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A Feasibility Study on In-Vessel Core Debris Cooling through Lower Cavity Flooding

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ABSTRACT

Feasibility study has been accomplished to evaluate the effectiveness of the in-vessel core debris cooling through lower cavity flooding using two dimensional finite difference scheme. The volume of corium pool and decay power rate generated in corium pool were evaluated as important parameters to the temperature distribution on the reactor vessel lower head through previous works. In this study, the corium volume based on the System 80+ core structure and time dependent decay power rate are considered for feasibility evaluation. In addition, preliminary plans for the in-vessel core debris cooling through lower cavity flooding as severe accident management strategy, i.e. flooding timing, method and capacity, are suggested based on the result of the numerical study, international tendency related to in-vessel core debris cooling through lower cavity flooding.

1. Introduction

In-Vessel Core Debris Cooling through Lower Cavity Flooding (IDCCF) has been highlighted as prevention method for reactor vessel during the severe accident due to the termination possibility of the accident progression [1-8]. If the IDCCF is possible to retain the corium in the reactor vessel, complex ex-vessel phenomena, i.e. corium-coolant interaction, direct containment heating, corium-concrete interaction, etc., can be prevented, therefore, is considered as one of accident management strategies in some countries. However, in the case of failure of the reactor vessel due to the insufficient cooling for in-vessel core debris, steam explosion induced by the corium-coolant interaction would threaten the containment integrity and provide direct release path of fission product to environment. According to the merits and concerns related to this cooling mechanism, considerable analytical works to calculate the temperature distribution of the reactor vessel lower head and experimental works to identify the heat transfer of corium pool generated in the lower head and the critical heat flux on the outer vessel of the lower head have been accomplished, however, feasibility of the IDCCF is not fully evaluated because of the phenomenological uncertainties and insufficient information.

This paper describes following three items. First, developed computer program is simply described concentrated on the developed program structure and heat transfer correlations which are judged to be useful for thermal analysis. Preliminary evaluation result (thermal analysis) of the feasibility of IDCCF follows. In author's previous work [9], the volume of corium pool and decay power rate generated in corium pool were evaluated as important parameters to the temperature distribution on the reactor vessel lower head. To evaluate the feasibility of the IDCCF, the realistic corium volume and decay power with time should be considered. In this viewpoint, the corium volume based on the System 80+ core structure and time dependent decay power rate were reflected in the thermal analysis. Lastly, preliminary plans for the IDCCF as severe accident management strategy, i.e. cavity flooding timing, method and capacity, are suggested based on the result of the numerical study, international tendency related to the IDCCF.

2. Description of Developed Computer Program

The computer program was developed to accomplish thermal analysis on the lower head of reactor vessel using two dimensional numerical scheme and useful heat transfer correlations [9]. The developed program was formulated based on the finite-difference scheme and written in C-language and composed by several modules; VESSEL_TEMP module to calculate the temperature distribution of the reactor vessel lower head using the results of the other modules, PROPERTY module to provide the properties of the materials, OPTION module to provide the initial conditions, lower head pressure, number of meshes and select heat transfer correlations, PROPST module to calculate the properties of water in the lower head based on the lower head pressure, WET_CRUST module to calculate the molten pool temperature in the lower head and the thickness of lower and upper crust, HEIGHT module to calculate the height and contact angle of the corium pool based on the volume of the corium pool, and HEAT_TRANS module to calculate the heat transfer based on the OPTION module.

To accomplish thermal analysis, identification and selection of heat transfer correlations to predict the phenomena result from the core melting is one of most important processes. Resulted phenomena are natural convection of corium pool, single-phase, nucleate and transition boiling heat transfer to the cavity water, CHF on the outer surface of reactor vessel, etc. To predict the natural convection of corium pool, Steinberner-Goldstein correlations for upper and side heat transfer and Mayinger correlation for lower heat transfer can be used as suitable correlations considering COPO test performed by Kymalainen et al [10]. Churchill & Churchill correlation [11] can be used to predict the single phase heat transfer from the reactor vessel lower head to the cavity water. In spite of many nucleate boiling correlations, it is very difficult to suggest the suitable correlation to predict the nucleate boiling heat transfer. Recently, Cheung et al. [12] suggested a correlation which can be applicable to the this condition. The correlation is based on experimental data using small-scale test vessel; therefore, some confirmation is needed to apply their correlation. Dhir correlation [13] was used to predict the heat transfer in developed program. To predict the critical heat flux, many small- and large-scale experiments have been accomplished. Some useful CHF correlations considering the inclination of heated surface, i.e. Guo & El-Genk, ULPU-2000, SBLB correlation, were developed from their experiments. However, the geometry of test vessel is considerably different from the typical conditions of nuclear power plant and some considerations are omitted. Some more studies should be performed to clarify the CHF phenomena because the CHF is directly related to the integrity of the reactor vessel. ULPU-2000 configuration II correlation [5] seems to be used as basic correlation.

