Effectiveness of External Reactor Vessel Cooling (ERVC) Strategy for APR1400 and Issues of Phenomenological Uncertainties

S. J. Oh and H.T.Kim
Nuclear Environmental Technology Institute, Korea Hydro & Nuclear Power Co., Ltd (KHNP)
103-16, Munji-Dong, Yuseong-Gu, Taejon, 305-380, Korea
Tel: 82-42-865-7603, Fax: 82-42-865-7609,
E-mail: se_oh@khnp.co.kr, hyeong@khnp.co.kr

1. Introduction

The APR1400(Advanced Power Reactor 1400) is an evolutionary advanced light water reactor with rated thermal power of 4000MWt [1]. For APR1400, External Reactor Vessel Cooling (ERVC) is adopted as a primary severe accident management strategy for in-vessel retention (IVR) of corium. The ERVC is a method of IVR by submerging the reactor vessel exterior. At the early stage of the APR1400 design, only ex-vessel cooling, cooling of the core melt outside the vessel after vessel is breached, is considered based on the EPRI Utility Requirement Document for Evolutionary LWR [2]. However, based on the progress in implementation of Severe Accident Management Guidance(SAMG) for operating plants, as well as the research findings related to ERVC, ERVC strategy is adopted as a part of key severe accident management strategies. To improve its success, the strategy is reviewed and we implemented necessary design arrangement to increase its usefulness in managing the severe accident.

In this paper, we examine the evolution of ERVC concept and its implementation in APR1400. Then, we review possible approach, including Risk-Oriented Accident Analysis Methodology (ROAAM), to evaluate the effectiveness of the strategy.

2. ERVC Strategy

The idea of submerging the reactor vessel during the severe accident has been considered since mid 80s. With the vessel submerged before the corium reaches the lower plenum, the decay heat from the relocated corium will be removed by the submerged water and the vessel will maintain its integrity. Experts developing SAMG Technical Basis Report [3] have considered the strategy useful and included as one of High Level Candidate Strategy. This, in turn, has been adopted as a severe accident management strategy for operating plants. In Finland, Lovisa plant has modified the design so that the ERVC strategy is highly effective in mitigating severe accident.

For ALWRs, Westinghouse incorporated the concept into the AP600 design so that its success will be maximized. The In-Vessel Retention by ERVC has been adopted for AP1000. For AP1000, streamlined insulation is adopted to increase the effectiveness. As will be discussed in the next section, Theofanous used the ROAAM process to quantify the vessel integrity for AP600 [4]. His goal is to categorize the failure of vessel for AP600 under IVR is physically unreasonable. Currently, this strategy has been adopted in one form or another in several ALWR designs.

For APR1400, ERVC is chosen as one of important severe accident management strategies. Conceptual drawing of the strategy applied to APR1400 is shown in Figure 1. Under the severe accident condition, after the initiation of RCS depressurization, one of shutdown cooling pumps(SCP) is used to inject water into the reactor cavity. The time to fill up to the lower head with one SCP is 30 minutes. The water will fill up the cavity to cold leg elevation. Then the injection mode will be transferred to boric acid makeup pump(BAMP) to maintain the level. Once the necessary actions are completed, the recovery of safety injection system and injection into the vessel is recommended to arrest the progression of severe accident.

The effectiveness of the ERVC strategy needs to be examined to determine its usefulness and to reflect in the level 2 PSA quantification. We followed the structured approach used by Theofanous et al. However, unlike AP600 study, we do not try to prove the vessel failure as physically unreasonable. Issues raised on the bounding melt configuration during the AP600 design certification review illustrate the difficulties to reach an agreement among experts at the level of 'physically unreasonable'. However, the structured evaluation and the reviewers' comment on the evaluation will be useful in evaluating the effectiveness and the necessary reference material for the sound expert elicitation process in PSA quantification.

3. ROAAM Process

Severe accident is a low likelihood, high consequence end of the events and there exists an inherent difficulty in quantification due to the phenomenological uncertainties. While performing NUREG-1150 study, expert elicitation process has been used extensively to overcome this difficulty [5]. Theofanous proposed to shift the paradigm from quantifying 'uncertainty' to resolving 'uncertainty' such as to meet safety goals. He proposed ROAAM process as a way to deal with the phenomenological uncertainties.

