4.2 Development Status of Severe Accident Analysis Code SAMPSON

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ABSTRACT

The Four years of the IMPACT, 'Integrated Modular Plant Analysis and Computing Technology' project Phase 1 have been completed. The verification study of Severe Accident Analysis Code SAMPSON prototype developed in Phase 1 was conducted in two steps. First, each analysis module was run independently and analysis results were compared and verified against separate-effect test data with good results. Test data are as follows: CORA-13 (FZK) for the Core Heat-up Module; VI-3 of HI/VI Test (ORNL) for the FP Release from Fuel Module; KROTOS-37 (JRC-ISPRA) for the Molten Core Relocation Module; Water Spread Test (UCSB) for the Debris Spreading Model and Benard's Melting Test for Natural Convection Model in the Debris Cooling Module; Hydrogen Burning Test (NUPEC) for the Ex-Vessel Thermal Hydraulics Module; PREMIX, PM10 (FZK) for the Steam Explosion Module; and SWISS-2 (SNL) for the Debris-Concrete Interaction Module. Second, with the Simulation Supervisory System, up to 11 analysis modules were executed concurrently in the parallel environment (currently, NUPEC uses IBM-SP2 with 72 process elements), to demonstrate the code capability and integrity. The target plant was Surry as a typical PWR and the initiation events were a 10-inch cold leg failure. The analysis is divided to two cases; one is in-vessel retention analysis when the gap cooling is effective (In-vessel scenario test), the other is analysis of phenomena event is extended to ex-vessel due to the Reactor Pressure Vessel failure when the gap cooling is not sufficient (Ex-vessel scenario test). The system verification test has confirmed that the full scope of the scenarios can be analyzed and phenomena occurred in scenarios can be simulated qualitatively reasonably considering the physical models used for the situation.
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Keywords Severe accident analysis, Mechanistic model simulation, Parallel processing, In-vessel retention, Ex-vessel phenomena

I. INTRODUCTION

IMPACT (Integrated Modular Plant Analysis and Computing Technology) is the name of a program and of specific simulation software, which will perform full-scope and detailed calculations of physical and chemical phenomena in a nuclear power plant for a wide range of scenarios [1-2]. The main analysis objectives are to show that light water reactors (PWR/BWR) maintain safety margin under hypothetical severe accident conditions and to investigate realistic measures of accident management by simulating the accident in detail and on a reasonable time frame using a parallel computer. To satisfy the objectives, it is required to calculate various behaviors of the fuel cladding damage, fuel melting, candling, crust forming, molten debris cooling, fission products release, etc., and to evaluate reactor vessel and containment structural integrity. Because these phenomena are difficult to investigate experimentally, analytical evaluations of the safety margin must be relied upon. To satisfy these analytical needs, computer simulations based upon fundamental physics principles and sophisticated modeling technologies are required. Separate software modules for micro-, meso-, and macro-scale modeling will be developed which may be combined in the most systematically efficient manner in a parallel environment.

The four years of the IMPACT project Phase 1 have been completed. At the end of the phase, demonstration simulations by combinations of up to 11 analysis modules developed for severe accident analysis in the SAMPSON Code (Severe Accident Analysis Code with Mechanistic, Parallelized Simulations Oriented towards Nuclear Field) were performed and verified [3-6]. The Basic Single-, Two-, and Multi- Phase Flow Analysis Modules (PLASHY: Parallel Large scales Analysis System for HYdrodynamics) have been Parallelized in various coordinate systems [7]. The physical models in the Boiling Transition Analysis Code (CAPE: Critical power Analysis code with Parallel Environment) have been completed and verified for both BWR fuels critical power and PWR fuels DNB prediction capabilities by comparison with fundamental experimental results [8]. The Fluid-Structure Interaction Analysis Code (FLAVOR: FLuid structure interaction Analysis code for VORtex induced vibration) was developed and verified for both 2 dimensional cross flow fluid-structure interaction analysis and 3 dimensional flow induced vibration analysis [9].

