Appendix-D

Severe Accident Analysis Report for FCI

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1.0 INTRODUCTION

1.1. Background and Objectives

Steam explosion is a phenomenon that rapid heat transfer between a hot and volatile cold liquid occurs and results in explosive dynamic pressure that may threaten the integrity of surrounding structures. This phenomenon has been a special interest as risk-significant events in nuclear power plants during severe accidents. In the accidents, reactor fuel and materials melts due to the lack of adequate cooling capability as observed in Fukushima accident and forms a molten material, called corium. In the actions of mitigating and eventually terminating the severe accident progression, cold water supplies to cool the corium by any means. In this process, molten corium contacts water, resulting steam explosions that may provide mechanical and functional damages to the surrounding structures and safety systems. Therefore, it is of importance to evaluate the energetics of steam explosions in the process of the safety assessment of nuclear power plants. In nuclear power plants, steam explosions can be classified into two in terms of the accident progression; in-vessel and ex-vessel. In-Vessel Steam Explosions (IVSE) occurs in the Reactor Pressurized Vessel (RPV) and Ex-Vessel Steam Explosions (EVSE) in the outside of RPV, that is, reactor cavity for APR1400.

In the 1970s, the steam explosion process was intensively studied for the fast breeder reactor, in which uranium oxide and sodium were fuel and coolant, respectively. While search was continued on steam explosions for the safety of the fast breeder reactor, a comprehensive risk assessment to estimate the likelihood of containment failure in Light Water Reactors (LWR) due to a steam explosion was conducted in 1975 and reported in WASH-1400 [Reference 1]. This study focused on two specific reactor designs; the Surry PWR and the Peach Bottom BWR-Mark I. For the steam explosion process, it was determined that the containment could be threatened by three possible damage mechanisms; (a) dynamic liquid phase pressures on structures, (b) static over-pressurization of the containment by steam production, and (c) solid missile generation from the impact of a liquid slug accelerated by the steam explosion.

In this analysis, the primary concern was a direct failure of containment caused by an energetic Fuel Coolant Interaction (FCI) causing missile generation (designated "α-mode" failure) was estimated to be about 10^{-2} per reactor year with the likelihood of water availability and triggering of the explosion as the major uncertainties. A group of steam explosion experts performed a series of periodic review on the potential risk of steam explosion especially by the α-mode failure in every 5, 10 years; SERG-I [Reference 2], SERG-II [Reference 3], and the OECD/NEA FCI Specialist Meeting [Reference 4]. The outcomes of the review were focused on mostly for the risk of the α-mode failure. The expert group recognized that α-mode failure required (1) large amount of corium relocation to the lower plenum, (2) large amount of corium-water mixing and (3) large energy conversions during steam explosions. However, those requirements are very difficult to be satisfied in the in-vessel scenarios because of (1) large vapor generation during the corium-water mixing due to the near saturation of water temperature and (2) less corium relocation due to internal structures and complex passages. Therefore, the estimated conditional probability of the α-mode failure had reached a consensus of resolution in a risk perspective, in the series of expert reviews as follow (See Table 1-3);

SERG-I: < 10^{-4}~10^{-2}, vanishingly small

SERG-II: < 10^{-5}~10^{-3}, physically unreasonable

Until that time, the experiments showed a distinctive result that prototypic corium melts were difficult to be triggered to initiate an energetic explosion and generated low energy conversion while single oxidic melt like Alumina provided a very energetic explosion. These observations triggered a debate of “melt material effects” on steam explosion energetics. The scaling of material characteristics had been an issue; sharp increase of viscosity during freezing, non-uniform freezing (surface freezing), non-condensable gas generation like hydrogen etc. However, limited theoretical understanding hinders to figure out the scaling involved. For the corium tests, limited numbers of experimental database are available and most of date there are large uncertainties in corium temperature (superheat), composition and properties. Although the difficulties in performing experiments, uncertainties in results and different definitions of explosion efficiency evaluations exist, there is a tendency that corium tests consistently provide lower energetics (conversion efficiency) compared with single oxide and metal tests such as Alumina tests in the KROTOS facility. Corium tests provide their explosion efficiencies low and zero in many cases, meaning no steam explosion. However, most of simulant tests show high explosion efficiencies. In particular, the alumina tests performed in KROTOS facility provided very strong energetics and have been considered the upper bounds of the experimental results. For the reason, many computational modeling for steam explosion have been validated with the alumina tests and consequently most of codes tend to estimate their explosion energetics more conservatively.

The SERENA-I project [2002-2005] [Reference 5] aimed to make a status of the steam explosion codes capabilities to calculate steam explosion in reactor situations and to find out what would be strictly necessary to undertake in order to bring understanding and predictability of steam explosion energetics to desirable levels for risk management. The conclusions drawn from the projects despite the differences in modeling and approaches and the large scattering of hypothesis and results for explosion processes that the reactor calculations indicated clear tendencies; for the in-vessel case the calculated loads were far below the capacity of the defined model intact vessel and for the ex-vessel case, the calculated loads, even low, are above the capacity of the defined model cavity walls. The scattering of the results raises the question of quantifying the containment safety margin for ex-vessel steam explosions. Two main uncertainties were accentuated; uncertainties in the pre-mixing flow patterns and geometry due to the lack of detailed experimental data and uncertainties on material influence on the energetics.