Theofanous et al., applied the ROAAM to AP600 in-vessel retention study. First, he identifies a representative scenario that would bound the severe accident progression in relation to the vessel integrity. For AP600, he chose the full core melt extension of the large break LOCA scenario from the level 1 PSA. For this bounding scenario, he investigated the pertinent phenomenological uncertainties. Under the ROAAM, the issue of vessel integrity has been divided into a set of sublevel questions as shown in Figure 2. The sublevel questions are the thermal load and thermal failure criterion, and structural failure criterion. The study showed that the thermal failure criterion is the limiting one [4]. The thermal failure question is whether the heat removal from the outer vessel wall is sufficient enough maintain the steady state and the wall integrity. Then, the issue becomes to define the thermal load inside the vessel for the bounding scenario and to determine whether it is below the heat removal limit, CHF at outer wall. The thermal load is calculated using two layer model (metallic layer above the heavier ceramic layer). A correlation based on ACOPO tests have been used to determine the thermal load. With the two layer model, the peak heat flux occurs at the metallic layer near the 90 degree location. To determine CHF limit under the postulated conditions, series of experiments, ULPU, have been conducted. For AP600 insulation geometry, ULPU-III test data are considered applicable. With the known CHF limit and thermal load, he performed sensitivity study to show the margin. A rigorous evaluation was performed and documented to claim that it met the criteria of 'physically unreasonable'.

The concept of 'physically unreasonable' is proposed by Theofanous as a part of ROAAM process. The probability level proposed can be summarized as followings [6]:

P= 0.1 : Behavior is within known trends but obtainable only at the edge-of-spectrum parameters

P=0.01 : Behavior cannot be positively excluded, but it is outside the spectrum of reason

P=0.001: Behavior is physically unreasonable and violates well-known reality. Its occurrence can be argued against positively

However, the quantitative measure corresponding to the definition seems to be arbitrary to these authors.

Once the calculation is done, he went through expert reviews similar to expert elicitation process. The issue not reaching an agreement is the possibility of intermediate core melt configuration. The melt configuration affects the metallic layer focusing effect and, in turn, affects the success probability of vessel integrity. The issue of chemical reaction and the adequacy of two layer model has been repeatedly raised by reviewers and hasn't reached a consensus even now. It would be ideal if one can gain the consensus from all the reviewers. However, the issue is when there is a disagreement. In our view, it is partly due to the term used ‘physically unreasonable’, which wouldn’t allow any doubts or uncertainties. It is difficult to connect with the probability of occurrence of 0.001.

4. Effectiveness of IVR strategy for APR1400

As stated, we would like to examine the effectiveness of the ERVC strategy and to determine its usefulness. Furthermore, we would like to utilize the result in the level 2 PSA quantification. The starting point is the level 1 PSA result. Representative scenarios are chosen from Level 1 PSA result. Then, using MAAP4 code, we examine the boundary condition for ERVC strategy. With the boundary condition fixed, the ROAAM framework utilized by Theofanous is adopted to examine the effectiveness.

- Representative Scenarios

Based on the core damage groupings of level 1 PSA, transients are grouped into one. For the transients, we chose LOFW-17 as the representative scenario. It has the fastest melt progression among the group. Time from shutdown is important because he decay heat from corium determines the thermal load. For the small LOCA group, SLOCA23 is chosen as the representative scenario. For the medium LOCA, MLOCA-4 is chosen. For the large LOCA, LLOCA-4 scenario is chosen. The top-level description for these four scenarios is summarized in Table 1. MAAP4 is used for evaluating accident progression. The standard RCS noding as shown in Figure 3 is used in MAAP-4 model. The core is modeled with 7 radial and 13 axial nodes. The containment model consists of 12 compartments and 39 flow paths as shown Figure 4. Detailed description is available in Ref. [7]. Table 1 summarizes the timing of the core melt progression.