II. MODULE VERIFICATIONS

The verification study of the code was conducted in four steps. First, each analysis module was run independently and analysis results were compared against separate-effect experiment data. Verification analyses included: CORA-13 (FZK) for the Fuel Rod Heat-up Module; HI/VI Test-VI-3 (ORNL) for the FP Release from Fuel Module; KROTOS-37 (JRC-ISPRA) for the Molten Core Relocation Module; Water Spread Test (UCSB) for the Debris Spreading Model and Benard's Melting Test for Natural Convection Model in the Debris Coolability Module; Hydrogen Burning Test (NUPEC) for the Containment Thermal Hydraulics Module; FARO-L14 (JRC-ISPRA) for the Steam Explosion Module; and SWISS-2 (SNL) for the Debris Concrete Reaction Module. Second, module combination tests for dominant phenomena were performed to improve the compatibility of related analysis modules with each other. Third, with the Analysis Control Module, these analysis modules were executed concurrently in the parallel computing environment (IBM-SP2 with 72 processing elements), to demonstrate the code capability and integrity for an in vessel event. The reference plant was Surry as a typical PWR and the initiating event was a 10-inch cold leg failure LOCA. The system analysis was performed to investigate in-vessel retention when water cooling option was adopted and gap cooling was effective (In-vessel scenario test). Fourth, the analysis was extended to the ex-vessel. The reactor pressure vessel failed when steam-cooling options was used and gap cooling was not sufficient (Ex-vessel scenario test).

1. Separate Effect Test Analyses

The 10 test analyses were performed to check programming of the 9 modules, shown in Fig.1.

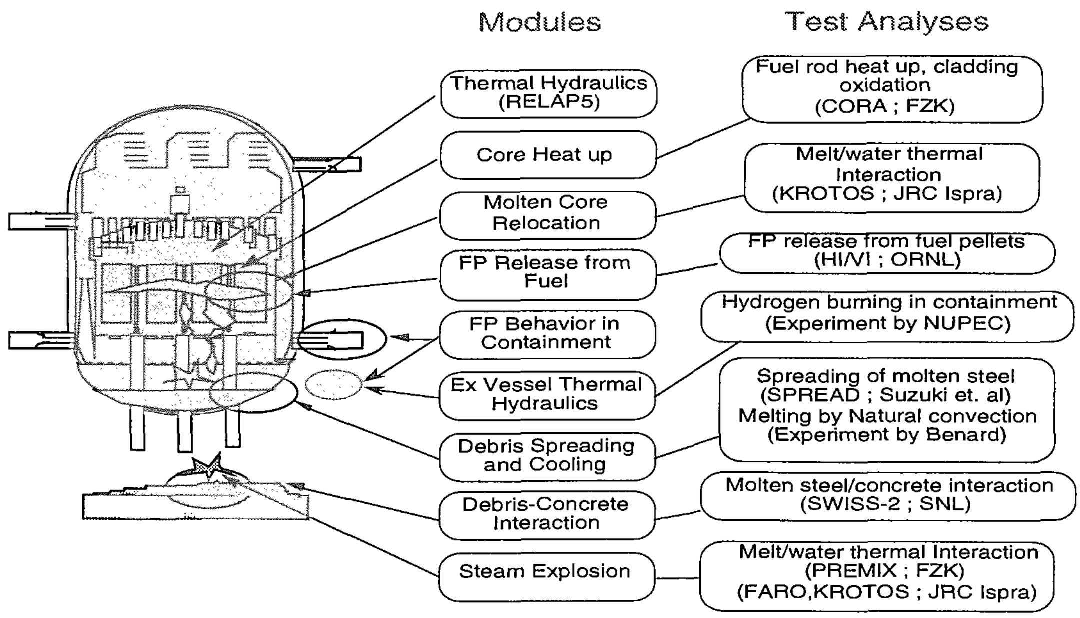

Figure 1 Modules and Test analyses for verification

CORA-13 TEST data were used for Heat up module validation. Fig.2 shows the comparison results. In experimental data, rod surface temperature history is divided into two parts. The first is heat up phase up to 1000 degrees and the second is Zr-water reaction phase with rapid heat generation. Calculation result shows good agreement with experimental data in both phases. But in hydrogen gas generation rate, there are some differences between calculation and experimental data. The module calculates hydrogen generation in the lower temperature region. After 4000second, hydrogen gas generation increase rapidly same as experimental data. The decrease of hydrogen gas generation was caused by lack of metal zirconium. That is end of Zr-water reaction. The peak of hydrogen generation in experiment was caused by reflooding.

