THERMAL HYDRAULIC AND CORE RELOCATION ANALYSIS ON FUKUSHIMA DAIICHI UNIT 1

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ABSTRACT

Fission product release from the core and deposition in reactor vessel and release to primary containment vessel are dependent on the reactor and plant conditions such as core melt progression sequence and external water injection rate. Liquid mass balance is evaluated with considering various leakages and injections simultaneously. Energy balance is also evaluated considering water discharge through various leakages and metal-water reaction simultaneously. At first, analysis on thermal hydraulic behaviors with core degradation on the RPV and PCV are performed for Fukushima Daiichi Unit 1 The leakage and rupture conditions (pressure and temperature) and opening sizes of RPV and PCV are provided. Alternative water injection flow rates are also provided using the boundary conditions provided by OECD/NEA BSAF project. A sensitivity study is performed to find out the pressures in RPV and PCV matching with plant measurements on the three flow rate curves and on the no flow condition additionally. For the best matching case another sensitivity study are performed on varying SC vent open sizes. The release rate of fission product vapors in the fuel and deposition of fission product aerosols on the various heat structures in RPV and PCV are estimated. Fission product aerosol release to the reactor building and environment is also estimated. A severe accident analysis code, MELCOR version 1.8.6 is used in this analysis.

KEYWORDS
Fukushima Daiichi Accident, OECD/NEA BSAF Project, Core relocation sequence, External water injection, MELCOR code

1. INTRODUCTION

In-plant thermal hydraulic and core relocation behaviors are estimated for Fukushima Units 1. In the OECD/NEA BSAF project two cases are analyzed; The one, 1) common case analysis using boundary conditions given by the project without any modification and the other case, that is, 2) best estimate case with some modification on the boundary conditions by the user. Analysis results of best estimate case on the severe accident of Fukushima Daiichi Unit-1 are described in this report. The analysis was performed by using MELCOR 1.8.6. Plant input data and information of geometry were obtained from TEPCO through BSAF project. The calculation time and result data form are the same with the common case. In this calculation, given boundary conditions were modified to minimize the difference between calculated value and measured value (RPV and PCV pressure). Provided below are discussions on the selection of best estimate boundary conditions and justifications.

The leakage information provided by BSAF is also used. The boundary conditions on leakages are leakage through SRM instrument pipe, SRV gasket leakage, MSL flange leakage, etc. They have some influence on the RPV and PCV pressure behaviors before RPV lower head failure. However, external water injection rate has a much stronger influence on pressure behavior than leakage.

Table 1. Major plant events log in Fukushima Daiichi Unit 1

Date/time	time since reactor trip (hr)	event
2011-03-11 14:47	0.00	earthquake (reactor trip)
2011-03-11 14:52	0.08	IC starts (A&B)
2011-03-11 15:03	0.27	IC stops (A&B)
2011-03-11 15:27	0.67	arrival of first tsunami
2011-03-11 15:35	0.80	arrival of second tsunami
2011-03-11 15:37	0.83	complete AC power loss (SBO)
2011-03-12 02:30	11.72	DW pressure reached 840 kPa[abs]
2011-03-12 05:46	14.98	fresh water injection starts
2011-03-12 14:30	23.40	SC vent starts
2011-03-12 14:50	24.05	SC vent stops
2011-03-12 15:36	24.82	explosion occurs at reactor building
2011-03-12 19:04	28.28	sea water injection starts
2011-03-12 20:45	29.97	sea water injection mixed with boric acid

Ref. : Examination of Accident at Tokyo Electric Power Co., Inc.'s Fukushima Daiichi Nuclear Power Station and Proposal of Countermeasures, October 2011, Japan Nuclear Technology Institute (JANTI), Examination Committee on Accident at Fukushima Daiichi Nuclear Power Station

2. MELCOR MODELING

2.1 RPV Model

In the case of Unit 1, the RPV consists of downcomer, lower plenum, core channel, core bypass, shroud dome and steam dome as described in Table 2. The core parts of each unit were modeled as a single control volume, so the effect of temperature distribution according to the location within multiple core cells was ignored. Active core part (channel + bypass) has 10 axial levels and 4 radial concentric rings. The course mesh of the RPV that is used in control volume hydrodynamics model is sufficient because this study is focused on the fuel relocation behavior to lower plenum and to PCV.

2.2 PCV Model

The primary containment vessel was divided into four regions such as pedestal, drywell, vent leg, and wetwell as described in Table 2. The elevation of the lower part of RPV was set to the top of the pedestal. When RPV pressure increases to SRV opening set pressure, then SRV is open and the steam in RPV releases to wetwell (suppression chamber). If RPV pressure decreases below the SRV closing set pressure, then SRV recloses. The vacuum breaker is located in vent leg. If the differential pressure between wetwell and drywell exceeds the set pressure, the steam and non-condensable gas such as hydrogen are not suppressed in the wetwell and bypass directly to the pedestal and drywell regions. PCV is inerted by nitrogen (N_2) gas so that hydrogen burning will not occur during normal operation. It is recorded in the plant log that wetwell (SC) venting was tried at about 23 h after reactor scram.

2.3 Cavity Model

One cavity model is employed to simulate MCCI reaction. Flat bottom cavity geometry is used in MELCOR code. Basaltic concrete is assumed. 13.5% of rebar composed of Fe is assumed. Initial inner and outer concrete radiuses are 3m and 6 m, respectively. Radial concrete depth is 3 m thick. Initial axial depth is 5 m and concrete thickness is assumed to be 3 m.