The SERENA-II project [2007-present] [Reference 6] has two objectives to reduce the uncertainties identified in the SERENA-I project for computer code validation; (1) to obtain more detailed data on pre-mixing flow patterns in particular distribution and geometry and (2) to determine explosion behavior of a spectrum of corium melt compositions reflecting accident scenarios by performing two experiments, TROI (KAERI, Korea) and KROTOS (CEA, France). For the purpose one, the KROTOS facility equipped with a level meter for average void fraction and a high-energy X-ray radiography to visualize the pre-mixing of steam explosions and the TROI facility equipped with a differential pressure meter for an average void fraction and an electro-capacitance tomography. At present, reliable experimental data for the quantification of void fraction in the pre-mixing phase of steam explosion were not obtained yet. For the analytical activities, the similar activities still continues but tries to model the material effects, such as solidification modeling. For instance, one of mechanistic codes developed in the framework of the SERENA-II, TEXAS-V implements the solidification model to update their capability of simulating the corium tests performed in the project

[Reference 7]. In summary, the development of adequate models to describe the steam explosion is challenging but made much progress in understanding of the steam explosion in reactor application to make a benign conclusion.

In summary, based on experiments and related analyses, the α-mode failure due to an in-vessel steam explosion is not considered a credible threat to containment integrity. However, in-vessel steam explosion is still a potential risk of reactor vessel failure, affecting to the accident management including in-vessel melt- retention strategy. In general, the vessel failure criteria are about a few hundreds kPa-s of impulse and depend on the specific reactors. Therefore, most of the in-vessel steam explosion analyses are conformed with the RPV structure analysis for their risk of vessel failure. Since the most of RPVs fulfills the criteria, the risk of in-vessel steam explosions was not in an issue of resolution but confirmation for old and new power plants. EVSE, on the other hand, has been one of the long pending unresolved issues in LWRs since in some existing LWRs water discharged from the reactor primary system accumulates in the reactor cavity under the vessel.

Therefore, in this report describes a comprehensive analysis of the hazard potentials involved in in-vessel and ex-vessel steam explosions in the APR1400 design. For steam explosion including IVSE and EVSE, its energetics and resulting loadings to RPV in the case of IVSE and cavity structure in the case of EVSE are evaluated. The analysis methodology employs the base case analysis with a mechanistic code for steam explosion, TEXAS-V.

1.2. Reactor Case Analysis of Steam Explosion Phenomena

1.2.1. Assessment of IVSE for Existing Nuclear Power Plants

Corradini et al. [Reference 49] extended the analysis of WASH-1400 with models for fuel-coolant mixing and explosion, i.e., expansion work into a Monte Carlo analysis to estimate the α-mode failure probability; namely given a core melt, the probability of α-mode failure was estimated less than 10^{-4} per reactor year. In 1985, in NUREG-1116 [Reference 2], the Steam Explosion Review Group (SERG) estimated the probability of the α-mode failure. This group of experts performed independent analyses and estimated that the conditional probability of the α-mode failure was much less likely than in WASH-1400 (10^{-2}~10^{-4}/yr as upper bound given a core melt). This group also recognized that these estimates were founded on the judgment that the amount of fuel-coolant was limited and/or the explosion yield was less than the maximum thermodynamic values. Included in their findings was the consensus recommendation that fundamental experiments needed to be performed at large scales to characterize fuel-coolant mixing and measure explosion yield as well as the effect of mixing on yield.

Theofanous et al. [Reference 8] performed a comprehensive analysis of α-mode failure and found the upper bound value to be quite low ($<10^{-4}$ per reactor year). Recently, Turland et al. [Reference 9] investigated a methodology for quantifying the conditional probability of the α-mode failure in the Sizewell B PWR. They estimated that the probability was approximately 10^{-4} and the effect of the system pressure elevation on the probability of this mode of failure was modest. Theofanous et al. [Reference 10] updated their original risk assessment [Reference 11] and concluded that even vessel failure by steam explosions might be regarded as physically unreasonable. Table 1-1 summaries the relevant studies on the effort of the IVSE issue resolution.

1.2.2. Assessment of EVSE for Existing Nuclear Power Plants

Theofanous et al., [Reference 12] performed a comprehensive approach to examine the steam explosion energetics in detail for the requirement of the design of nuclear systems by providing the dynamic aspects of explosions and the resulting dynamic loads on adjacent structures. The results were provided on two-dimensional explosions from their micro-interaction model in the ESPROSE.m code.