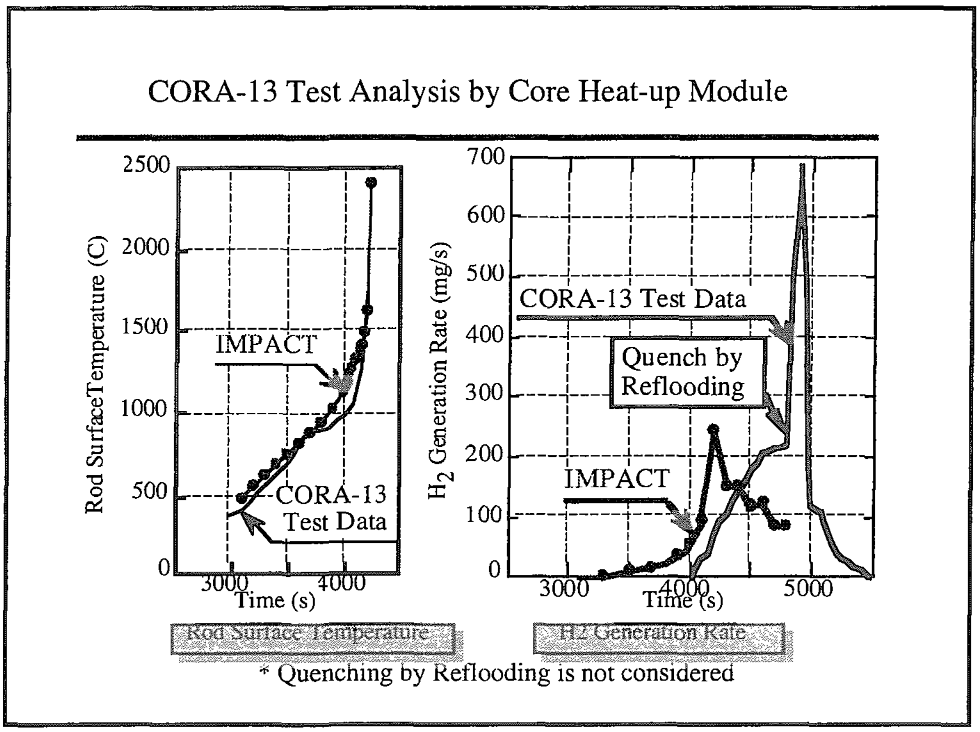

Figure 2 Comparison results of CORA-13 test analysis by the core heat up module

The molten core relocation module solves multi-phase multi-components thermo-hydraulic equations. This module can deal with these number of equations. The left hand side of Figure 3 shows pressure behaviors. The calculation pressure rises faster than that of experiment because of large amount of heat exchange caused by faster fragmentation of molten corium. The discrepancy on this region is caused by neglecting the heat losses in experiment. The right hand side figure shows position of melt front. This model calculates slow penetration speed.

Figure 4 shows calculation results of Kr release compared with experimental data and data obtained by correlation. In experiment, as fuel temperature goes up, Kr release rate is increasing. Both calculation and correlation results of Kr release are nearly the same agreement with experimental data. The Calculation result shows strong influence of fuel temperature change. In this calculation FP release due to grain growth was considered.

