Simulation & analysis of the severe accident of the Unit 1 of Fukushima Daiichi NPP

In the framework of the activities devoted to the development of the ENEA-Casaccia "NPP Engineering Simulator", a RELAP/SCDAPSIM code simulation model of the Unit 1 of Fukushima Daiichi NPP was developed and applied for the analysis of the severe accident occurred on March 11, 2011 in Japan. The model accurately reproduces the BWR-3 primary system and the Mark I containment. BWR-3 NPP public available data of a similar unit were used for setting up the model and performing its steady state and transient validation. Main events' reconstruction of the Fukushima scenario was based on the official Japanese data. The first 24 hours of the accident were simulated, beginning with the reactor scram as a consequence of the earthquake, and reproducing the behavior of the main engineered safety features. Results showed that the core uncovering and degradation began at +2 hours after the tsunami wave hit the plant. Core melting was predicted to occur in the subsequent 6 hours, with a fuel relocation at the bottom of reactor pressure vessel. RELAP/SCDAPSIM code models calculated the severe damage of the reactor boundary allowing to estimate the time of the consequent containment over-pressurization, that resulted well beyond the design limits. The calculated results are consistent with the ones performed by different Japanese Institutions and, together with the developed model, constitute valuable tools for ENEA in view of its future role as Technical Support Organization for the National Nuclear Safety Authority

Simulazione ed analisi dell'incidente severo dell'Unità 1 della centrale nucleare di Fukushima Daiichi

Nell'ambito delle attività dedicate allo sviluppo del Simulatore Ingegneristico di Impianti Nucleari nel Centro Ricerche ENEA "Casaccia", è stato sviluppato un modello per il codice di calcolo RELAP5/SCDAPSIM dell'Unità 1 dell'impianto nucleare di Fukushima Daiichi (Giappone). Tale modello è in grado di simulare il comportamento termoidraulico del circuito primario (tipo BWR-3) ed del contenimento (tipo Mark I) ed è stato impiegato per eseguire un'analisi dell'incidente severo occorso a tale impianto l'11 marzo 2011. I principali eventi occorsi immediatamente dopo il terremoto sono stati ricostruiti ed implementati. Le prime 24 ore dell'incidente sono state simulate, analizzando il comportamento del combustibile nucleare, del contenimento e dei principali sistemi di sicurezza. I risultati mostrano come il danneggiamento del combustibile sia iniziato dopo circa 2 ore dall'inondazione causata dallo tsunami. La fusione del nocciolo è avvenuta nelle seguenti 6 ore, con una rilocazione della parte fusa nel fondo del recipiente in pressione. Il codice RELAP/SCDAPSIM ha consentito di calcolare il danneggiamento del recipiente in pressione stesso e la conseguente sovrappressione del contenimento causata dal rilascio di massa dal primario. I risultati dei calcoli si sono dimostrati in accordo con le analisi svolte da diverse istituzioni giapponesi. Il modello sviluppato e le analisi effettuate risultano quindi particolarmente rilevanti in vista del futuro ruolo dell'ENEA come Organizzazione di Supporto Tecnica (TSO) per l'Autorità per la Sicurezza Nucleare

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ENEA is in charge of research and development activities in the field of nuclear safety technology. The acquisition of the RELAP5/SCDAPSIM code [1] and the development of a model simulating the severe accident of Unit 1 of the Fukushima Daiichi NPP are tasks devoted to the development of the ENEA-Casaccia "Enhanced NPP Engineering Simulator". The code selected for simulation is based on the well-known RELAP5 thermal-hydraulic code. It can model the overall Reactor Cooling System (RCS) thermal-hydraulic response, including the (zero dimensional) neutron kinetic and the thermo-mechanical behavior of the fuel rod. During the severe accident simulation, the core structures damage progression is modeled by means of the code SCDAP while the code COUPLE calculates the thermo-mechanics interaction between the molten material and the reactor pressure vessel lower head.
This paper reports the thermal-hydraulic modelling of the reactor, engineered safety features and containment respectively. Then, the nodalization qualification process is illustrated. The main events occurred at the Fukushima Daiichi Unit 1 are recalled according to the present knowledge, and the preliminary results of the simulations of the accident are showed. The effects of some significant parameters on the simulation results are also investigated and presented. In the last paragraph, the conclusions and the connections with the ENEA ongoing and future activities are presented.