Table 1. MHFR on the bottom of metallic layer for major accident scenarios

category	Percentage of Total CDF (%)	Steel mass molten (M_{steel}), (tons)	Zirconium oxidation fraction (f_{ox})	Core melt fraction (f_{U02})	Time to Full Core Melt (hr)	MHFR
LOFW	35.2	32	0.38	0.85	10.14	0.50
SLOCA	26.7	28.4	0.42	0.78	9.5	0.51
MLOCA	9.6	32.7	0.44	0.88	5.6	0.62
LLOCA	2.3	25.2	0.34	0.82	3.72	0.74

- CHF Correlation

Regarding the CHF during the ERVC, two series of tests are available. First, is the ULPU series of tests conducted by Theofanous et. al., at UCSB. The other set of tests is SBLB series by Fan B. Cheung et. al., at Penn State University. ULPU tests were conducted with a sliced geometry with the full height representation [4]. SBLB tests were conducted with a hemispherical three-dimensional representation with 1/5 scale[8]. These two tests are complementary to each other: one showing the three dimensional effect, the other showing the effect of gravity head and the nature of flow circulation. Key findings from ULPU series of tests are: (1) ULPU-V tests show that the CHF limit at 90 degree location reaches above 2 MW/m2 for streamlined insulation geometry representative of AP1000, and (2) CHF under natural circulation seems to be higher than those under forced convection. The test geometry for ULPU series is based on AP600 and AP1000 design. However, since it is a sliced representation, it is applicable to APR1400 except the effect of keyway region.

For APR1400 geometry, Cheung et al performed sets of tests representing APR1400 geometry. Specifically, there are two sets of tests. First series of test examines the insulation geometry typical of currently operating plants, OPR1000(Optimized Power Reactor 1000). They investigated two phenomena particularly: the effect of flow bottleneck due to the insulation geometry at the midpoint and the CHF at the 90 degree location. The flow bottleneck did reduce the CHF to about 1 MW/m^2. The CHF at 90 degree location has not been reached with the maximum heat input which was 1.5MW/m^2. The second series of test examines the effect of streamlined geometry and coating. For the streamlined geometry without coating, CHF limit of 2 MW/m^2 has been obtained [9]. CHF measurements from these tests are shown aggregately in Figure 5 as a part of the effectiveness evaluation of APR1400 IVR

- Effectiveness of IVR strategy for APR1400

MELTTHERM program has been developed for this investigation. The program calculates the wall heat fluxes based on two-layer configuration assumption (metallic layer on top of oxidic layer). The program is based on the formulation developed for thermal load calculation by Theofanous et al in AP600 study [4]. ACOPO correlation is used for wall heat fluxes. Metal mass as well as the total corium poured into the lower plenum is user inputs. Detailed discussion on MELTTHERM can be found in Ref [10].

MAAP4 code calculation result at the time of vessel breach is used as the inputs for MELTTHERM corium condition. The output from MELTTHERM calculation is shown in Figure 5. As shown in the figure, LOFW and SLOCA has a sufficient margin. LLOCA has the lowest margin. Hence, the sensitivity study on LLOCA scenario is performed. For LLOCA scenario, MAAP4 condition at the time of vessel breach (3.72hr from shutdown) is 82% of core has melted and 25 ton of steel mass is in the lower plenum. First, we extrapolated the melt progression and came up with the full core melt in 4.22 hr. Based on APR1400 lower structure geometry, we estimated that 40 ton of steel mass would participate in melting for full core melt. Then, we defined a limiting condition as full core melt at 3.72 hr with steel mass of 30 tons. Both cases are shown in figure 6. Considering the uncertainties, some experts would like to see more margin for the limiting case with OPR1000-like insulation geometry. However, the evaluation shows that the IVR strategy is quite effective severe accident management strategy for APR1400.

- The issue of melt configuration

Most of the issues raised during AP600 IVR study seem to be resolved except the issue of corium configuration in-vessel: namely, the forming of heavier metallic layer at the bottom. The issue has been raised during the review as a possible intermediate state. This has been further complicated by the result produced by RASPLAV and MASCA. Considering the limitation and the amount of available information, this requires further investigation. Recently, Seiler et. al., have examined this issue considering physico-chemistry effect [11]. However, this issue requires further investigation and study to reach a resolution. At this time, an acceptable approach in quantification seems to be using the expert elicitation process. In formulating APR1400 SAM strategies, we have included late in-vessel injection strategy to minimize the impact of such uncertainty.