Detail descriptions of developed computer program are described in author's previous work [9].

3. Thermal Analysis

In author's previous works [9], the corium volume and the decay power rate generated in corium pool were evaluated as important parameters to the temperature distribution on the reactor vessel lower head. In this viewpoint, the volume of corium pool was calculated considering System 80+ core structure. Using the calculated corium volume and suitable decay power rate with time, the thermal analysis have been performed. Details are described as follows.

Calculation of Volume of Corium Pool

Volume of corium pool is calculated based on the System 80+ core structure, which is consisted of fuel rod, burnable poison rod, 12 element full strength control element assembly, 4 element full strength assembly and part strength control assembly. As the geometries of each rods are somewhat complex configuration, simplified forms are assumed to calculate the mass of each rod. The upper structure was excluded in present calculation. Geometrical dimensions for each rods are shown in table 1. Core structure is composed of UO_2, Zircaloy, B_4C, Ag-In-Cd, Inconel and Al_2O_3, whose calculated masses are shown in table 2. Based on the calculation, the volume of corium pool is 16.54m³. For comparison with AP600, mass and volume of each elements for AP600 and present study are also provided in table 2.

Table 1. Geometrical Dimensions of Core Element

	Fuel Rod (56876)	BPR (2112)	12 Element FSC (576)	4 Element FSC (80)	Part Strength CEA (100)
Material	UO_2	Al_2O_3-B_4C	B4C/Felt Melt and Reduced Dia. B_4C	Ag-In-Cd	Inconel-625
Pellet Diameter	0.826 cm	0.780 cm	1.872/1.712 cm	ID 0.635 cm /OD 1.86 cm	1.872 cm
Pellet Length	0.9906 cm	1.27 cm	-	5.08 cm	5.08 cm
Pellet Density	10.38 g/cm^3	2.52 g/cm^3	1.84 g/cm^3	~ 10.2 g/cm^3	~ 8.4 g/cm^3
Clad Material	Zircaloy-4	Zircaloy-4	Inconel-625	Inconel-625	Inconel-625
Clad ID	0.843 cm	0.843 cm	1.895 cm	1.895 cm	1.895 cm
Clad OD	0.970 cm	0.970 cm	2.07 cm	2.07 cm	2.07 cm
Clad Thickness	0.0635 cm	0.0635 cm	0.089 cm	0.089 cm	0.089 cm
Diametral Gap	0.01788 cm	-	0.0229 cm	0.0305 cm	0.0305 cm
Active Length	381 cm	345.44 cm	375.92 cm	375.92 cm	378.46 cm

Table 2. Mass, Density and Volume of Each Element

	Present Calculation			AP 600		
	Density (kg/m^3)	Mass (kg)	Volume (m^3); a	Mass (kg)	Volume (m^3) ;b	b/a
UO_2	10,400	120,000	11.53	75,900	8.68	1.32
Zircaloy	6,300	24,817	3.94	21,300	3.48	1.13
B_4C	2,400	1,049	0.44	-	-	-
Ag-In-Cd	9,250	1,194	0.13	2,900	0.43	0.30
inconel	7,700	2,395	0.31	-	-	-
Al_2O_3	6,744	1,274	0.19	-	-	-
Total Volume			16.54	-	12.59	1.31

Decay Heat with Time

To realistically predict the temperature distribution, decay power with time should considered in thermal analysis due to the long time progression of severe accident. In this study, the correlation suggested by Wigner-Way, which was used in Reactor Safety Study, is included to perform the thermal analysis:

$$P_d(t)=0.0622P_0[t^{-0.2}-(t_0+t)^{-0.2}]$$

Here, P_0, t_0 and t are the thermal power and operation time before shutdown and time after reactor shutdown, respectively.