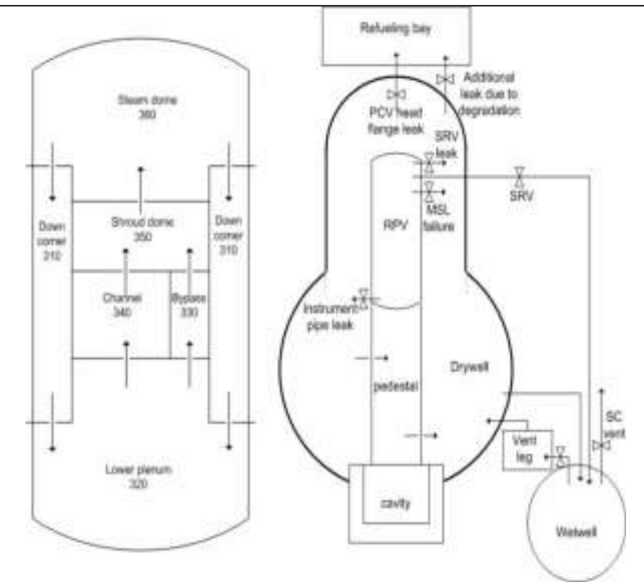

Fig. 1 Nodalization of RPV and PCV

Table 2. Control Volume Assignment

CV number	CV Name
RPV	
CV310	Downcomer
CV320	Lower plenum
CV330	Bypass
CV340	Channel
CV350	Shroud dome
CV360	Steam dome
PCV	
CV100	Pedestal
CV101	Drywell
CV105	Vent leg
CV200	Wetwell
Others	
CV402	Reactor building
CV408	Refueling bay
CV700	Environment

2.4 Various Leakage Models

Beside the normal opening flow paths from RB to TB or from RB to ENV, various transient leakages are assumed due to the increasing temperature and pressure of control volumes and heat structures. The leakage triggering conditions and opening areas are summarized in Table 3. In Unit 1 analysis, main steam line (MSL) failure (FL903) and PCV head flange leakage (398) were considered. In addition, instrument pipe leakage, suppression chamber vent and enlargement of leak size due to degradation were also modeled. Leakage or rupture models are suggested by the BSAF project as common case boundary conditions. RPV water can be leaked through SRV valve opening (FL362). The coolant in the RPV discharges to suppression chamber (SC) by opening SRV valves. SRV opening and closing pressures are provided by TEPCO. As temperature or pressure builds up in RPV instrument pipe (SRM/IRM/TIP) (FL901), SRV gasket (FL598), and MSL flange (FL903) can be degraded resulting in leak or rupture. The size and degradation condition (temperature and pressure) are provided by BSAF as common case boundary conditions. These conditions can be modified when best estimate analysis are done by each institute.

2.5 SC Vent Operation (FL921)

Manual SC vent is modeled (FL921). The size of vent valve and real opening size is suggested by the project. FL921 describes the flow rate of SC vent operation from 23.4 hr to 24.27 hr. Full flow area of SC vent large valve is 1.0363×10^{-2} m^2. Simulated valve open areas are 6.5%, 10%, 20%, and 30% of full open area.

Table 3. Leakage and Rupture Flowpath Assignment

FL number	From CV	To CV	FL name	Failure conditions	Leakage area (m^2)
From RPV to PCV					
FL362	360	200	SRV cyclic open and closure	OPEN P > 7.38 MPa CLOSE P < 7.00 MPa	7.54E-3
FL999	360	200	SRV stuck open	Tgas(CV360) > 1000K	7.54E-3
FL598	360	101	SRVs gasket leakage	Tgas(CV360) > 773K	7.54E-3
FL901	320	101	Instrument pipe leakage	Tgas(CV320) > 1000K	1.4E-4
FL903	360	101	MSL flange leakage	IF (P(360) – P(101)) > 0.2MPa, THEN LM-CREEP RUPTURE	1.36E-3
FL399	320	100	RPV lower head penetration failure	T(PEN) > 1273K	1.0E-2
From PCV to RB					
FL398	101	408	PCV head flange seals failure	P(CV101) > 7.5E5 Pa	
FL907	101	402	PCV leakage at 50 h after reactor trip	TIME > 50 HR	2.0E-4
FL909	101	402	PCV leakage at 110 h after reactor trip	TIME > 110 HR	2.5E-4
FL921	200	402	SC venting	23.4 < TIME < 24.27 HR	1.0363E-2 x 6.5%
From RB to ENV					
FL950	408	410	Refueling Bay opening at Explosion	TIME > 24.82	25.0

2.6 Lower Head Failure Model (FL399)

Two failure modes were considered. One is the penetration due to high temperature (>1273.15 K) and the other is creep rupture failure based on a Larson-Miller parameter. Lower head penetration failure and main steam line (MSL) rupture are modeled with Larson-Miller creep rupture model.

2.7 Isolation Condenser Operation

Operation times on Isolation condensers (IC) are provided in the TEPCO report. Utility functions as a function of RPV pressure is suggested in SNL Fukushima analysis report. This utility function is used in this analysis. Earthquake occurs at 14:46 at March 11, 2011. IC A and B are operated during 11 minutes from 14:52 to 15:03. IC A is operated and stopped three times more before tsunami hit at 15:35.

2.8 Alternative Water Injection

Alternative water injection (AWI) flow rates of fresh water and sea water is provided by the BSAF project. Alternative water is modeled to be injected to reactor core channel (CV340) with constant temperature of 300K. Fresh water injection starts at 15 h from reactor trip. Sea water injection starts at 28 h. Three kinds of injection flow rates are provided by BSAF project, but one more case "no injection flow" is analyzed additionally in this paper.