5. Summary and Conclusion

APR1400 has adopted ERVC as one of key severe accident management strategies. The structured evaluation showed the effectiveness of the strategy. However, it is difficult to reach an agreement for issue closure and one has to utilize the expert elicitation process for quantification. In this regard, the structured approach is useful to provide an explicit, formal and structured information to experts.

The ROAAM process would be ideal if one can gain the consensus from all the reviewers. Now, the issue is when one can not reach consensus, how one goes about the quantification of the failure probability. Within the ROAAM framework, the bridge to the quantification is provided. However, the probability assigned is somewhat arbitrary and there should be a bridge between the two to have a consistent scale of measure. Furthermore, it has been our experience that the reviewers, when presented with ROAAM-like analysis result, were biased toward the failure and very conservative in assigning success probability. They tend to ignore the low probability of reaching the so called ‘bounding’ scenario which was chosen to account for the various limiting (so called ‘bounding’) assumptions incorporated originally to avoid the scenario dependency.

It is my experience that there should be a bridge between the two to have a consistent scale of measure to handle the large uncertainty related to severe accident and level 2 phenomenological uncertainty systematically.

REFERENCES

1. Korea Hydro & Nuclear Power Co., APR1400 Standard Safety Analysis Report, 2001.
2. Electric Power Research Institute, Advanced Light Water Reactor (ALWR) Utility Requirements Document, Rev.7, 1995
3. Electric Power Research Institute, Severe Accident Management Guidance Technical Basis Report, EPRI TR-101869, 1992
4. Theofanous T. et. al., “In-Vessel Coolability and Retention of a Core Melt”, DOE/ID-10460, 1995
5. U.S. Nuclear Regulatory Commission, Severe Accident Risks : An Assessment for Five U.S. Nuclear Power Plants, NUREG-1150, 1989
6. Theofanous T. and Yang H. , “The Probability of Liner Failure in a Mark-1 containment, Part 1: Probabilistic Framework and Results”, Nuclear Technology, Vol. 101, March 1993 pp299-331
7. S. J.Oh et al., “In-Vessel Retention Technology Development and Use for Advanced PWR Design in the USA and KOEA”, K-NERI Final Report, 2004
8. F. Cheung and Y.Liu, ‘Critical Heat Flux Experiments to Support In-Vessel Retention Feasibility Study for an Evolutionary Advanced Light Water Reactor Design’, EPRI TR-1003101, November 2001.
9. J. Yang and F. Cheung, “Critical Heat Flux for Downward Facing Boiling on a Coated Hemispherical Vessel Surrounded by an Insulation Structure,” Proceedings of ICAPP 2005, Seoul, Korea
10. J. W. Park et al., In-Vessel Core Debris Coolability of the APR1400, Korea Hydro & Nuclear Power Co. 2001.
11. J. M. Siler et al., ”Analysis of In-Vessel Retention with Special Interest in Physico-Chemistry Effect”, Proceedings of ICAPP 2005, Seoul, Korea

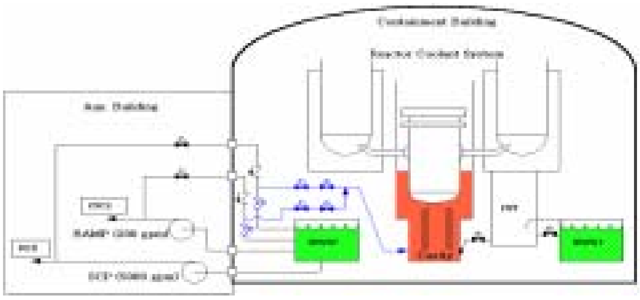

Fig. 1 Basic Design of ERVC for IVR and Cavity Flooding System (CFS) of the APR1400