Model of Fukushima Daiichi Unit 1 Npp

The reactor and the reactor coolant system

Fukushima Daiichi Unit 1 is a BWR-3 reactor, designed by General Electric, equipped with isolation condensers [2], [3], see Figure 1.
The thermal-hydraulic (TH) nodalization models the Reactor Pressure Vessel (RPV), the steam lines, the Safety & Relief Valves (SRV/RV), the Isolation Condensers (IC) and the two recirculation lines including the centrifugal pumps and the jet pumps. The turbine and the turbine bypass are also modelled, as imposed boundary conditions. Active and passive heat structures are used for modelling the nuclear fuel and the structural materials of the RPV and of the reactor coolant system. The active core is divided into five independent zones by means of five TH channels. Moreover, one TH channel is used for modelling the radial reflector and further five independent TH channels are used for modelling the moderator bypass. The latter is associated with the RELAP/SCDAPSIM control rod blade component. General Electric 8x8 fuel assembly (FA) data are used and appropriate peaking factors and axial power shape are imposed. ICs are modelled and connected to the upper part of the downcomer (DC) and to the two recirculation lines. All the SRV/RV are modelled by using opening and closing set points reported in reference [2]. In Figure 2, the nodalization scheme and the core radial scheme are presented.
The overall nodalization is composed of around 1000 hydraulic volumes. The geometrical data for setting up the RPV model and the reactor coolant system was retrieved by public available documentation concerning an identical unit [4].

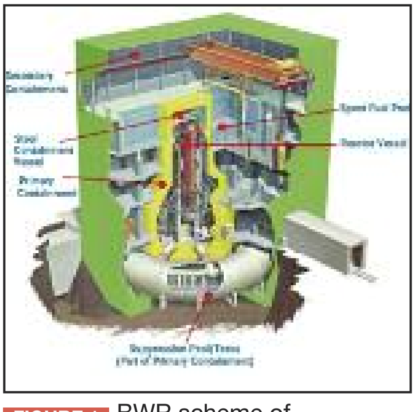

FIGURE 1 BWR scheme of Fukushima Daiichi Unit 1

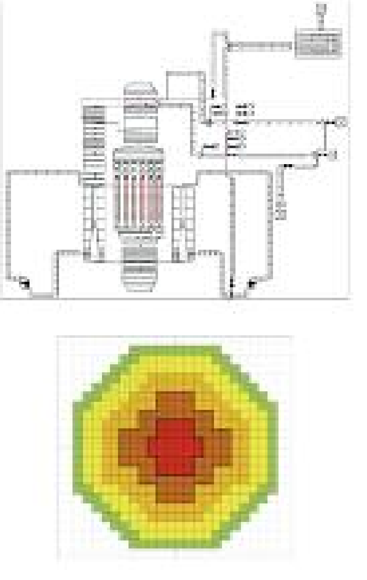

FIGURE 2 RELAP/SCDAPSIM TH nodalization scheme (up) and core modelling (bottom)

The containment

MARK I type containment is present at the Fukushima Daiichi Unit 1. This containment type was modelled by using a nodalization scheme properly set up for simulating the three-dimensional flow paths in big volumes (Figure 3). The bulb-shaped drywell and torus-shaped wetwell are represented with a series of pipes and branches preserving the volumes of the relevant sections. The venting system (header and downcomers), the spargers and the vacuum breakers are also modelled.

The containment nodalization is coupled to the reactor coolant system nodalization via the SRV/RVs, discharging into the wetwell liquid volumes and into the drywell atmosphere.

Model qualification

A steady-state qualification was achieved by running a null transient and by verifying the main plant parameters. BWR-3 steady state data of a similar unit were used [4]. Results are showed in Table 1.