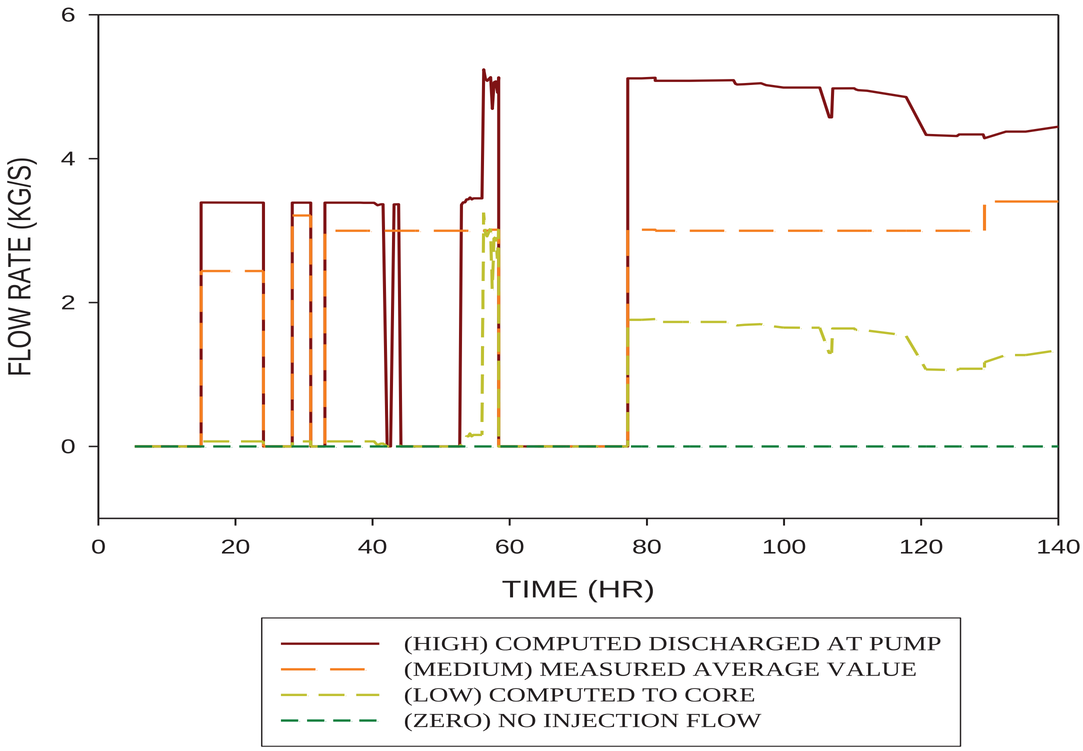

Figure 2. External water injection mass flow rate (kg/s)

3. THERMAL HYDRAULICS BEHAVIOR IN RPV AND PCV

3.1 Evaluation on appropriateness of AWI scenario

A sensitivity study is performed for given four kinds of AWI scenario in order to see which injection curve is the most appropriate. The measured RPV and PCV pressures in the plant are used as a target parameter to decide which AWI scenario is the most appropriate. Figure 3 shows the results of this sensitivity study. No injection scenario is decided to be the best matching with measured data. The other three scenarios on AWI result in unrealistic over pressure compared to measured pressure data. In these high pressure environments in RPV and in PCV, the fire pump may not be able to inject external water into RPV and PCV. The thermal hydraulic behaviors in RPV plus PCV are discussed for the no injection scenario.

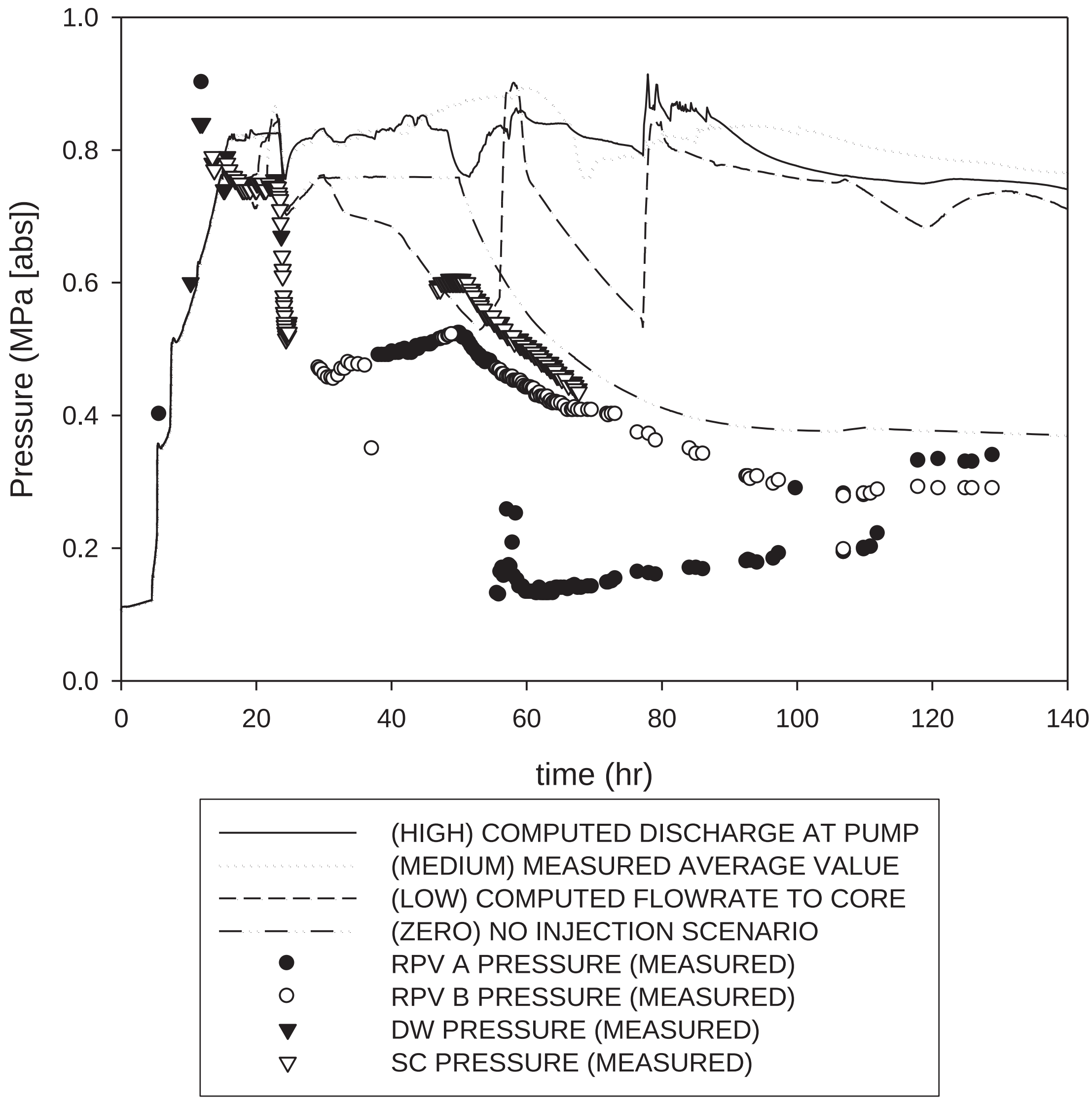

Figure 3. Calculated PCV pressures for 4 AWI scenarios and comparison with measured data