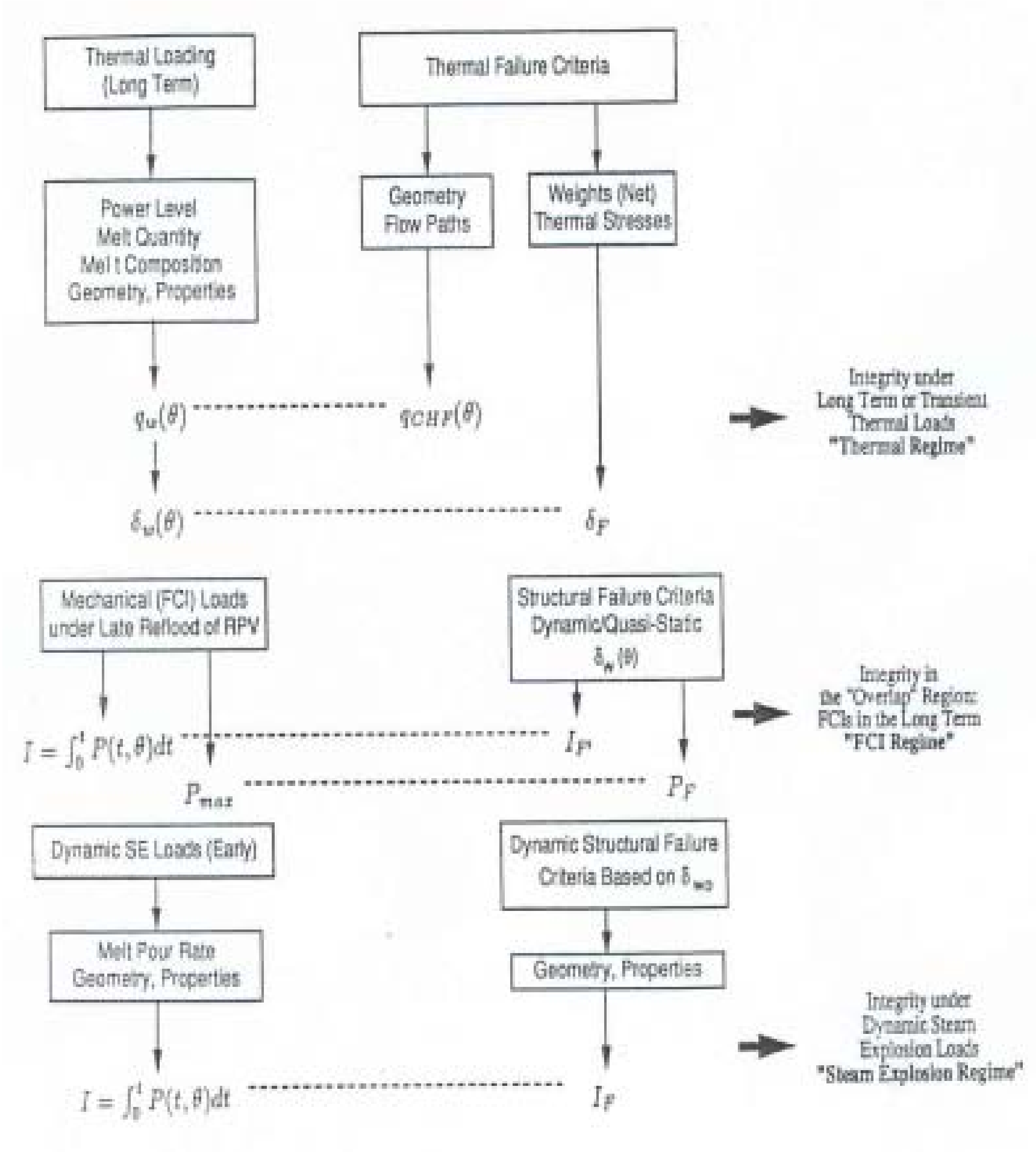

Fig. 2 An overall view of the in-vessel retention issues

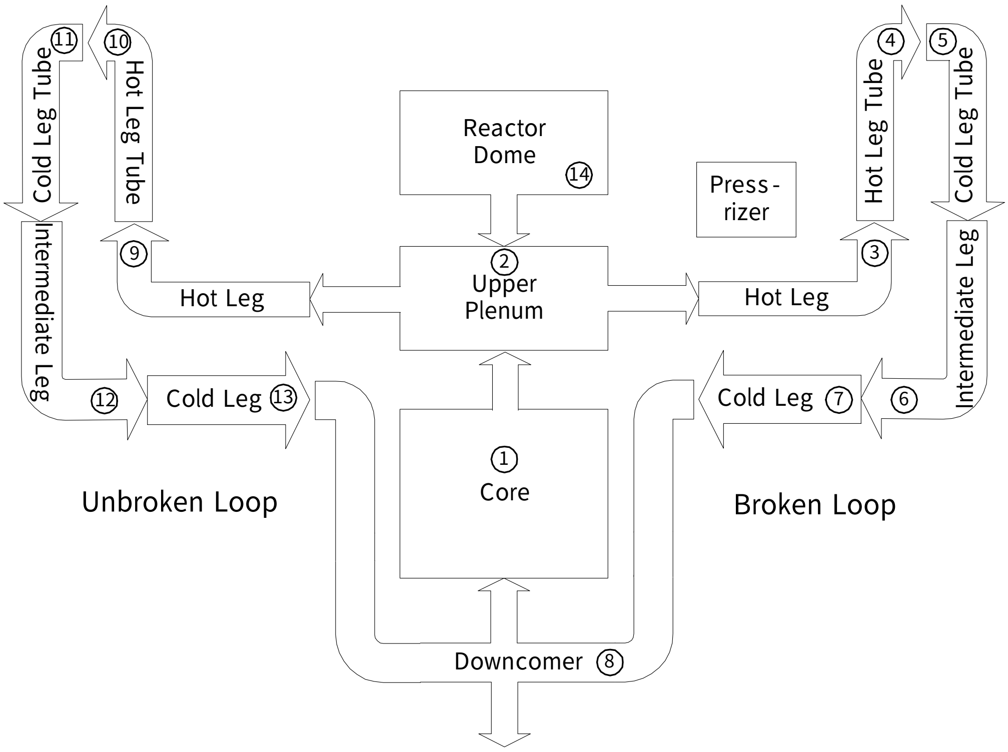

Fig. 3   MAAP4 Code Analysis Model for RCS in APR1400