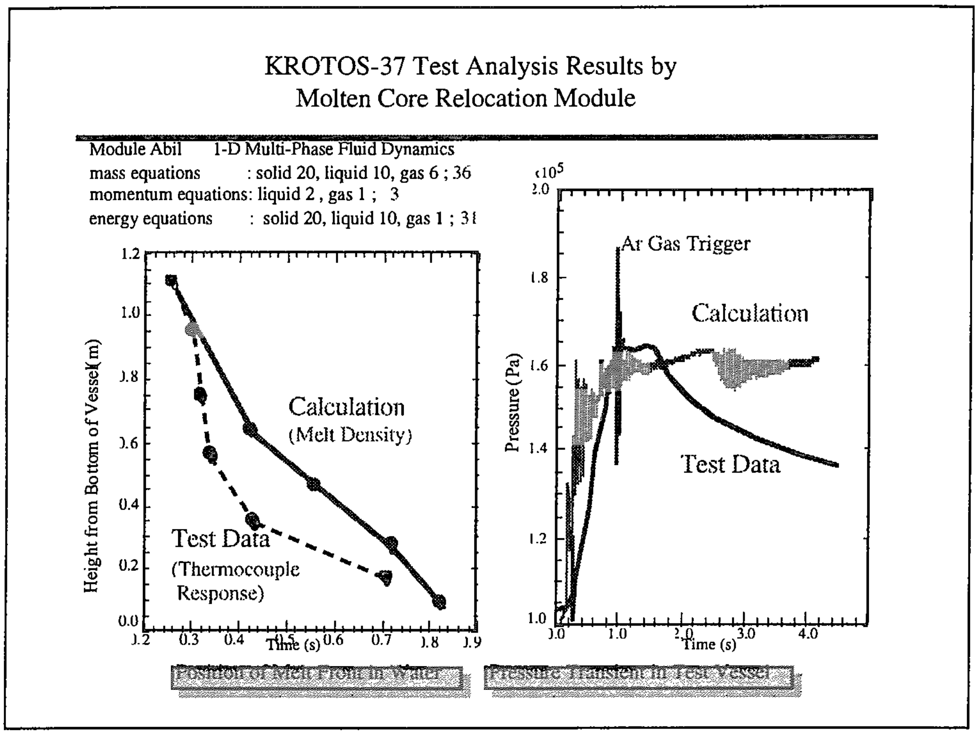

Figure 3 KROTOS-37 Test Analysis by the Molten Core Relocation Module

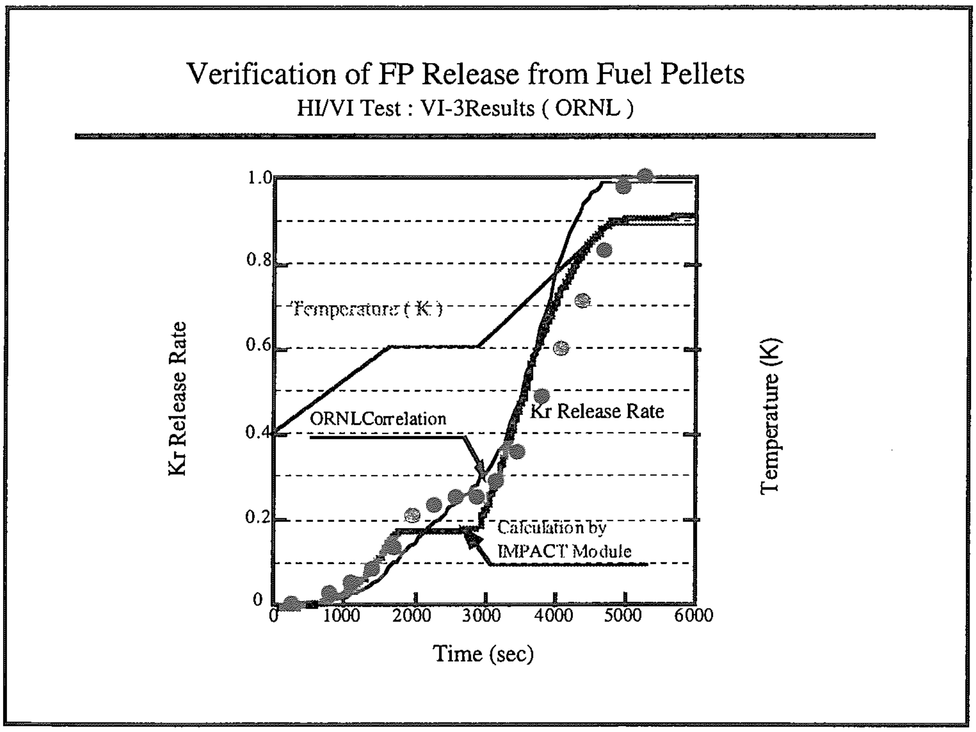

Figure 4 HI/VI Test Analysis

In severe accident, afer core melt ,molten debris go down to the lower plenum. And then spread and cooled. For debris spreading two dimensional Cartesian Nabier-Stokes equations are solved. And we have two type of models for debris cooling, 2D natural convection model and Simplified bulk model. For debris bed cooling we use 1D Lipinski model, and 3D heat conduction model for reactor vessel heating. Creep rupture is evaluated by Larson Miller parameter. Both spreading and cooling models were examined by test data.