3.2 Water level and pressure of RPV

Figure 4 shows collapsed in-shroud water level in RPV. Almost all the liquid water in the RPV was eliminated at 6.8 h after reactor scram as shown in figure 4.

Figure 5 shows the RPV pressure behavior up to eight 15 h. You can see the effect of IC operation befrom 1 h after reactor scram. You can see also the fast opening and closing of SRV from 1 h to 4h. However, cycling of SRV is slower between 4 h to 5.2 h. At 5.2 h SRV is assumed stuck open.

Figure 5 shows pressure transients of RPV and PCV up to 140 h after reactor scram. The pressure in RPV is decreased during first one hour due to IC operation. SRV opens and closes cyclically from 1 h to 5.3 h. SRV stuck open is assumed at 5.3 h.

Figure 6 shows integrated leakage masses on various leakage flow paths from RPV to PCV. Initial water mass in RPV was 137 tons. Among 137 tons of initial RPV water mass, about 90 tons are discharged to SC during initial 5 hours by the cyclic opening of SRV (FL362). Only 47 tons of water is discharged through various leakages and ruptures thereafter. Total 127 tons of water leaked from RPV due to various leakages assumed. The 10 tons of water mass difference between 137 tons of initial water and 127 tons of total water leaked up to 10 h is consumed by metal oxidation reaction.

Lower head failure (FL399) starts at 6.74 h by CRD tube penetration failure. However, the molten corium was not ejected through the reactor pressure vessel. In order to eject the debris, a melt fraction of 0.1 (total molten mass divided by total debris mass) is necessary in MELCOR.

Figure 7 shows integrated leakage masses on various leakage flow paths from PCV to RB. Total 60 tons of liquid discharged from PCV to RB. Figure 7 shows that total leakage mass of drywell head flange seals failure (FL398). The leakage path is from drywell (CV101) to refueling bay (CV408). Total leaked mass is 10 tons. The leakage starts at 13 h and ends at 20.2 h. The real hydrogen explosion occurred at about 25 h at the real plant of Fukushima Daiichi Unit 1. This leakage might be a main reason of reactor building roof (refueling bay) explosion of Unit 1.

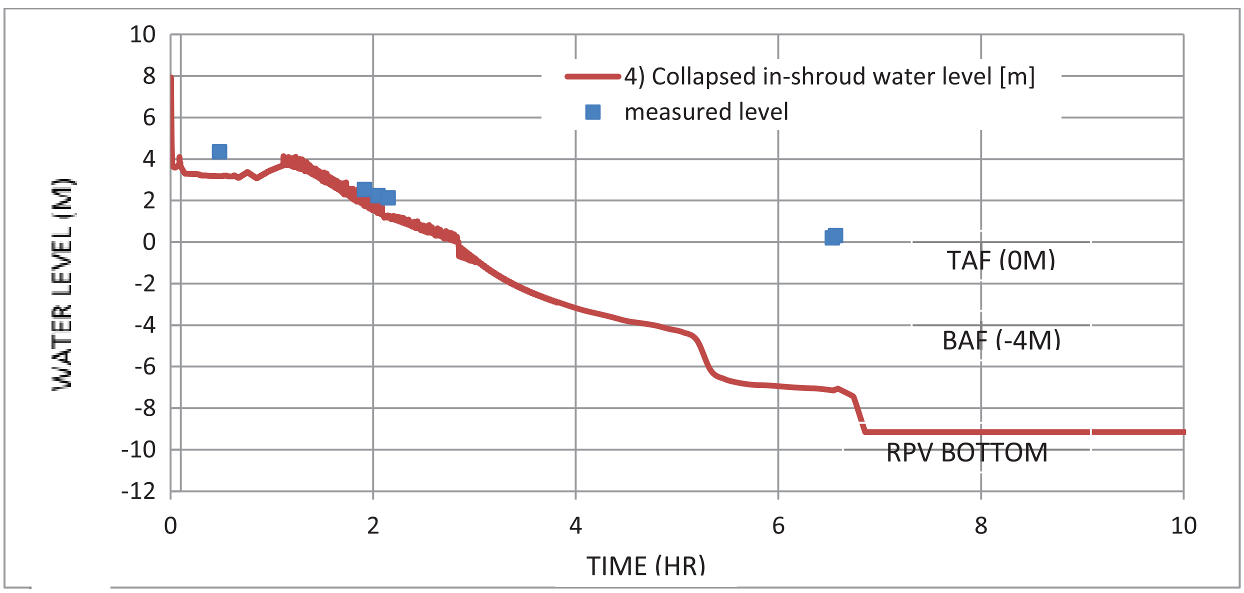

Figure 4. Collapsed in-shroud water level (m)
(level 0 m is top elevation of core fuel (TAF) and level -4m is bottom of active fuel (BAF))