The following step was the "on-transient" qualification. This was performed by reproducing a turbine trip event occurred in a similar unit [5] in 1992 and by checking some of the main parameters. Results are reported in Figure 4 and Figure 5, showing the trends of the reactor pressure dome, the core and the recirculation lines mass flows. A qualitative agreement is demonstrated by such calculations notwithstanding an imperfect knowledge of the imposed sequence of main events (i.e., the detailed operation of the steam dump system is not available).

The Fukushima Daiichi Unit 1 accident simulation

The main sequence of events of the accident at Unit 1 (Table 2) was reconstructed by using Japanese official documentation [2], and then used for the numerical simulation.

The main assumption is that no reactor cooling was achieved anymore after the tsunami flooding, because of the total station blackout and the loss of the passive cooling by the ICs. Some reactor cooling was re-established by using the core sprays several hours later. Containment venting was imposed after roughly 24 hours from the beginning of the events. Simulation was stopped

Parameter	Plant parameter	RELAP5/ SCDAPSIM	Error (%)
Core Thermal Power (W)	1.38E+09	1.38E+09	*N/A*
RPV dome pressure (MPa)	6.98	7.02	*0.54*
Total mass flow (kg/s)	5622	5605	*-0.30*
Bypass flow (kg/s)	—	341	*N/A*
Recirculation line mass flow (kg/s)	1308	1311	*0.23*
Steam Lines total mass flow (kg/s)	685.7	685.0	*-0.10*
Reactor Level (m above the TAF)	4.109	4.163	*1.31*
FW mass flow (kg/s)	677.5	686.0	*1.25*
FW Temperature (K)	452	452	*N/A*

TABLE 1 Steady state qualification – significant parameters

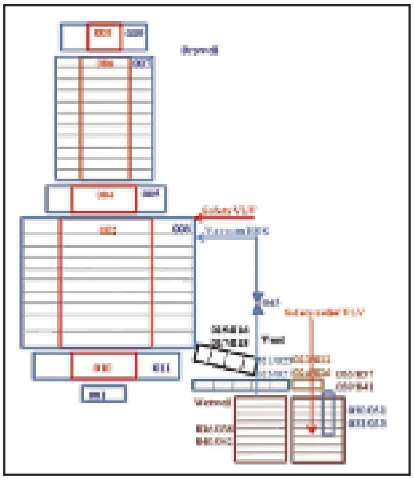

FIGURE 3 RELAP/SCDAPSIM TH nodalization of the MARK I containment

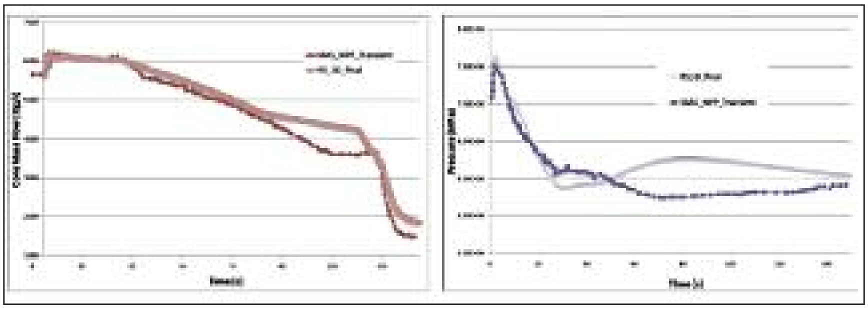

FIGURE 4 On-transient qualification: Core Mass flow (kg/s, left), Steam Dome pressure (MPa, right)