Chu et al. [Reference 13] performed the analysis for the Swedish BWRs to examine their SAM strategy of the lower drywell flooding to mitigate against the effects of melt release into the drywell during severe accident. With the use of the THERMAL-1 code, they investigated the effectiveness of the water pool to protect lower drywell penetrations by fragmenting and quenching the melt as it relocated downward through the water.

The SERENA Phase I (SERENA-I) project [Reference 5] was the most comprehensive international collaboration exercise on steam explosion, aiming to verify the prediction capability of the FCI codes and to identify the major uncertainties which limit the confidences in those predictions. The major uncertainties related to the explosion processes were identified in the projects. For the uncertainties in the pre-mixing phases, it was noted that the codes tends to overestimate void with respect to the integral experimental data (not local but overall average such as level swell data), resulting underestimation of heat transfer and difficulties to reproduce the pressure data. The void distribution is a key parameter of steam explosion processes and resulting energetics, considering as a key limiting factor on steam explosion strength. In general, void fraction of over 50%, i.e., high water depletion in the FCI mixture, drastically reduces the steam explosion energetics. For the uncertainties in the explosion phases, the material effect that associates with melt fine fragmentation is one of significant factors. There is no model to describe this effect. Also, the composition of corium melt is specifically scenario-dependent as well as design-dependent.

Based upon the model verification exercise of the codes, the SERENA-I conducted the code applicability and identification of uncertainties on estimation of steam explosion loads in reactor cases, Figure 2.3 illustrates the initial and boundary conditions for the reactor case analyses (in-vessel and ex-vessel) in the SERENA-I project. The analysis aimed to verify the codes participated at the project with plausible generic reactor situations; i.e., for in-vessel, the multi-jet configuration and for ex-vessel, a large single jet ejecting from the bottom of the reactor vessel (Figure 1-1).

Magallon et al., [Reference 5] summarized the main conclusions of the SERENA Phase-I projects after the exercise of the FCI codes to reactor situations: (1) in the absence of pre-existing loads, IVSE would not challenge the integrity of the vessel, (2) damage to the cavity which may challenge the integrity of the containment is to be expected for ex-vessel explosion, and (3) the level of the loads cannot be reliably predicted due to the large scatter of results. Those conclusions suggested the need of further efforts on EVSE to decrease the scatter of the predictions to acceptable levels for reactor situations and led to the Phase-II of the SERENA project initiated 2007 to the present (2011). The project focuses on the increase confidence in the calculated containment safety margins for ex-vessel reactor case conditions by performing well-controlled and well-defined prototypic corium experiments, TROI (KAERI) and KROTOS (CEA) as well as increasing the capabilities of FCI models/codes. At present, four FCI codes, JASMINE by JNES, JEMI (IKEMIX/IKEJET) with IDEMO by IKE, MC3D with a group of participants and TEXAS-V by UWM

and VTT are used [Reference 6].

Zuchuat et al. [Reference 14] performed the assessment of the in-vessel and ex-vessel steam explosion risk for three Swiss Beznau (Westinghous PWR), Gösgen (Siemens/KWU PWR) and Leibstadt (GE BWR-6) nuclear power plants. For the ex-vessel analysis, their analysis methodology for the assessment of uncertainties in the potential ex-vessel FCI impulse loads was based on the work of Khatib-Rahbar et al. [Reference 47] which considered the uncertainties in the accident progression variables and the model parameters in probabilistic manner. Khatib-Rahbar et al. [Reference 47] constructed probability density functions for key parameters in SA scenarios and code models and prepared about 1000 to 4000 samples using the stratified Mote-Carlo method (LHS: Latin Hypercube Sampling) code [Reference 48]. Their analysis used the TEXAS code for the deterministic analysis and formulated the probabilistic analysis to treat the EVSE uncertainties. Their results showed that the conditional failure probabilities of the pedestals at the Beznau plant and at Leibstadt were varied between 0.26 and 0.73 and 0.1 and 0.5, respectively. However, since the conditional probability of containment failure given an EVSE energetics is not only the EVSE but also other conditional failure probabilities such as steam generator support failure by the pedestal failure and containment penetration tear failure due to steam generator support failure, the mean conditional containment failure probabilities for Beznau and Leibstadt plants due to EVSE were determined to be $8x10^{-4}$ and $1x10^{-2}$.

Moriyama et al. [Reference 15] evaluated the containment failure probability due to ex-vessel steam explosions using the similar methodology performed by Zuchuat et al. [Reference 14] for typical Japanese LWR plants. The analysis was performed with the 200 sampled stratified Monte Carlo method (LHS) and the 2-D JASMINE code for steam explosion calculation. The analysis estimated the mean conditional containment failure probabilities of 6.4×10^{-2}, 2.2×10^{-3} and 6.8×10^{-2}, for the BWR suppression pool case, the BWR pedestal case and the PWR cavity case, respectively. For the PWR cavity case, the mean conditional failure probabilities were much higher than those for Zuchuat's Beznau plant [Reference 14] since their analysis did not considered other possible conditional failure probabilities due to limited information about the plant structure.