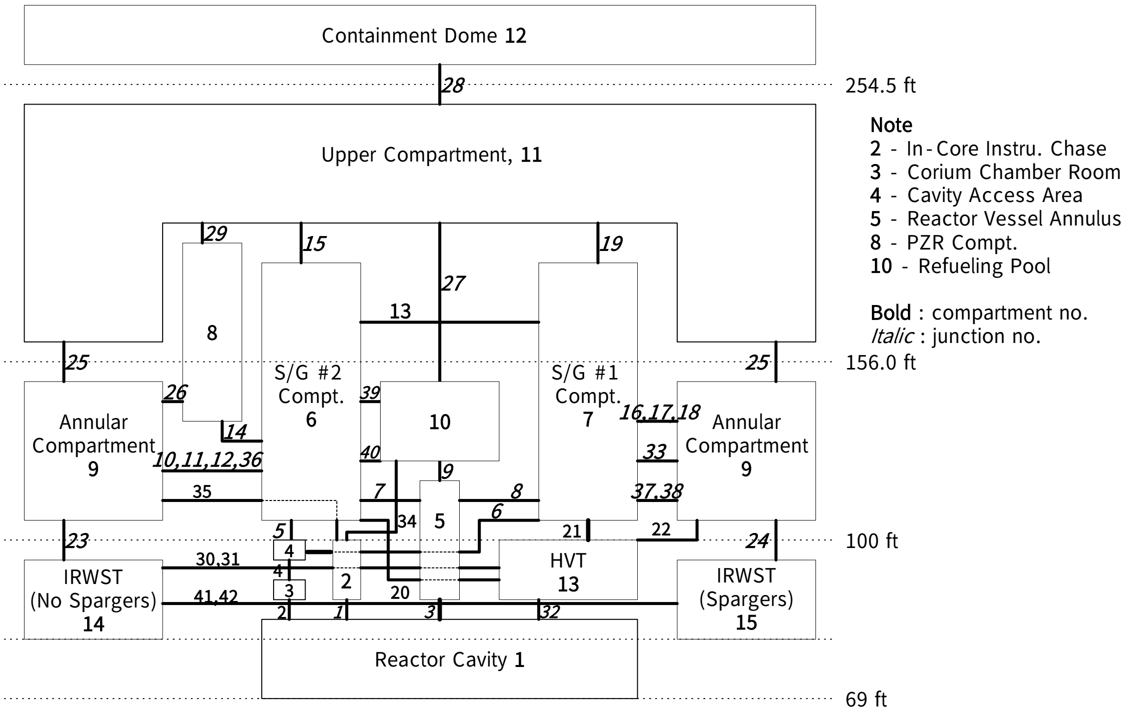

Fig. 4   MAAP4 Code Analysis Model for Containment in APR1400

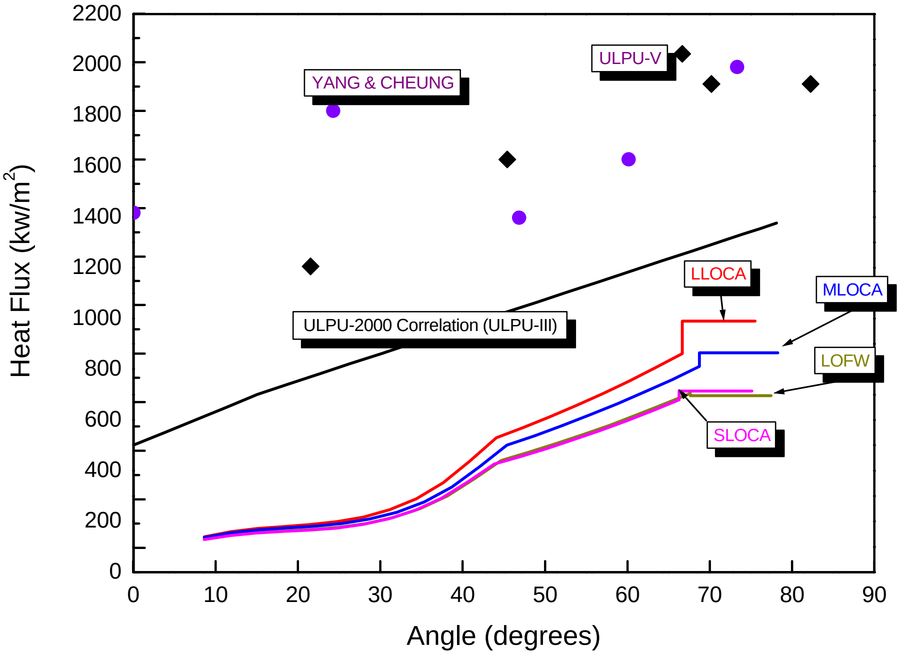

Fig. 5 Sensitivity Study of Heat Flux