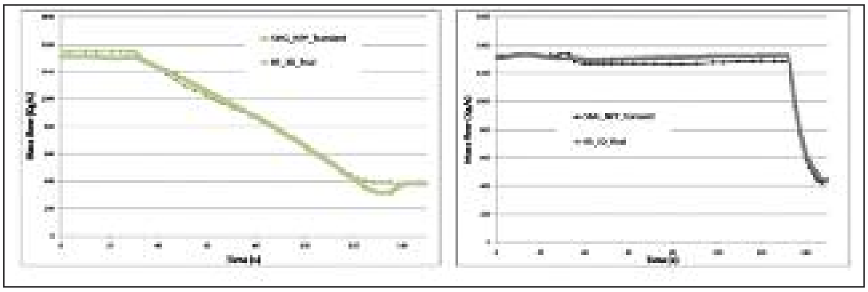

FIGURE 5 On-transient qualification: Recirculation line mass flows (kg/s), line A (left), line B (right)

Main Event	Time (s)
Earthquake, Reactor Scram, Turbine Stop Valve closure	0.
Bypass valve opening	+30.
Bypass valve closure, MSIV closure and Reactor Isolation	+60.
IC opening	+360.
IC closure	+1060.
IC openings/closures (3 times)	from +1860. to +2880.
TSUNAMI flooding	+3060.
Fresh water injection by core spray (2.4 kg/sec)	+54000.
Containment venting	+85440.
End of Containment venting	+86580.
Stop of fresh water injection [*end of simulation*]	+86820.

TABLE 2 Main events simulations time

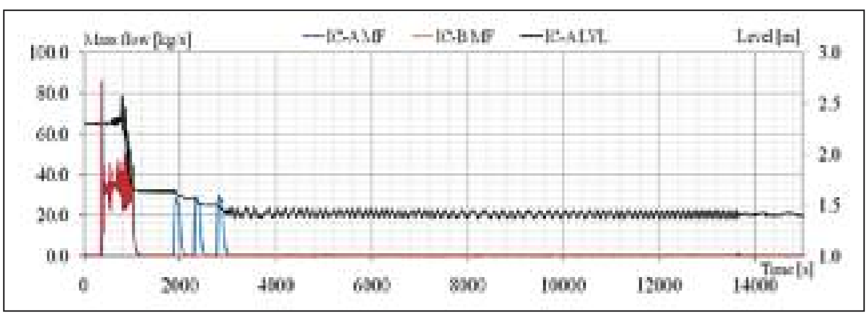

FIGURE 6 Isolation Condensers A&B Mass Flows (kg/s) and IC-A Level (m)

immediately after, because of the hydrogen explosion that occurred into the reactor building.

Seven phases of the accident were identified and simulated:

- Phase 1: scram, by-pass pressure control, reactor isolation by main steam isolation valves (MSIV) closure [0-60 secs]
- Phase 2: energy removal by the ICs [+360 to 3060 secs]
- Phase 3: End of cooling (stop of ICs), loss of RPV inventory, water level decreasing up to the Top of Active Fuel (TAF) [~ 3060 secs to +7000 secs (2hr)]
- Phase 4: Core uncovery and degradation, H_2 formation [from +2 hr to +3.4/4 hrs]
- Phase 5: Core melting [+3.4/4 hrs to 3.8/8 hrs]
- Phase 6 : RPV bottom damage and break [+3.8/8 hrs to +15 hrs]
- Phase 7: Containment overpressurization and venting [+15 hrs to +24 hrs].

The results of the reference calculation are showed in the figures below. The IC-A level and ICs mass flows are showed in Figure 6. These passive systems were actuated for removing the core decay heat during the first phases of the accident. The reference simulation was conservatively supposed to completely cease the operation of the isolation condenser systems after the tsunami wave. It should be noted that this is a point that is still being discussed by the Japanese authorities.