Cizelj et al, [Reference 16] perform an analysis to provide an estimation of the pressure loads on the typical PWR power plant cavity structures due to the steam explosion and to assess the cavity integrity. They used a simplified conceptual steam explosion model for the steam explosion load analysis and a structure analysis model in CFX-5.7.1. They analysis results show that their proposed steam explosion model estimated the conversion ratio of 1% that corresponds to the steam explosion pressure of 40 MPa. The structure analysis simulated by the ABAQUS code for this case showed no damage to the cavity walls. For taking account of the large uncertainties involved in steam explosion model and phenomena, they performed another calculation considered to be more conservative case; the steam explosion energy conversion ratio is 10%. The analysis suggested the maximum steam explosion pressure of 250 MPa (see, Figure 1-2). Figure 1-3(a) indicates that the maximum pressure of approximately 115 MPa is estimated at the cavity wall. The corresponding impulse of 345 kPa-s is deducted from the plot by assuming a triangular pressure profile at the cavity wall with the maximum number of 115 MPa and the pulse width of about 3 ms. Structure analysis results as shown in Figure 1-3(b) shows some minor to medium localized damage in the cavity wall, suggesting that the cavity wall maintains its structural integrity.

Table 1-2 summarizes the relevant studies to evaluate the risk of EVSE in nuclear power plants.

The information suggested that the evaluation of EVSE risk requires the accurate structural response analyses against the steam explosion loads in parallel to perform uncertainty analyses against the key parameters influenced by accident scenarios.

1.2.3. Assessment of EVSE for New Nuclear Reactor Designs

Murphy et al. [Reference 17] evaluated the ex-vessel steam explosion loads against the cavity structure of advanced light water reactors. They proposed and demonstrated a three-step analysis methodology to overcome of a drawback of applicability of 1-D TEXAS code to reactor cases which consisted of (1) analysis for mixing of molten fuel in a large coolant pool with IFCI for obtaining 2-D mixing characteristics and with TEXAS for calculating detailed mixing parameters and subsequent explosion calculation, (2) simulation of explosion with TEXAS, IFCI for explosion propagation, and (3) evaluation of dynamic pressure loading with a hydrodynamic code like CTH for far-field propagation. For the SBWR specific conditions, explosion pressure at the center of the explosion at the axial depth of 3.5 m up from bottom reached about 60 MPa at maximum and decreased down to 20 MPa at 0.75 m apart from the center. It was cautiously concluded that the usefulness of the proposed methodology was identified and needed to be verified with direct comparison with FCI energetic data as it comes available.

The assessment of the ex-vessel steam explosion for the plant design and operation certificate for the AP600 nuclear power plant [Reference 18] was performed. The analysis assumed the global hinged mode of vessel failure and performed with the PM-ALPHA/ESPROSE.m code and the TEXAS-V codes. The estimated maximum impulse at the bottom of cavity was ranged between about 150 and 650 kPa-s. Although the typical impulse of 25 kPa-s may cause damage to the cavity wall, only about 5% elongation of the containment liner due to the maximum impulse of 650 kPa-s was estimated. This result confirmed that the likelihood of the containment failure was remote and thus USNRC granted its design certificate and construction license for AP600.

For the licensing for the AP1000 power plant, the similar analysis [Reference 19] to the AP600 plant was performed by the consideration of the bottom and the side vessel failures. The estimated impulse at the bottom of the cavity was ranged between 9 and 300 kPa-s. Since the maximum impulse of ex-vessel steam explosion for AP1000, 300 kPa-s, was well below that for AP600, the likelihood of the containment failure was bounded to the AP600 plant.

A number of new nuclear reactor designs are under the review process which includes APWR (Mitsubishi Heavy Industry), ESBWR (GE-Hitachi), etc. The APWR design employs the SAM strategy of cavity flooding as a severe accident mitigation and prevention measure. Therefore, the risk of the EVSE needs to be evaluated. Although the detailed analysis is not readily available, the limited information showed that the analysis has been performed with the TEXAS-V code and the JASMINE code. The EVSE loads to cavity wall were evaluated with the LS-DYNA code. From the analysis, it suggests that the risk of EVSE is remote because of the maximum pressure less than 84 MPa, remaining the cavity wall below the elastic strain and also the displacement of the reactor vessel less than 0.1 m (4 inches).

The ESBWR design also adopts the cavity flooding strategy as a severe accident measure with the core-catcher, called BiMAC. The risk of the EVSE was analyzed by the PM-ALPHA/ESPROSE.m code with the DYNA3D structure analysis code. The analysis showed that the maximum pressure

and the impulse at the bottom of the cavity reached up to 500 MPa and 150 kPa-s, respectively. However, it was concluded after the DYNA3D analysis on the cavity wall and the BiMAC core catcher structures that the violation of the containment leak-tightness and of the BiMAC function was unreasonable.