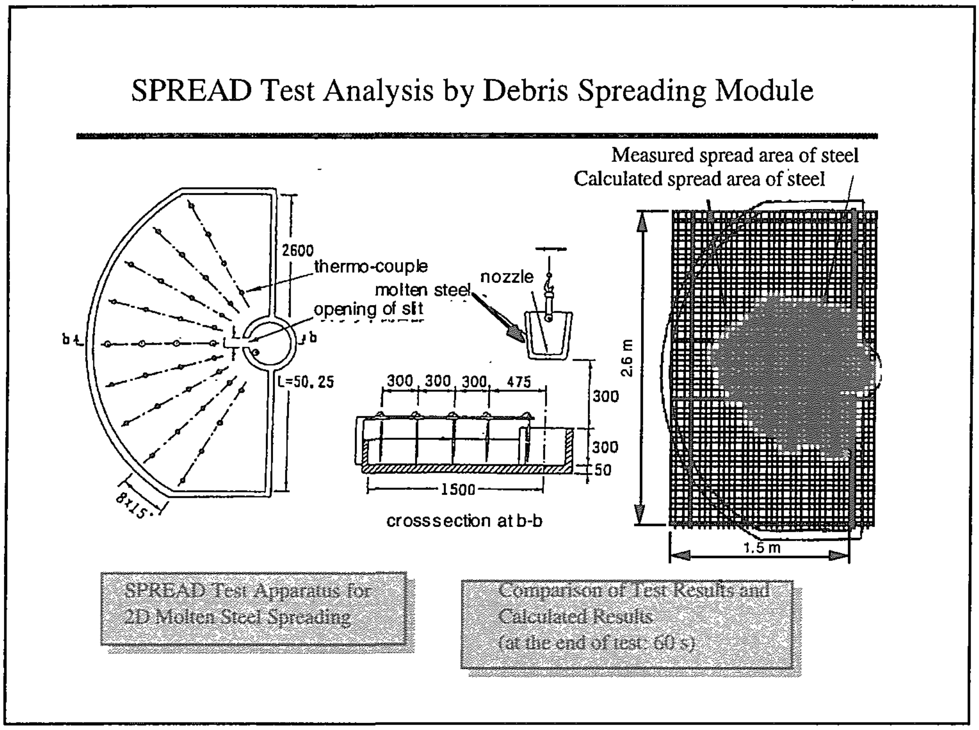

Figure 5 the SPREAD Test Analysis by Debris Spreading Module

The Spreading model was tested by the SPREAD TEST date. Figure 5 shows test sections. Molten steel falls down on this point and then spread through the opening slit. Calculation result shown on this figure with experimental data. The Calculated Spreading area and distance intend to be larger than that of experiment. But differences are not so large.

In case of pressure vessel rupture occurred, debris concrete interaction is need to be evaluated. The SWISS-2 test used for verification of the debris concrete interaction analysis modul. In the teat, Sus304 was heated by induction heating, and the top surface was cooled by water. Figure 6 shows the results. Ablation depth and top surface heat flux were compared with experimental data. Up to 30 minutes calculation result shows good agreement with experimental data, but after that it shows under estimation. And not only this, there is another difference. Experimental data go up exponentially after 30 minutes, while calculation results increase linearly. So ablation model needs to be modified. About heat flux , except immediate after water injection region calculation result shows good agreement with experimental data. The difference may be caused by subcooling effect.

Fig.7 shows the verification results of fine fragmentation propagation model in steam explosion module. The explosion initiated at an elevation of 0.15 m, as in the experiment and the prototype code correctly calculated propagation upward.