Thermal Analysis

Prior to the thermal analysis, analysis for base case was performed considering the parameters of table 3, suggested correlations and some assumptions; cavity flooded with 323 K coolant prior to the slumping of corium, constant coolant temperature of cavity water for analysis time, negligible effect of the thermal insulation, uniform behavior of upper crust, constant temperature of upper structure and maintenance of it's integrity, etc. The inner and outer radius for thermal analysis are 2.40 and 2.64m.

- Base Case

  According to the calculation of the base case, the height and contact angle of corium pool are 1.69m and 72.88°, respectively. Temperature distributions are higher and crusts are thinner than the previous work due to the higher heat generation rate of corium pool. Temperature distributions with radius at 10000 sec and with time at 2.4 m are shown in fig. 1 and 2, respectively. According to the analysis, inner wall (~1cm) up to contact angle seemed to be melted for long time after about 4000 sec. Creep analysis should be considered to evaluate the integrity of reactor vessel. In this case, the decay power with time is not considered.

- Effect of decay heat generation rate
  Corium volume based on the System 80+ and Wigner-Way correlation are used to realistically predict the temperature distribution of reactor vessel. To identify the effect of heat generation rate, thermal analyses for four cases are performed using the power level shown in table 4. Power levels with time for 1 day operation, 1 year operation and base case are shown in fig.3. According to the fig. 3, power levels for 1 day and 1 year operation are lower than that of base case after 400 and 1400 sec, respectively. The corium pool and inner wall temperature behaviors of each cases are shown in fig. 4 and 5, respectively. According to the fig. 4, pool temperature reaches peak temperature and decreases with time in the case of using time dependent decay power. Figure 6 shows the inner wall temperature behavior with time for 1 day operation. The peak wall temperature occurred at about 8000 sec and decrease with time, which means the cooling rate is larger than the heat generation rate. This shows that the IDCCF is somewhat effective to cool the outer wall of the reactor vessel and can retain the corium in the reactor vessel.

Table 3. Parameters Used in Base Case

Heat Generation (MW)	48.3 (1.27%)
Volume of Corium Pool (m3)	16.54
Emmissivity	0.5
Temperature of Upper Structure (K)	1600

Table 4. Decay Heat Level used in Calculation

	Decay Power Level
Case 1	1.27 % (1hr) = 48.3
Case 2	0.84 % (4hr) = 31.9
Case 3	Operation time = 1 day
Case 4	Operation time = 1 year

4. Preliminary Plans for Accident Management

Flooding Timing

According to the available literature, Finland and U.S suggested criteria related to the flooding timing. However, only the timing suggested by Finland is related to the IDCCF concept and that suggested by U.S. is related to cavity flooding for cooling the corium at the cavity. These criteria are as follows;

- Finland ; high core exit temperature (Quantitative value is not provided)
- U.S. BWR ; water level of reactor vessel < 2/3 core (Millstone 1)
- U.S. PWR ; no RCS injection (Millstone 3)

In severe accident management, one of most effective strategies is injection into RCS. Therefore, if this is possible, water should be injected into RCS. If not injected into RCS, water source should be maintained at storage tank for later use or, if evaluated to be effective, water can be injected into the reactor cavity. In the latter case, the IDCCF is possible. As alternatives for flooding timing identification, thermocouples for core exit temperature and reading the water level of reactor vessel can be used. However, there is a considerable uncertainties and no insurance of instrumentation survivability under severe accident conditions.

Flooding Method

International tendency related to method of cavity flooding for cooling the corium at the cavity floor is use of passive system. However, there is a need, in the case of the IDCCF, of more water than cavity flooding. Therefore, combination of passive and active system, i.e. fire hose, is suitable for flooding the cavity. The use of combination seems to increase the reliability of cavity flooding system.

Flooding Capacity
It is difficult to find the water level for reactor submergence. However, to enhance the effectiveness of the IDCCF, it is better that the cylindrical part of reactor vessel is flooded with cavity coolant. According to EPRI technical report [6], the overall decay heat could be removed if the cylindrical part of the reactor vessel is occurred. Moreover, the result from the UCLA [8] shows that the stress level for the submergence of lower head could be higher than that for the submergence of cylindrical part.