Aya and Corradini [Reference 20] performed the ex-vessel steam explosion analysis using the TEXAS-V code to model FCI phenomena to obtain the maximized energetics and the CFD package (ANSYS V11.0) to model the impact of the explosion dynamic loading in the cavity wall.

Moriyama et al. [Reference 15, 21] performed analysis at JAEA for the typical Japanese BWRs and PWRs. In this analysis, the failure criteria for the wall failure in the BWR Mark-II suppression pool, RPV support failures at the pedestal area in the BWR Mark-II reactors and in typical PWR cavity were used.

- Suppression pool wall failure fragility was defined a failure at the lateral displacement of 20% of the structure thickness by the fluid-structure analysis with AUTODYN-2D code, i.e., impulse of 46.5 MN-s at 50% of log-normal probability distribution with error factor of 2.

- Fragility for the RPV support failure at the pedestal area in Japanese BWR Mark-II Reactors was defined as a failure of the anchor bolts that support RPV, i.e., kinetic energy of 39 MJ at 50% of log-normal probability distribution with error factor of 3.5.

- Fragility for the RPV support wall failure in typical Japanese PWR cavity was defined as a failure in terms of the vessel up-thrust by considering the failure energy of the wall structure that gives a constraint on the hot/cold legs against upward movement of the RPV, i.e., assuming 0.5~2% steel ratio (fraction of rebar cross section in the wall), the energy absorption of 14~94 MJ is obtained and thus kinetic energy of 30 MJ at 50% log-normal probability distribution with error factor of 2.5.

Esmaili et al., [Reference 19] estimated impulses of steam explosion for AP1000 at the bottom of the cavity ranged between 9 and 300 kPa-s. Since the maximum impulses of ex-vessel steam explosion for AP1000, 300 kPa-s, was well below that for AP600, the likelihood of the containment failure was not expected.

The ESBWR design [Reference 22] also adopts the cavity flooding strategy as a severe accident measure with the core-catcher, called BiMAC. The risk of the EVSE was analyzed by the PM-ALPHA/ESPROSE.m code with the DYNA3D structure analysis code. The analysis showed that the maximum pressure and the impulse at the bottom of the cavity reached up to 500 MPa and 150 kPa-s, respectively. However, the DYNA3D analysis on the cavity wall and the BiMAC core catcher structures shows that the violation of the containment leak-tightness and of the BiMAC function was unreasonable. In general, the PWR cavity walls can hold higher dynamic loadings comparing to those of BWR and the core catcher is not designed for withstanding high dynamic pressure or impulses.

MHI [Reference 23] performed the structure analysis using the LS-DYNA code to evaluate the explosion loads to the APWR cavity wall. The maximum dynamic pressure of less than 84 MPa remains the cavity wall below the elastic strain and also the displacement of the reactor vessel less than 0.1 m (4 inches). It is also reasonable to postulate that the structural strength of the APWR cavity can be similar to that of the APR1400 cavity.

Hwang, et al., [Reference 24] performed analysis to assess the integrity of cavity walls in the KNGR empirically examined by evaluating the maximum loading on the cavity wall from the experimental evidence that the maximum conversion ratio of steam explosion observed was approximately 3%. The resulted maximum loading on the wall was less than 2.5 psi-s that is safely lower than the criteria of the containment failure.

Table 1- 1 Summary of relevant studies on in-vessel steam explosion

	Sources	Results
Expert Review	WASH-1400 (1975)	α-mode failure probability of 10^{-2}/yr
	NUREG-1116 (1985)	α-mode failure probability of 10^{-2}~10^{-4}/yr
	NUREG-1524 (1995)	α-mode failure probability of < 10^{-3}/yr Vanishingly small Physically unreasonable
Experiment	OECD/NEA-SRMFCI (1997)	No new evidence that would change or violate the conclusion of NUREG-1525 that ALPHA-mode failure is not risk significant
	ALPHA Tests (1995)	No steam explosion at pressure > 1.0 MPa and at saturated water Negligible α-mode failure probability for medium and high pressure accident scenarios
	FARO Tests (1997)	No or weak steam explosion for UO_2-ZrO_2 mixture
	KROTOS Tests (1997)	
Plant Analysis	Corradini & Swanson (1981)	α-mode failure probability of <10^{-4}/yr
	AP600 (1999)	α-mode failure: physically unreasonable (ROAAM)
	SERENA-I (2005)	Steam explosion would not challenge the integrity of the vessel in the absence of pre-existing loads.