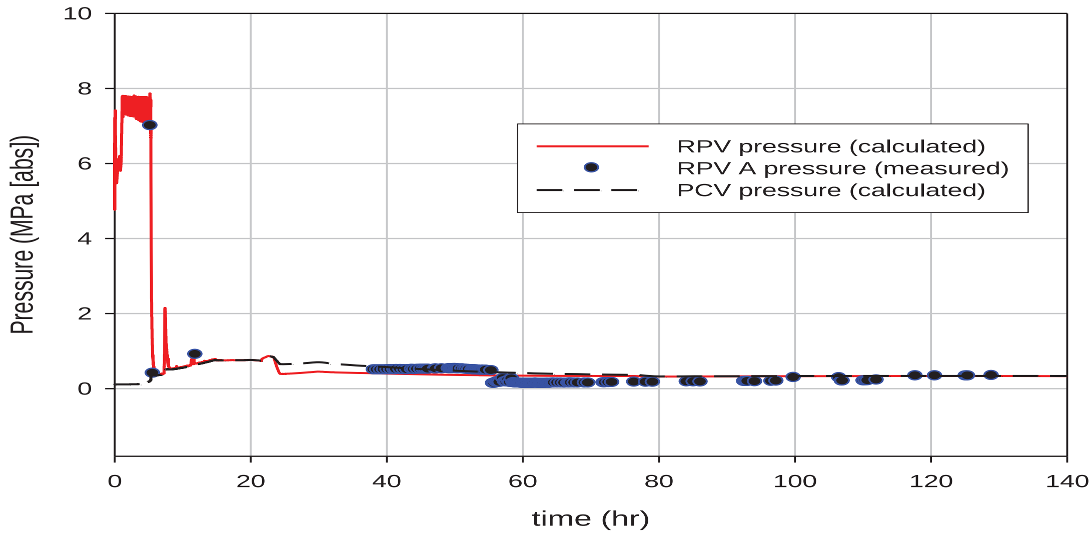

Figure 5. Pressure of RPV and PCV

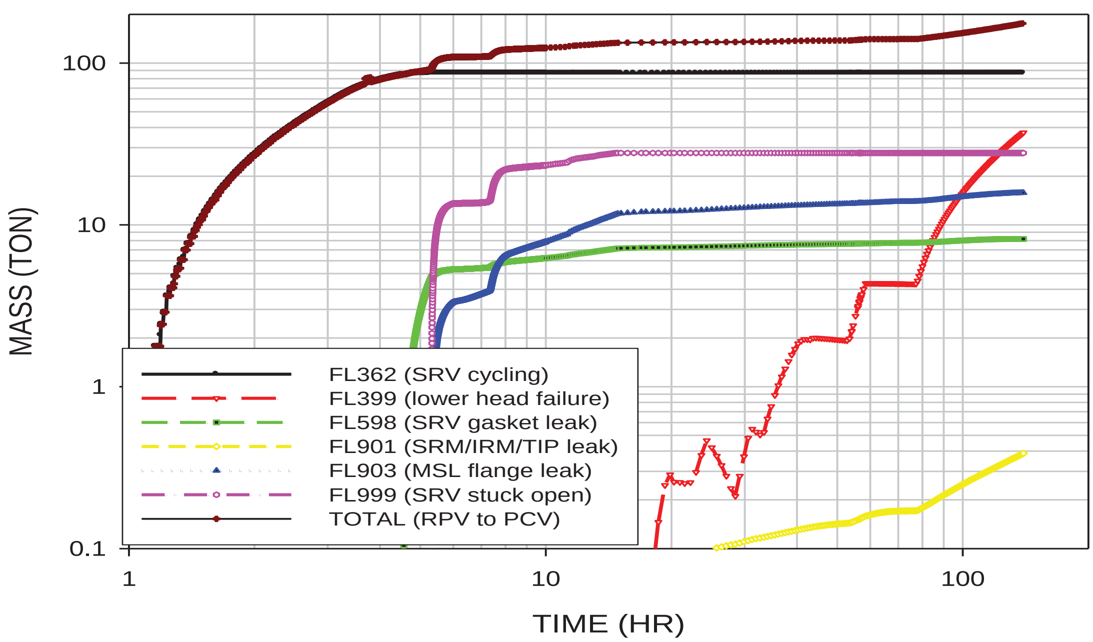

Figure 6. Integrated leakage masses from RPV to PCV

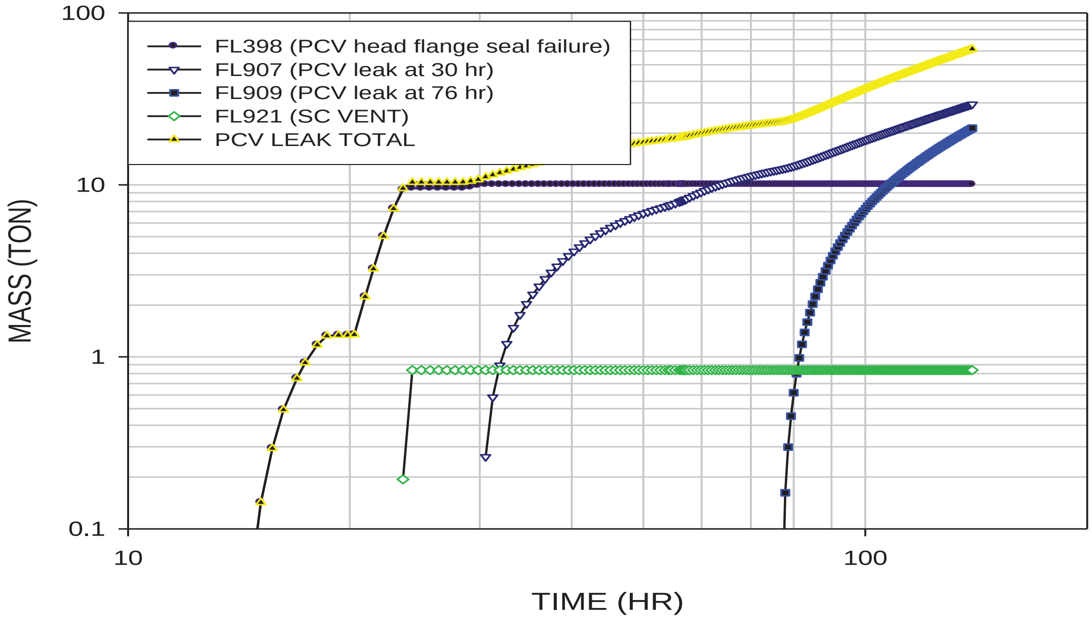

Figure 7. Integrated leakage masses from PCV to RB

4. CORE RELOCATION ANALYSIS

Core heat up and subsequent core relocation occurs due to the loss of coolant inventories through various leakages. Hydrogen can be generated from zirconium water reaction and from steel water reaction. The zircaloy oxidation reaction occurred at 4.5 h.

