phenomena, several cases were analyzed using the developed thermal-mixing analysis model. For the total loop stagnated case, Fig. 6 shows temperature and vector distribution at 600 seconds on the vertical planes of the cold legs and vessel downcomer region adjacent to the vessel wall. As an example for the partial loop stagnation (Case 4), Fig. 7 shows the temperature and vector distribution at 600 seconds on vertical planes of the cold legs and vessel downcomer when the flow rate of loop A is 199.6 Kg/sec. As shown in Fig. 7, it can be identified that the thermal stratification phenomena and the cold plume of loop B are affected by the strong flow of loop A. Fig. 8 shows the thermal stratification phenomenon generated in the stagnated cold leg piping adjacent to reactor vessel wall. The thermal stratification phenomena were generated in all cold legs due to the flow stagnation of all loops as shown in Fig. 6. On the other hand, the thermal stratification phenomena were just generated in the cold leg of loop B due to the partial loop stagnation as shown in Fig. 7. Those asymmetric phenomena were identified in all cases of partial loop stagnation. The temperatures of the vessel downcomer for the partial loop stagnated cases were estimated higher than those of the total loop stagnated case because of the continuous injection of hot fluid. As an analysis result of the total loop stagnated case, Fig. 9 shows the variations of the coolant temperature adjacent to the mid-weldment of the reactor vessel beltline region (See Figs. 2 and 6). As time progresses, the temperatures at locations 'a' and 'd' under the cold legs of the loop A and B sides were exponentially decreased to about 70℃. And, the temperatures at locations 'b' and 'c' were somewhat higher than those under the cold legs. The analysis results of the partial loop stagnated cases show the variations of the coolant temperatures at the same locations as the total loop stagnated case (See Figs. 10, 11, and 12). In the Case 2 where coolant flow of loop A was maintained at 99.8 Kg/sec, the coolant temperature at location 'd' under the cold leg of loop B was stabilized at about 150℃ after quickly cooling down about 50℃ (100 seconds). However, since the coolant temperatures at locations 'a', 'b', and

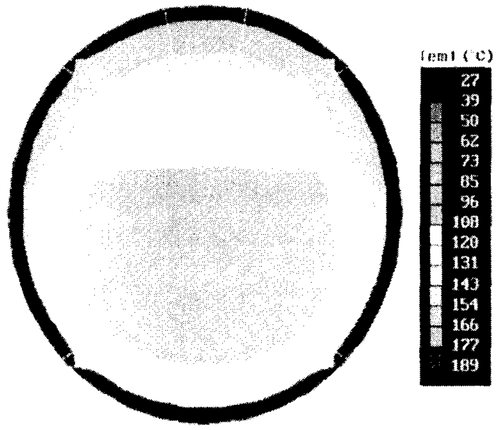

Fig. 8 Thermal stratification phenomenon of stagnated cold leg piping

'c' were stabilized at about 190℃, considerable mixing due to the strong flow of loop A was identified in almost all parts of the vessel downcomer except for location 'd' under the cold leg of loop B. The temperature difference between location 'd' and the others was maintained about 42℃. In the Case 3 where coolant flow of loop A was maintained at 149.7 Kg/sec, the coolant temperature behavior was similar to the Case 2. However, the coolant temperatures at all locations were somewhat higher than those of the Case 2 because of mixing by the stronger flow of unstagnated loop than that of Case 2. The temperature difference between location 'd' and the others was maintained about 45℃. In the Case 4 where coolant flow of loop A was maintained at 199.6 Kg/sec, the coolant temperature behavior was also similar to the Case 2 and Case 3. However, the coolant temperatures at all locations were somewhat higher than those of the Case 2 and Case 3 because of mixing by the stronger flow of unstagnated loop than those of Case 2 and Case 3. The temperature difference between location 'd' and the others was maintained about 48℃. According to these results for partial loop stagnated cases, the coolant temperatures at all locations of the vessel downcomer were increased by flow rate increase of unstagnated loop and the temperature difference between location 'd' and the others, was maintained about 42 to 48.