Table 1- 2 Summary of relevant studies about ex-vessel steam explosion

	Sources	Results
Expert Review	NEA/CSNI/R(99)24 (1999)	Insufficient experimental database Real material tests are needed
Experiment	FARO Tests (1997)	No or weak steam explosion for UO_2-ZrO_2 mixture
	KROTOS Tests (1997)	
	SERENA-I (2005)	Damage to the cavity which may challenge the integrity of the containment is to be expected
Plant Analysis	Zuchuat et al. (1997)	Conditional containment failure probability of < 1.0×10^{-2}
	AP600 (1998)	Likelihood of the containment failure is remote
	AP1000 (2004)	Likelihood of the containment failure is bounded to the AP600 plant
	Moriyama (2006)	Conditional containment failure probability of < 6.8×10^{-2}
	Cizelj (2006)	Some minor to medium localized damage in the cavity wall, suggesting that the cavity wall maintains its structural integrity

Table 1- 3 Alpha-Mode Failure Probability Estimates given Core Melt Accident (Cited from Table E.1 in Reference 3)

Participant	NUREG-1116 (SERG-1, 1985)	NUREG-1524 (SERG-2, 1995)	View on Status of Alpha-Mode Failure Issue
Bankoff (USA)	$< 10^{-4}$	$< 10^{-5}$	Resolved from risk perspective
Berthoud (France)	-	Very unlikely	No statement on resolution
Cho (USA)	< WASH-1400*	$< 10^{-3}$	Resolved from risk perspective
Corradini (USA)	10^{-4} - 10^{-2}	$< 10^{-4}$	Resolved from risk perspective
Fauske (USA)	Vanishingly small	Vanishingly small	Resolved from risk perspective
Fletcher (Australia)	-	$< 10^{-4}$	Resolved from risk perspective
Henry (USA)	-	Vanishingly Small	Resolved from risk perspective
Jacob (Germany)	-	Probably Low	Not resolved from risk perspective, needs more quantitative evaluation
Sehgal (Sweden)	-	Physically Unreasonable	Resolved from risk perspective
Theofanous (USA)	$< 10^{-4}$	Physically Unreasonable	Resolved from risk perspective
Turland (UK)	-	$< 10^{-3}$	Resolved from risk perspective

*WASH-1400 best estimate $< 10^{-2}$, SERG-1 consensus estimate $< 10^{-3}$

The SERG-1 column in this table shows the range of estimates to be 10^{-2} to 10^{-4}. The NUREG-1116 shows the range to be 10^{-2} to 10^{-5}. The latter document contains estimates from additional SERG-1 experts are not listed here

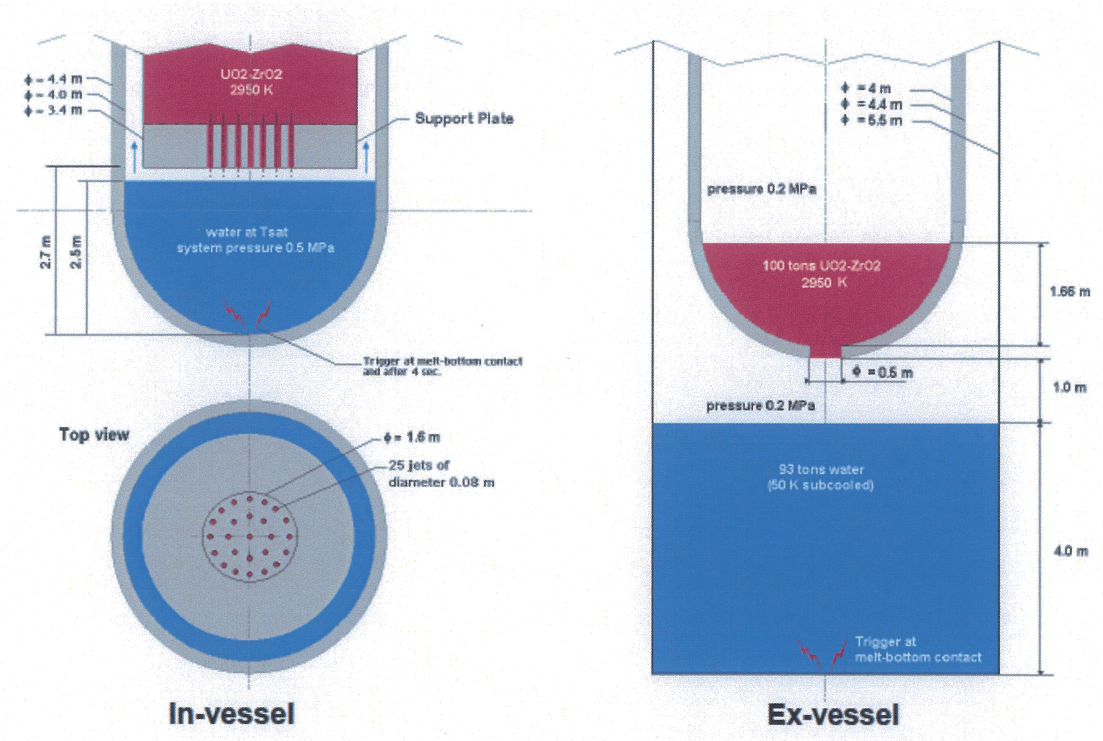


Figure 1- 1 Configurations of Reactor Case Calculation for the FCI Codes in the SERENA-I Project