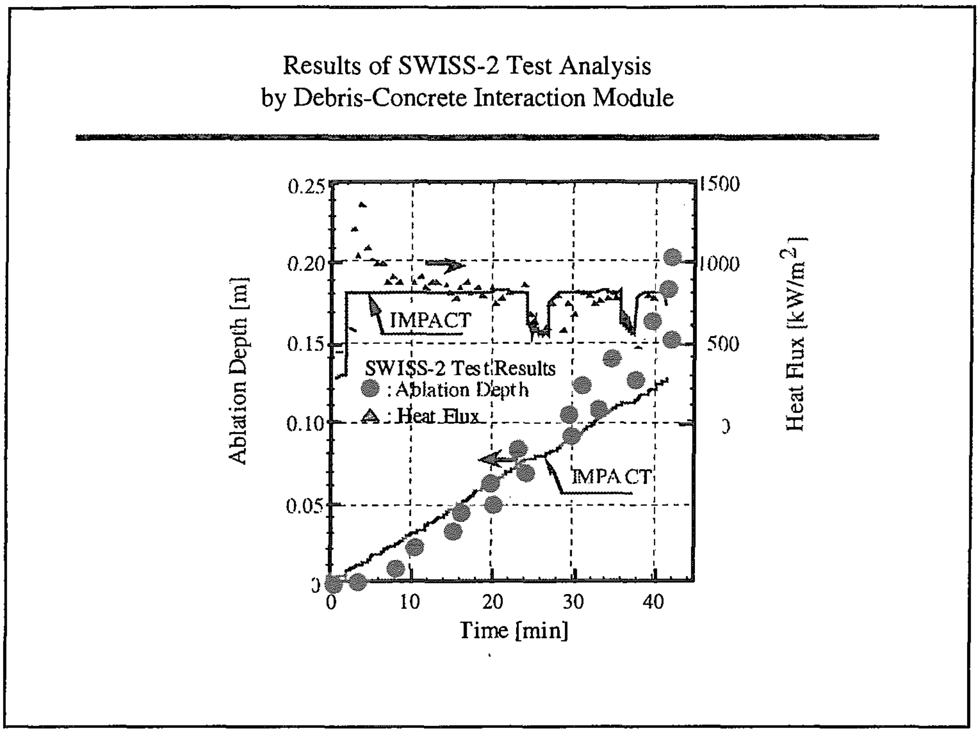

Figure 6 the SWISS-2 Test Analysis by Debris Concrete Interaction Module

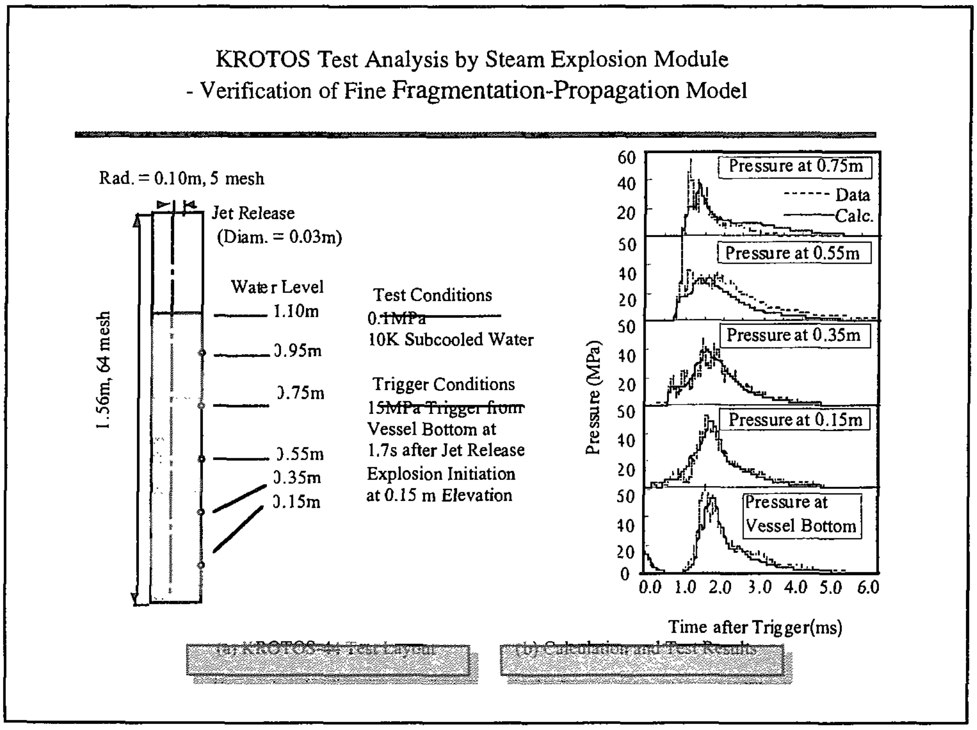

Figure 7 the KROTOS Test Analysis by Steam Explosion Module

III. IN-VESSEL SCENARIO TEST

For the initial quasi-steady state period, the SSS activated the Thermal Hydraulics in primary system Analysis Module and the Fuel rod Heat-up Analysis Module. Normal and transient thermal hydraulics results obtained here were same as usual transient analysis. Upon transient initiation, the FP Release from Fuel Analysis Module and the Thermal Hydraulics in Containment Analysis Module were called. Additional modules were activated and others terminated according to the change in event. The analysis will be terminated when vessel wall temperature decreases and becomes steady state.

1. Core Melt

After the cladding failure at 715s, core melt and relocation behavior was analyzed until 1400s with FRHA and MCRA. The cladding maximum temperature increases to 1500K, upon which the Steam-Zircaloy reaction begins, and then rapidly increases to exceed the fuel failure temperature condition of 2100K.

The fuel melt behavior is shown in Fig. 8. The fuel region is divided into 5 channels and 10 nodes. The second fuel node from the top (node 9) in center channel (channel 1) was melted first at 1114s and then the molten region spread to surrounding nodes. Fuels in Channel 3 were not melted until 1250s while ones in channel 4 has already started to melt, because steam velocity in channel 3 was high compared with other nodes in channel 4 due to the effect of 1 dimensional model (cross flow among channels can not be simulated) and cooling was very effective while fuel temperatures were high. A total of 23 fuel nodes were melted (melt fraction was 1/3), while the fuel in Channel 5 is still intact at 1400s. Pressure in the vessel decreased slowly, and spike occurred after 1114s due to fuel failure.