Figure 8 shows hydrogen generation rate in core. From the total 612 kg of hydrogen generated by metal water reaction, 490 kg (80%) and 122 kg (20%) of hydrogen are generated from zircaloy and steel, respectively. The hydrogen is generated from 4.5 h to 23 h.

Debris ejection to cavity occurs from 20 to 23 h. No material remains at the core after this time as shown Figure 9.

Figure 10 shows the heat generation and removal rates in core. Red line from 4.5 to 20 h indicates heat generation by metal water reaction.

Figure 11 shows decay heat generation in core and cavity. The difference between DCH-CORPOW and CORE-EFPD-RAT is due to the contribution of volatile fission products. Volatile fission products move to other locations (RB, TB, ENV) than core or cavity regions.

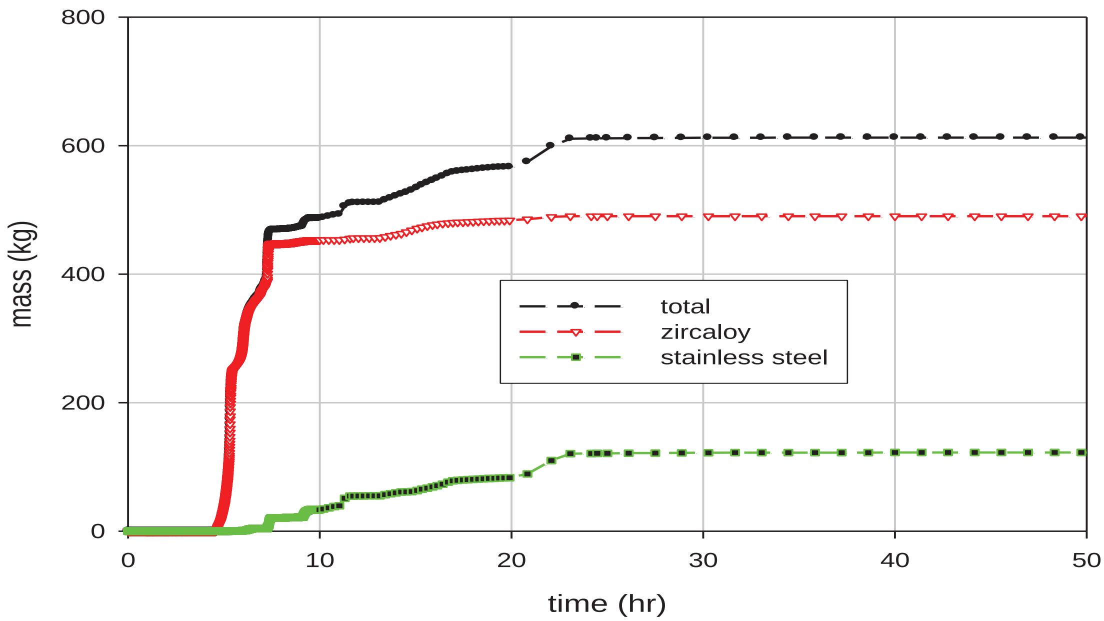