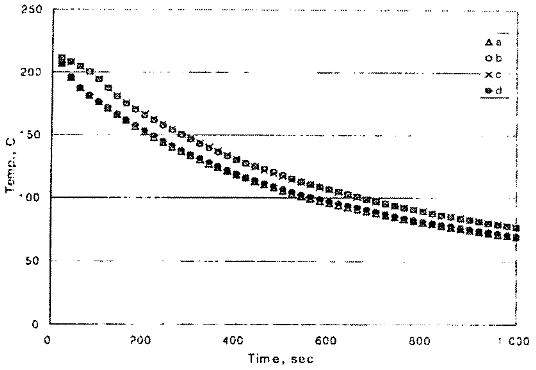

Fig. 9 Temp. comparisons of vessel downcomer for total loop stagnation (Case 1)
∗ a, b, c, and d : see Fig. 6

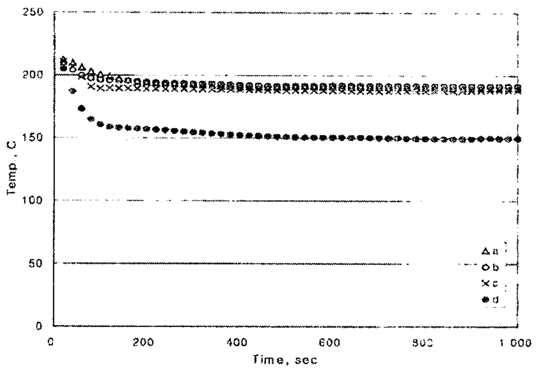

Fig. 10 Temp. comparisons of vessel downcomer for partial loop stagnation (Case 2)
∗ a, b, c, and d : see Fig. 6

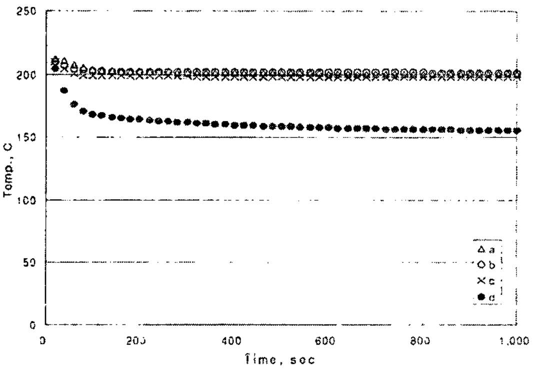

Fig. 11 Temp. comparisons of vessel downcomer for partial loop stagnation (Case 3)
∗ a, b, c, and d : see Fig. 7

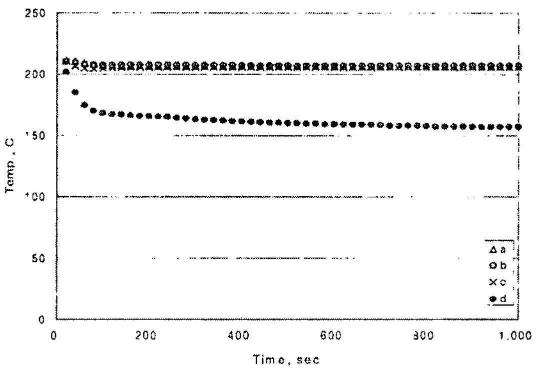

Fig. 12 Temp. comparisons of vessel downcomer for partial loop stagnation (Case 4)
∗ a, b, c, and d : see Fig. 7

Eventually, it is expected that, for partial loop stagnated cases, the cold plume generated in the vessel downcomer due to the thermal stratification phenomena of the stagnated loop is almost neutralized by the strong flow from the unstagnated loop but is not fully eliminated (See Figs. 7, 10, 11, and 12). If the thermal-hydraulic system analysis results rather than the thermal-mixing analysis results are reflected in PFM analysis, non-conservative failure probability of reactor vessel will be reflected in PTS evaluation. Based on these results, it is concluded that, for 2-loop reactor coolant system, the detailed thermal-mixing analysis of PTS evaluation may be performed for both partial and total loop stagnated cases.

4. Conclusions

RG 1.154 requires that a detailed thermal-mixing analysis in PTS evaluation should be performed for the thermal stratification phenomena generated in cold legs due to HPSI. In cases of partial loop stagnation, previous studies have used the results of thermal-hydraulic system analysis to calculate the failure probability of reactor vessel in PFM analysis. Because the cold plume generated in the stagnated loop of the partial loop stagnated cases are neutralized by the strong flow from the unstagnated loop. To determine the feasibility of the assumption, a thermal-mixing analysis model for a pressurized water reactor was developed using PHOENICS code. The analysis model was validated by comparison with the May 105 test results using the CREARE 1/2 scale test facility. And, the thermal-mixing analyses were performed for a case of the total loop stagnation and three cases of partial loop stagnation. The input flow rates of the unstagnated loop for the partial loop stagnated cases were taken to be 1.0, 1.5, and 2 times more than the flow rate determined in previous studies (Theofanous, 1984).

For the partial loop stagnated cases, the cold plume generated in the reactor vessel downcomer due to the thermal stratification phenomena of the stagnated loop is almost neutralized by the strong flow from the unstagnated loop but is not fully eliminated. Thus, it is concluded by this study that a detailed thermal-mixing analysis in PTS evaluation for a 2-loop reactor pressure vessel may be performed for not only total loop stagnated cases but also partial loop stagnated cases.

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