depending on the Angle of RPV wall for major accident scenarios

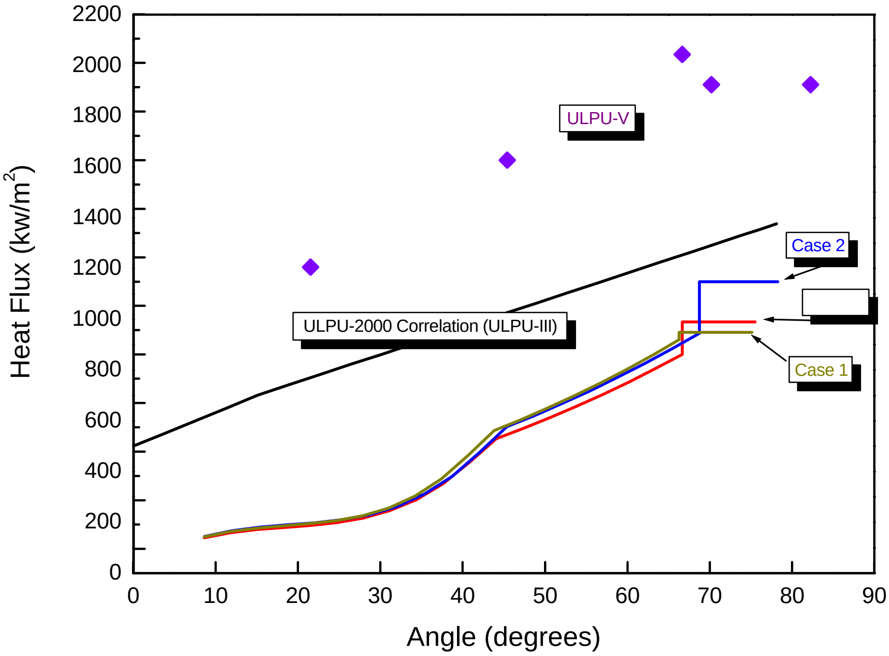

Fig. 6 Sensitivity Study of Heat Flux depending on the Angle of RPV wall for LLOCA