Figure 8. Hydrogen generation rate

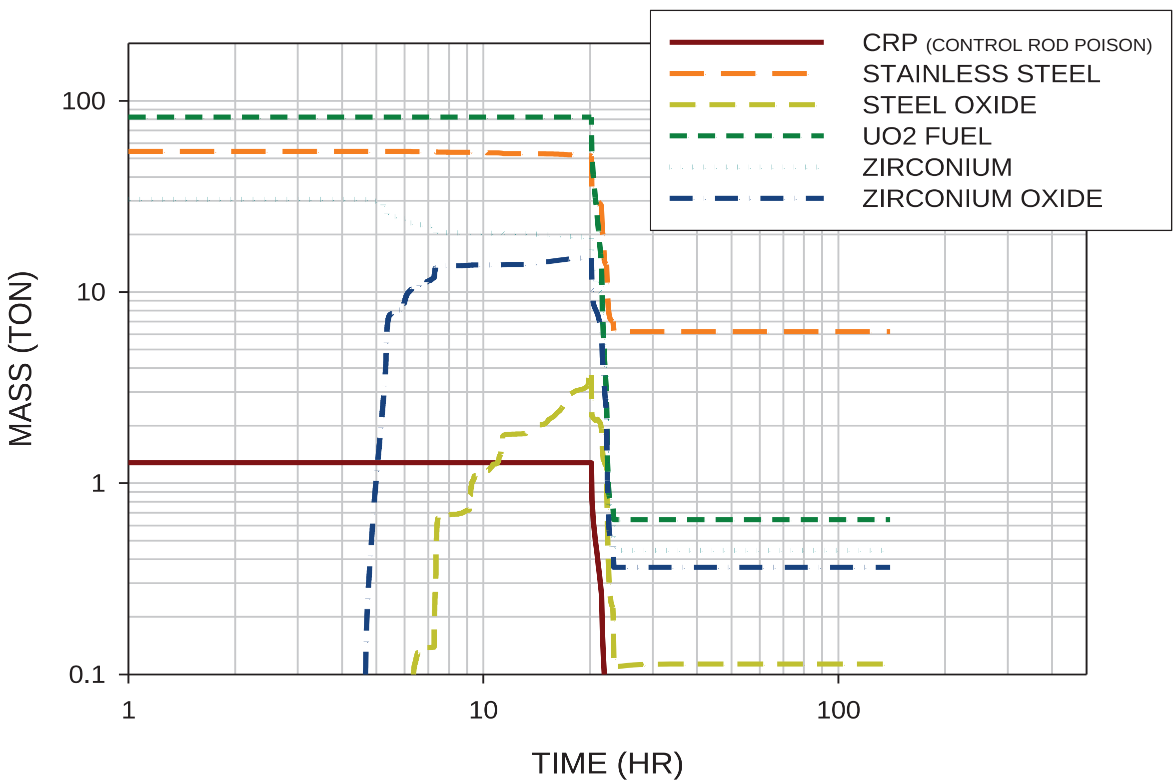

Figure 9. Core material mass change

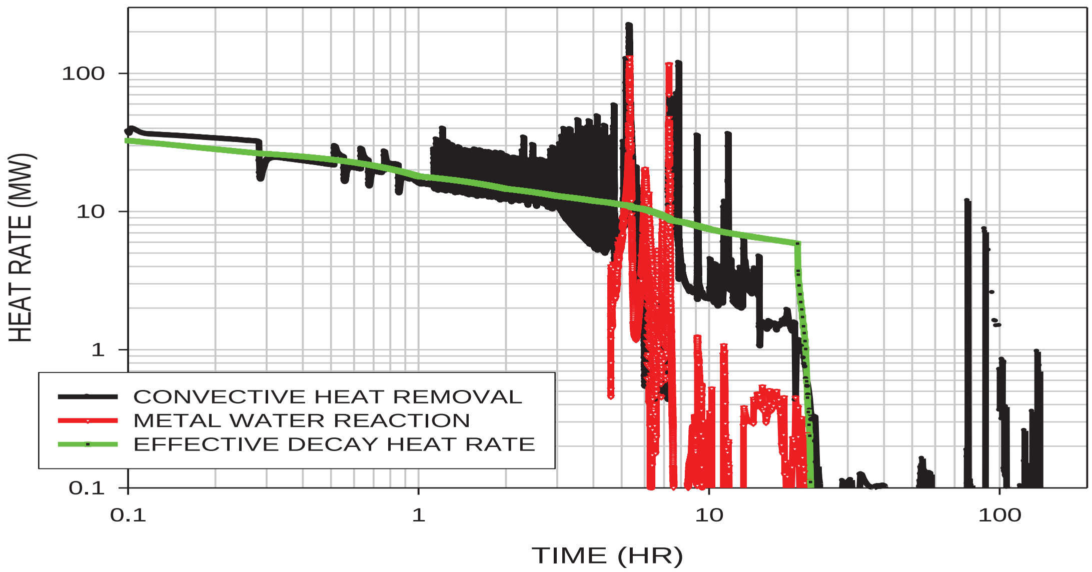

Figure 10. Heat balances at reactor core

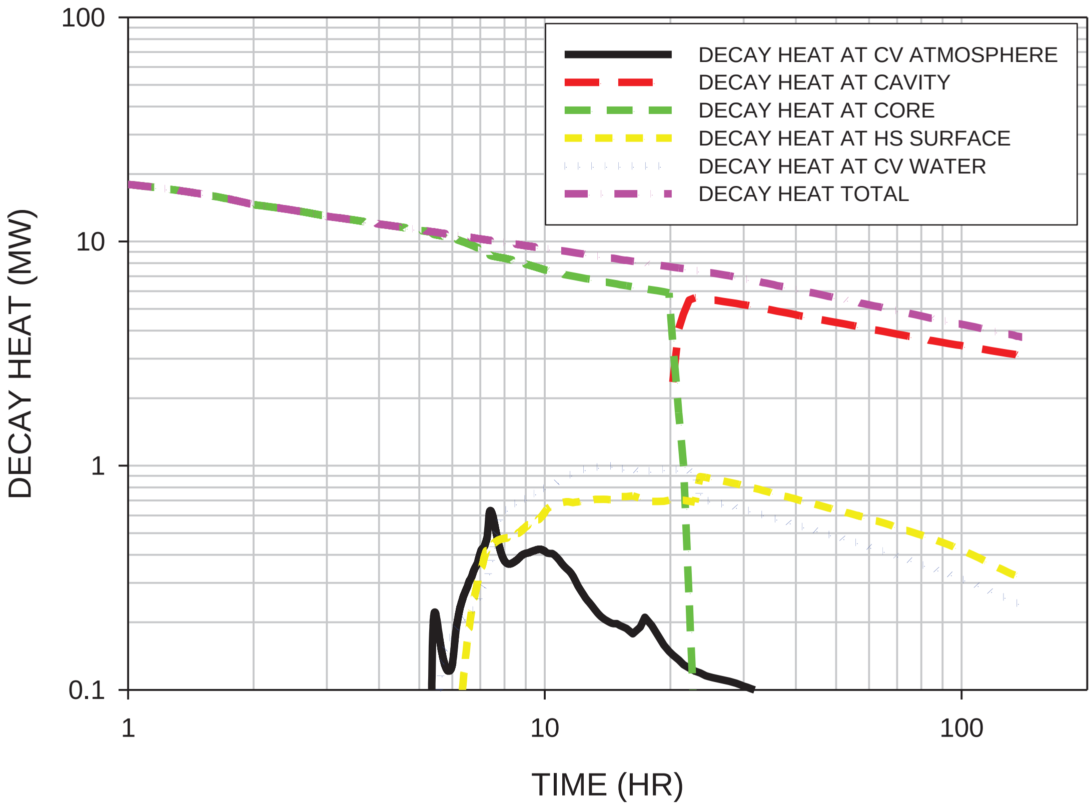

Figure 11. Distribution of decay heat to various places calculated by RN1 Package

5. FISSION PRODUCT RELEASE TO COMPARTMENTS

Figure 12 shows cesium (Cs) concentration (airborne plus deposited on heat structures) in each compartment. Total 6% of initial inventory of cesium released to the environment at time of 140 h. The other 94% are retained in the compartments. Approximately 40%, 30%, and 15% initial inventory of cesium are retained in the suppression chamber, pedestal + drywell, and RPV, respectively. Retained fraction in turbine building and reactor building is less than 1% at time of 140 h.