Other Considerations
- Risk-benefit analysis should be carefully performed considering the benefits of the mechanism and risk of steam explosion resulted from the vessel rupture.
- If use containment spray system, the heat transferred to upper structure of reactor vessel, the RCS boundary could be removed and condensing the steam prevent the containment over-pressurization. However, hydrogen ignition problem should be carefully considered.
- If the IDCCF is possible to cool the heat transferred from the in-vessel corium, containment bypass, such as the steam generator tube rupture should be considered as one of important concerns.
- If the IDCCF would be used as accident management strategy, further works related to procedure and guidance, instrumentation, decision-making responsibility, etc., should be performed.

5. Conclusions and Recommendations

Thermal analyses have been performed to evaluate of the IDCCF using developed computer program with suitable heat transfer correlations and realistic conditions. According to present evaluation result, the IDCCF could effective method to cool the melting core in the reactor vessel and terminates the accident progression. However, for long time, the vessel temperature near the inner most wall stays at high temperature, therefore, creep analysis should be performed to evaluate the effectiveness of IDCCF. Moreover, the plant-specific design should be included to assess the overall feasibility of the IDCCF.
Preliminary plans for the IDCCF as severe accident management strategy, i.e. flooding timing, method and capacity, have been suggested like as follows;

- Flooding Timing ; No RCS injection
- Flooding Method : Combination of passive and active method
- Flooding Capacity ; At least up to the cylindrical part of reactor vessel

Finally, to use the IDCCF as severe accident management strategy, further works related to procedure and guidance, instrumentation, decision-making responsibility, etc., should be performed.

Acknowledgment
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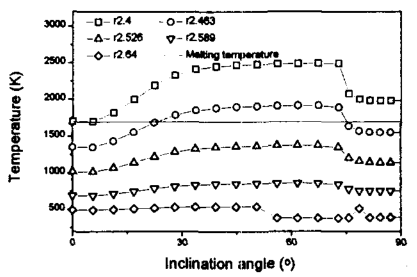

Fig. 1 Wall temperature distribution of the reactor vessel (power generation rate = 2.91MW/m³, time=10000sec)

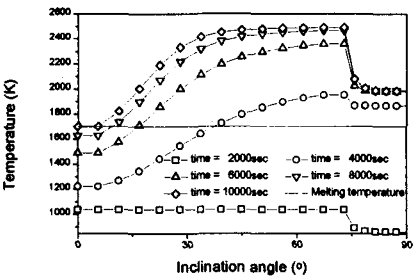

Fig. 2 Inner wall temperature behavior with time (power generation rate = 2.91MW/m³)

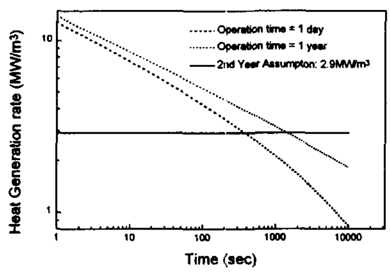

Fig. 3 Decay power levels

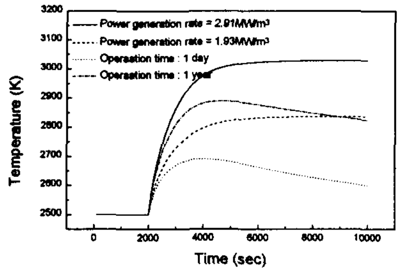

Fig. 4 Pool temperature behaviors

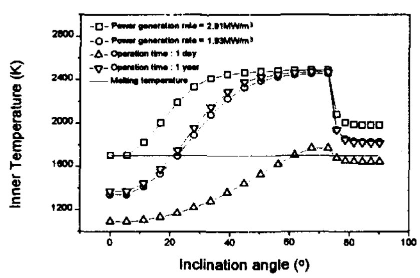

Fig. 5 Inner wall temperature behaviors (time=10000sec)

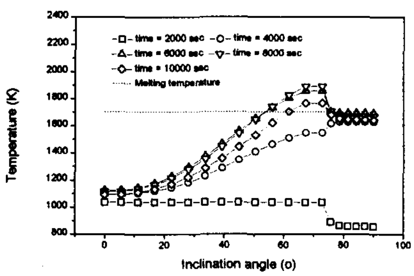

Fig. 6 Inner wall temperature behaviors (Operation time = 1 day)