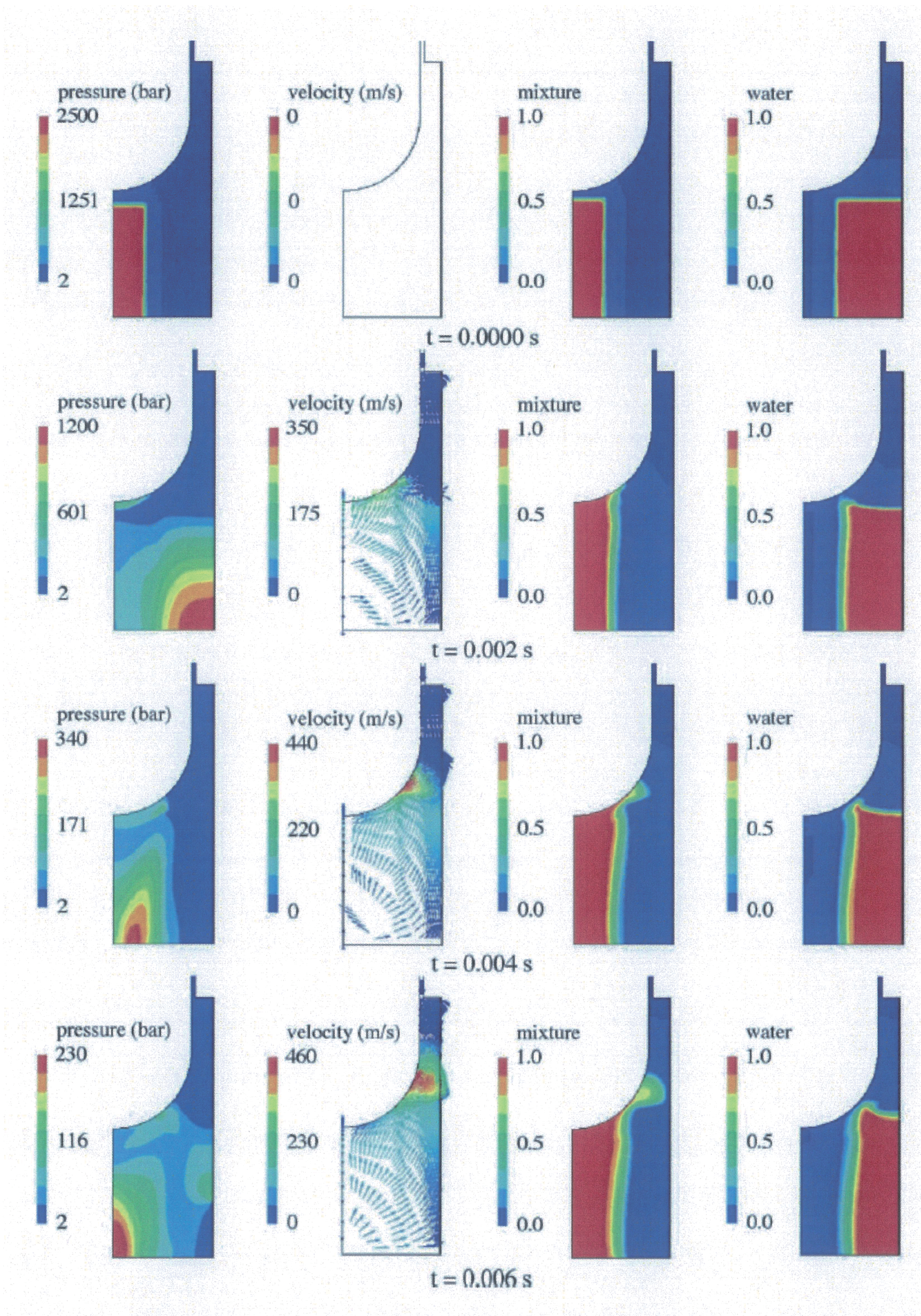
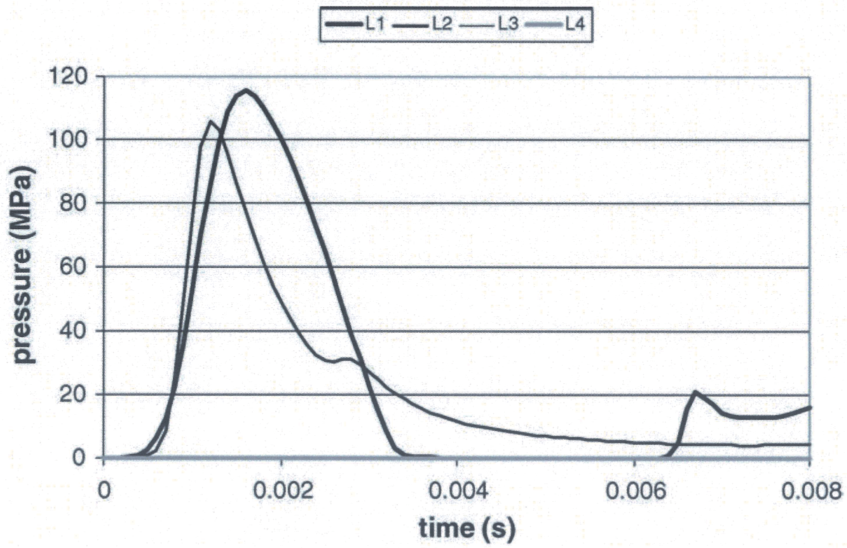
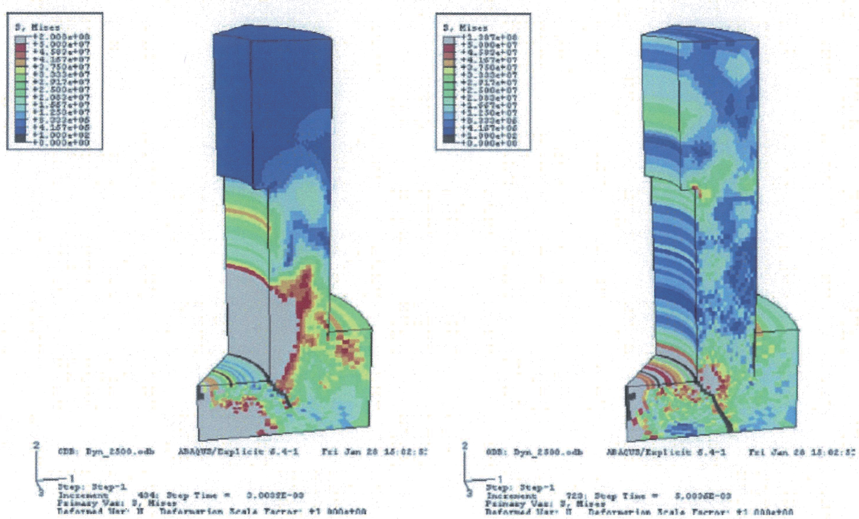