Following the tsunami, the RPV level decreased, reaching the Top of Active Fuel (TAF) in 1 hour (onset of core uncovery), because of the loss of cooling by ICs and mass inventory released via the SRV towards the containement. Figure 7 provides the timing of the water level in the reactor pressure vessel downcomer and shroud sides, and the maximum cladding temperature calculated in the core. About 10 minutes later the collapsed level drops below the TAF, the cladding temperature starts to rise slowly. After 3h5min the caldding temperature exceeds 1200 °C. Then, few minutes later the power of the steam zirconium interaction becomes predominant compared to the de-

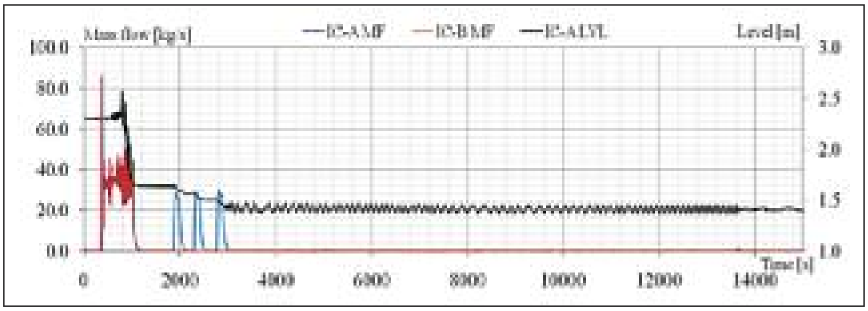

FIGURE 7 RPV DC and in-shroud level (m), Hot spot clad temperature (°C)

cay power, and the rate of cladding temperature increases drastically. The first fuel melting temperature is calculated in the upper part of the core after about 3h20min.

RPV pressure is kept constant by the actuation of the SRV that, on the other hand, causes a slow increase in the containment pressure (Figure 8). A large drop into the RPV pressure was registered by the NPP instrumentation [6] and it was modelled in the simulation by imposing a RPV lower head break (Figure 8). This loss of RPV integrity is supposed to be caused by the degradation effects caused by the fuel slumping in the bottom part of the lower plenum and it is calculated by the COUPLE module of the RELAP/SCDAPSIM. Consequently, because of the energy released into the containment by the lower plenum break and because of the H_2 releases, the containment pressure spikes around 0.8 MPa. The COUPLE module then correctly predicts the energy transfer between the molten fuel still kept into the lower head and the containment atmosphere (Figure 9). The following pressure drop, several hours later, is obtained by simulating the containment venting [6]. Such procedure led to the hydrogen explosion into the reactor building in the real NPP.

Several sensitivities were run in order to assess the effect of the different code models on the transient evolution. One example of such sensitivities is showed in Figure 10, where different models were applied for bounding or determining when a pool of molten material in the core region slumps to the lower head. In particular, on the left part of the picture, the degradation sequence led to the complete core slumping in 7.2 hours, while, on the right part, the degradation sequence resulted in a major core melting in 8.3 hours.

A sensitivity analysis was also run for taking into account the loss of mass inventory through the pump seals (~25 gallons per minute were imposed, [7]), resulting in an anticipation of the core degradation by half an hour. Calculation of the total hydrogen production resulted in roughly 450 kg. Sensitivities were performed on

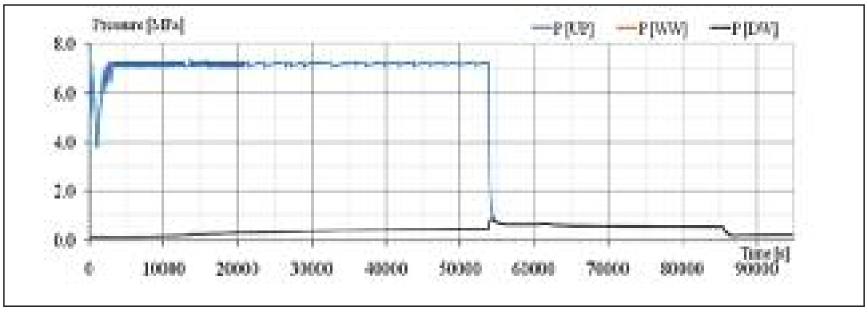

FIGURE 8 RPV and Drywell pressures (MPa)