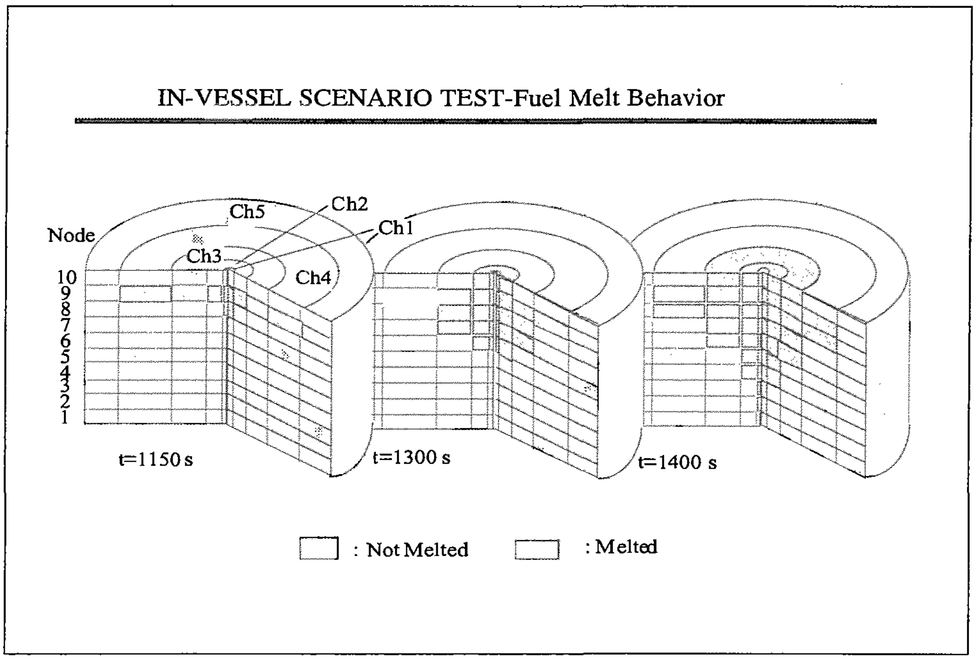

Figure 8 In-Vessel scenario test, Fuel melt behavior

Melt and particle fraction became large when fuel melted and fell from the upper region nodes, and the vapor fraction increased when the node itself melted and fell into lower plenum. Volumetric flow rates of melt content from channel 2 into lower plenum major parts are of fuel particle in this case due to the fuel failure criteria modeled. Total 22 ton of melt have been fallen at time 1400s, over one third of the mass was fallen from Channel 3.

2. Debris Cooling

When the molten debris falls into lower plenum for the first time, debris-spreading model starts and the falling is stopped then natural convection model starts. The time interval of debris falling continued 130s and total 2.0*10^4kg of molten debris was piled on the lower plenum. Therefore, debris-spreading analysis continues to 130s, when steam gap cooling was effective, and then debris-cooling analysis by natural convection model, to 3600s, when gap cooling was effective since water remained in the bottom of vessel. Vessel walls temperature increased while spreading and then decreased while natural convection cooling as shown in Fig. 9.

Outer region was solidified first and crust was formed, because the cooling water was available at the top of molten fuel and between crust and vessel side-wall. Firstly, the debris was solidified due to cooling by the water covering the upper surface and in the gap. Next, solidification advanced upon the inner debris pool by heat transfer at the melt front. After all the debris was solidified, the heat conductivity calculation of the debris was continued. The RPV wall temperature decreased near the fluid temperature at 300s due to cooling by the gap water because it was assumed that the temperature of water in the gap was limited to the saturation point. The removal heat at the gap was added to the water over the upper surface in the models. The vessel wall temperature decreased rapidly compared with molten debris temperature to the level of cooling water temperature at 300s and stayed in steady state till 3600s by gap cooling, as shown in Fig. 9. The in-vessel retention was achieved in this case scenario.