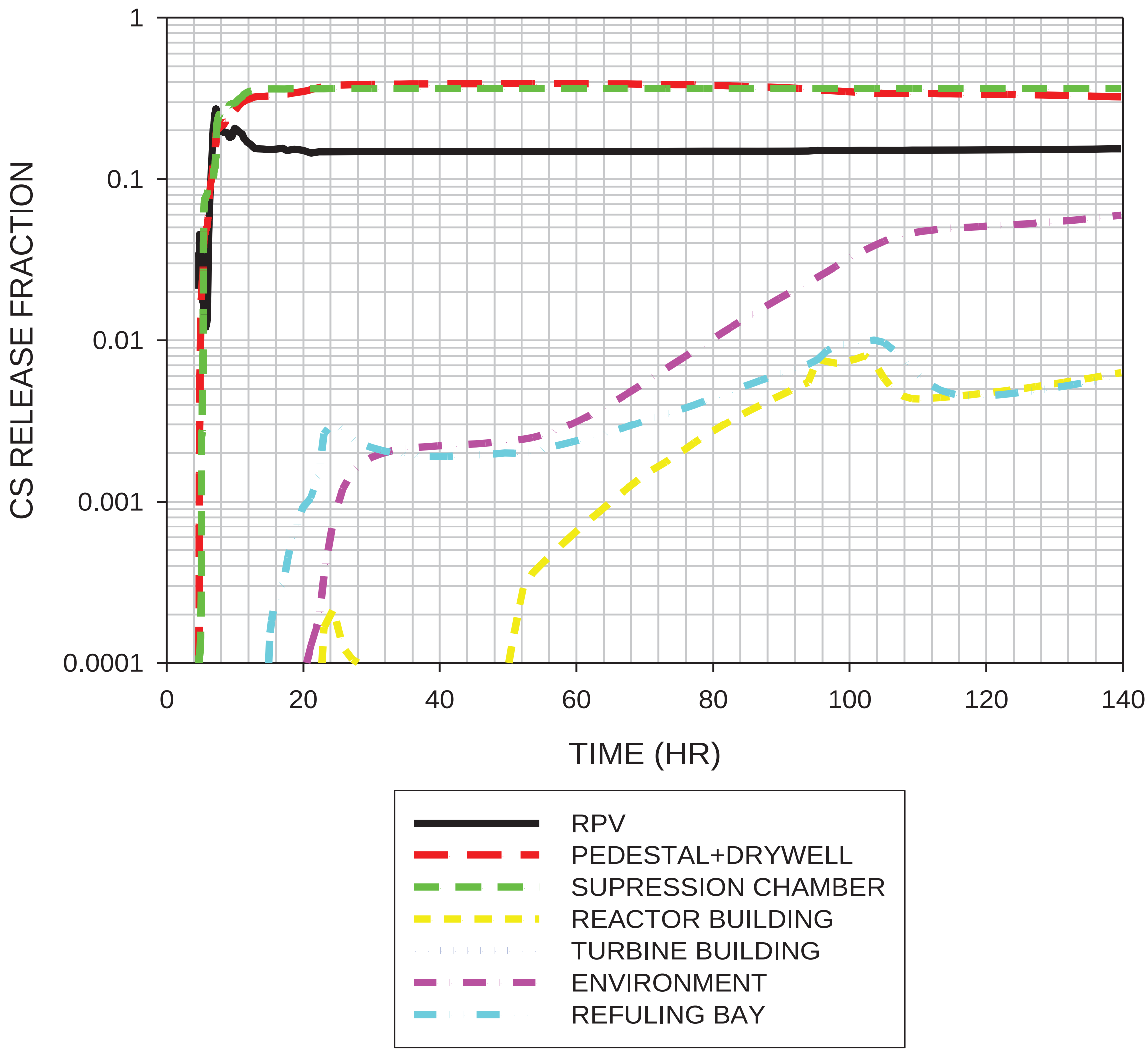

Figure 12. Cs Concentration in each Compartments (airbone + deposited on heat structures)

6. CONCLUSIONS

MELCOR analysis on Fukushima unit 1 estimated that the core damage and lower head penetration failure start at 5 h and ends at 20 h after reactor scram, respectively based on the boundary conditions provided by BSAF project. The pressure in the PCV after the failure of RPV lower head is not matched well between measured and calculated values. Therefore, in the best estimate case analysis, the external water injection is assumed not to be injected into the RPV at all. In this case, the pressure in the RPV and PCV is well matched with plant measurement data. The fresh and sea water which were tried to be injected to the RPV by fire pump truck would be diverted to other plant systems through bypass paths.

RPV lower head failure is believed to have occurred based on this analysis. Most of the corium released through lower head penetration is believed to have interacted with basemat concrete on the pedestal floor. It is estimated that more than 95% of core materials are relocated to the pedestal region due to the failure of the reactor vessel lower head.

The fission products generated from the damaged core are released to the wetwell and drywell at first by SRV cycling open and closures. After the failure of the lower head penetrations at 20 h, fission products retained in the molten corium or debris beds are released to the pedestal. Finally they are released to the other locations such as reactor building, turbine building and environment through various leak paths already existed before the SC venting operation or after the venting. The explosion at the refueling bay might occur due to leakages of hydrogen from drywell head flange seals degradation due to high pressure and temperature in the drywell top region. Total 6% of initial inventory of cesium released to the environment at time of 140 h.

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