Figure 1- 2 CFD Simulation of Steam Explosion Shock Propagation in a Typical PWR Cavity Geometry done by Cizelj et al. [Reference 16]



(a)



(b)

Figure 1- 3 ABAQUS/Explicit Simulation of the Dynamic Response of the Cavity Wall due to Steam Explosion Loadings done by Cizelj et al. [Reference 16]

2.0 METHODOLOGIES OF STEAM EXPLOSION ANALYSIS FOR APR1400

2.1. Introductions

In this chapter, the methodologies of steam explosion analysis for APR1400 are described. General methods of the analysis include several steps; determination of initial and boundary conditions, evaluation of steam explosion loads, assessment of structure integrity and sensitivity analysis. First, the initial and boundary conditions with the parameter ranges for the sensitivity analyses for both IVSE and EVSE are determined by the severe accident analysis codes, such as MAAP4.0.8. Second, the steam explosion energetics and corresponding loads to the structures in questions such as RPV for IVSE and cavity wall for EVSE are evaluated by the FCI mechanistic code such as TEXAS-V. Third, ABAQUS calculations are carried out to evaluate the RPV structure integrity against the estimated IVSE loads for the IVSE case and to evaluate the cavity structure integrity against the estimated EVSE loads for the EVSE case. Finally, sensitivity analyses of the initial and boundary conditions of key parameters affecting to the steam explosion loads against the surrounding structures are performed. In the case where the safety margin of the structure against the steam explosion loads, uncertainty analysis for the ranges of key parameters against the structure fragility curves is performed to evaluate the conditional failure probability.

2.2. Steam Explosion Analysis Methodology for APR1400

2.2.1. In-Vessel Steam Explosions (IVSE)

The analysis of steam explosion in the real reactor vessel depends on the mass of corium available, the location of the corium contact with water, the corium jet characteristics such as diameter, velocity as well as number of corium jets, etc. Those parameters is not easily estimated in accuracy since they are significantly depends on the accident progression and interaction among available materials and structures. Therefore, for the conservative point of view in the analysis, it is assumed that the RPV internal structures that may influence to the consequence of IVSE are not considered. This simplified configuration allows maximizing the interaction between the corium and water, resulting more bounding estimation compared to the geometry-constraint IVSE. In addition, the TEXAS-V code provided more conservative estimation of IVSE loading since the code is one-dimensional in nature providing the maximum energetics at the given FCI conditions by adjusting the radial mixing zone.

The steam explosion energetics depends largely upon the corium mass participated in the interaction. Therefore, it is assumed that the artificial trigger is provided by the corium jet contact at the bottom of RPV. The less conservative results will be obtained if the corium jet is triggered before or after the bottom contact of corium leading edge to the RPV wall.

2.2.1.1. **Failure Criteria of RPV for IVSE**

For the RPV failure criteria, there are number of approaches in literature. In this section, those criteria are reviewed to be used in the present analysis to evaluate the integrity of RPV in the APR1400 power plants.

Bang et al, [Reference 25] had reviewed several failure criteria as described below. They recognized that Berman [Reference 26] as well as Bohl and Butler [Reference 27] used failure