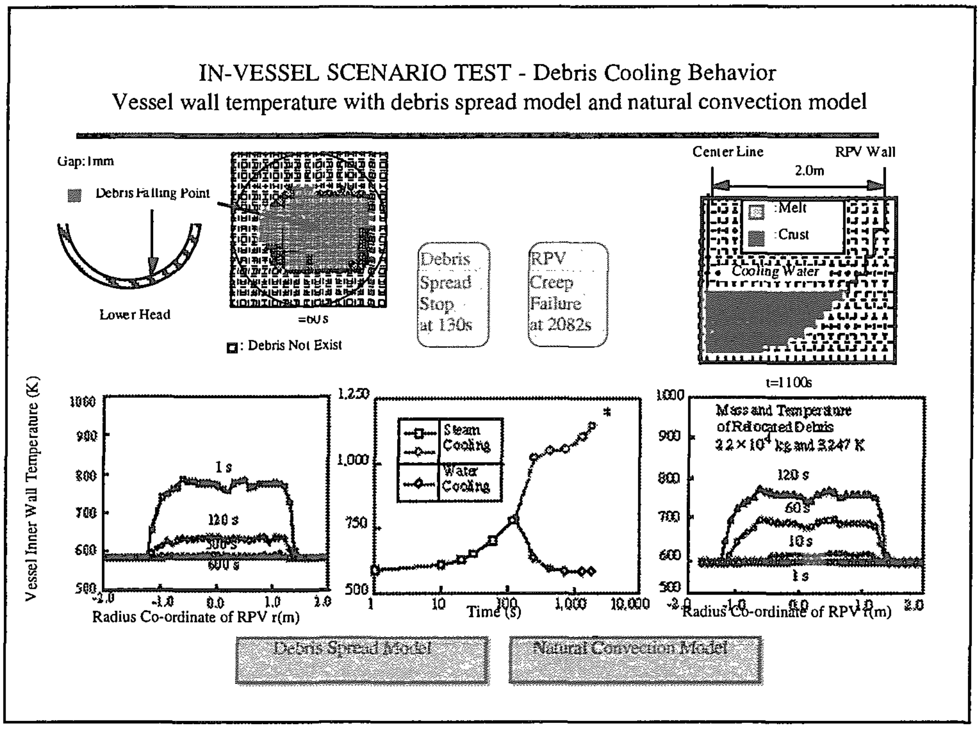

Figure 9 Vessel wall temperature with natural convection model

IV. EX-VESSEL SCENARIO TEST

The fuel heat up, core melt and relocation, and debris cooling phenomena were also analyzed in the ex-vessel scenario test as same as in the in-vessel scenario test. The system integrated analysis continued through RPV failure to long term debris concrete interaction in the containment vessel, upon which the Simulation Supervisory System ended the calculation 1h after the debris concrete interaction.

1. Reactor Pressure Vessel Failure

After spreading is stopped at 130s, the natural convection model analyzes debris-cooling behavior. Outer region was solidified first and crust was formed, because the cooling water was available at the top of molten fuel and steams existed between crust and vessel side wall. Melt pool and crust regions are separated after 300s and then the state is maintained. In the scenario, cooling water was not available and the capability of steam cooling in gap between crust and vessel side wall was lower than water cooling at the gap cooling model, then vessel wall temperature increased as shown in Fig. 9, and finally vessel creep failure occurred at lower part of vessel of the highest temperature at 2082s.

2. Debris Spreading

When the vessel fails and the molten debris falls into primary containment vessel floor at the first time, Debris Spreading Module starts and the spreading is stopped then Debris Concrete Interaction Module starts. Spreading speed was rather high (spreads 1.8m at 30s) than in pressure vessel bottom because floor is flat. Concrete floor temperature increases according to debris spreads.

3. Debris Concrete Interaction

One dimensional analysis system of vertical dimension is 2m water depth, .23m debris thickness and 5m limestone concrete thickness with 3.2m wide, and horizontally 3.2m thick concrete. Ablation depth of lower concrete by debris was 8.2cm and side, 4.7cm respectively at 60min, because the lower direction heat transfer coefficient between molten debris and concrete floor is higher than side direction one between debris and concrete wall. Both upward heat flux from debris and its temperature decreased as ablation proceeded, to 160kW/m^2 and 2010K respectively at 60min. Flammable gas release rate increased according to the ablation rate due to concrete decomposition and gas interaction with molten debris, and the final amounts of CO and H2 were 200kg and 11kg respectively until 60min after the molten debris fell onto the floor.

The system verification test has confirmed that the full scope of the scenarios can be analyzed and phenomena occurred in scenarios can be simulated qualitatively reasonably considering the physical models used for the situation. In-vessel scenario test has shown the possibility to evaluate in vessel retention, while Ex-vessel scenario test the ability to calculate whole scope of severe accident.

VII. CONCLUSIONS

Four years of the IMPACT project Phase 1 have been completed with financial sponsorship from the Japanese government's Ministry of International Trade and Industry. At the end of the phase, demonstration simulations by combinations of up to 11 analysis modules developed for the SAMPSON Code were performed and verified.

The verification study of the SAMPSON was conducted in two steps. First, each analysis module was run independently and analysis results were compared against separate-effect experiment data with good agreement. Second, with the Simulation Supervisory System, these analysis modules were executed concurrently in the parallel environment, to demonstrate the code capability and integrity. The target plant was Surry as a typical PWR and the initiation event was a 10-inch cold leg failure. The system analysis is divided to two cases; one is in-vessel retention analysis (In-vessel scenario test), the other is analysis of phenomena when the event is extended to ex-vessel due to the Reactor Pressure Vessel failure (Ex-vessel scenario test). Through test analyses, the Simulation Supervisory System called and terminated analysis modules as appropriate according to the progression of plant phenomena, controlled the parallel processing, and the analysis modules showed qualitatively good responses.

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