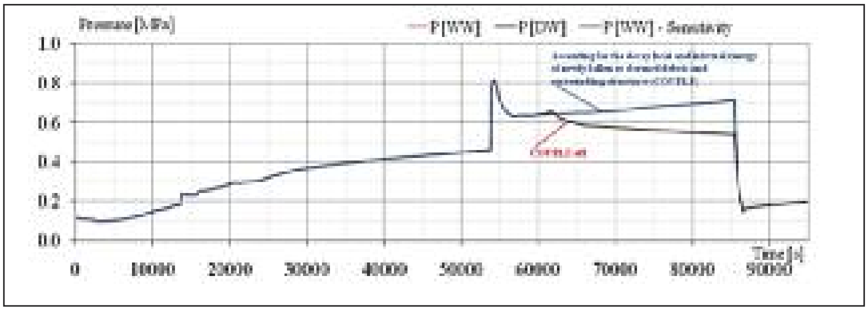

FIGURE 9 Containment pressure (MPa) – COUPLE on/off model sensitivity

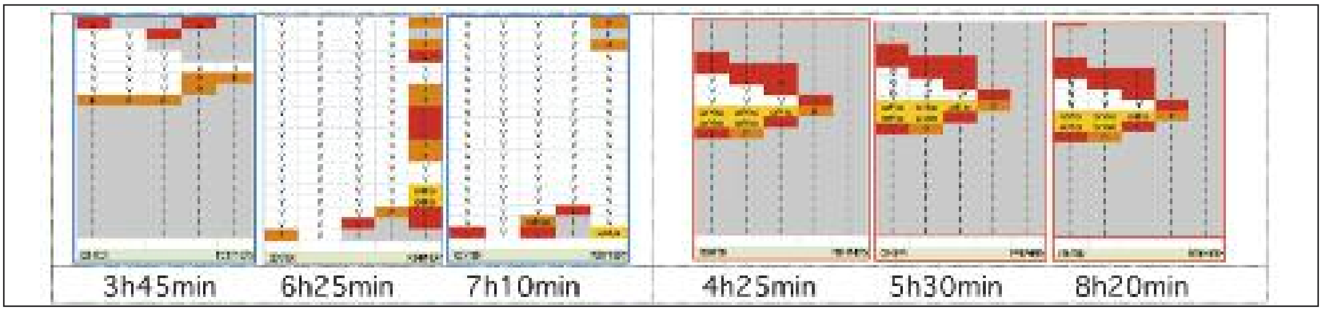

FIGURE 10 Core degradation snapshots. Earliest (left) and latest (right) damage progressions

Event	ENEA – R5/SCDAPSIM	NISA - MELCOR	TEPCO - MAAP
Core Exposure	2 hrs	2 hrs	3 hrs
Core Damage	3 hrs	3 hrs	4 hrs

TABLE 3 Comparison of the main events simulated by different institutions

the steam starvation model, on the oxidation extent for assessing the durability of the cladding oxide shell, and on the application of the COUPLE model.
The most relevant events of accident progression are presented in Table 3. The results of the reference simulation are compared with the analyses performed by the Japanese institutions [6] using independent severe accident codes like MAAP[8] and MELCOR [9].

Conclusions

A numerical model of Unit 1 of Fukushima Daiichi NPP was developed and applied for the simulation of the severe accident occurred on March 11, 2011 in Japan. The analysis was performed by modelling the complete RPV, the reactor coolant system, the engineering safety features and the containment. Steady state and transient validation were obtained by using available data of a twin unit. The severe accident analysis and the sensitivities calculations performed allowed to identify the main occurred phenomena, to bound the timing of the accident progression and to evaluate the amount of hydrogen generated during the core degradation. The time sequence of the events was similar to the one calculated by the Japanese institutions using different codes and methods. The preliminary comparison of the main calculated values with some plant data also showed some degree of agreement. Follow-up activities will be needed in the future in order to improve the numerical model capabilities, to investigate possible countermeasures in similar postulated accident sequences, to identify and address the fission product release and transport and to test new code models.
The continuation of such work will be part of the efforts being performed at ENEA "Casaccia" Research Center for reconstructing the nuclear simulation competences and rebuilding an Enhanced Engineering Simulator.

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