Annals of Nuclear Energy 123 (2019) 59–77

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

Updated analysis of Fukushima unit 3 with MELCOR 2.1. Part 1: Thermal-hydraulic analysis

L. Fernandez-Moguel *, A. Rydl, T. Lind

Paul Scherrer Institute, OHSA C02, Villigen PSI 5232, Switzerland

ARTICLE INFO

Article history:
Received 27 June 2018
Received in revised form 20 August 2018
Accepted 7 September 2018

Keywords:
Fukushima
Plausible scenario
Modelling
Unit 3
Core degradation

ABSTRACT

The accident sequence of Fukushima Unit 3 was analysed with MELCOR 2.1. Hundreds of calculations have been performed and three plausible scenarios which predicted remarkably well the main signatures (i.e. pressure in RPV and containment and containment water level) have been selected and studied in the present analysis.

The main signatures, namely pressure in the RPV and drywell and water levels were satisfactorily reproduced by the three proposed plausible scenarios. Major uncertainties concerning the time of RPV failure, possibility of MCCI as well as the transport of hydrogen responsible for the explosion of unit 3 and unit 4 were addressed in this study. The study was complemented with plant observations such as containment inspections, muon measurements and photographs taken during the accident. The results point out that the plausible sequence lie between case 1 and case 2 presented in this study.

The state of the core after 350 h seems to be in agreement with our case 2, where ca. 64 tons of debris and stainless steel structures remained in the lower head and ca. 70 tons of debris were ejected to the pedestal. Taking into account the limitation in the modelling of the failure modes of a BWR, this estimation is very uncertain. Further discussion will be provided in the second part of the paper where the fission product release and transport will be studied and compared against available source term calculations.

1. Introduction

Since the Fukushima Daiichi nuclear accident on 11 March 2011, caused primarily by the tsunami following the Tōhoku earthquake, there have been significant efforts to understand the accident sequence as well as the extent of damage for units U1-U3. One example is the OECD-NEA NEA Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station (BSAF) project, aimed to evaluate and analyse the likely end-state of the reactor core after the accidents at the Fukushima Daiichi nuclear power stations (OECD/NEA/CSNI, 2014). PSI is a member of the project; however the results provided in the present paper only describe the individual contribution of PSI to the BSAF project and they are solely focused on U3 sequence.

The extensive studies by several institutes (Gauntt et al., 2012; EPRI, 2013; Naitoh et al., 2013; Bonneville and Luciani, 2014; Cardoni et al., 2014; Pellegrini et al., 2014; Robb et al., 2014; Yamanaka et al., 2014; Fernandez-Moguel and Birchley, 2015; Sevón, 2015; EPRI, 2015; Sonnenkalb and Band, 2015; Bratfisch et al., 2015; Bonneville et al., 2015; Herranz et al., 2015a; Naitoh et al., 2015; Luxat et al., 2015; Pellegrini et al., 2016; Luxat et al., 2016; Nagase et al., 2016), together with the reactor building and containment inspections have helped to better understand the accident progression as well as the different phenomena which have taken place during the accident. Nevertheless, there are still uncertainties such as time and mode of reactor pressure vessel (RPV) failure, mode and fraction of debris ejected to reactor vessel support pedestal floor, extent of molten core concrete interaction (MCCI), etc. This is especially true for U3 where the predicted time of the main failure events vary widely: in some plausible sequences the RPV lower head was predicted to have remained intact (Pellegrini et al., 2016), whereas in (Fernandez-Moguel and Birchley, 2015; Sevón, 2015; Sonnenkalb and Band, 2015; Bratfisch et al., 2015; Naitoh et al., 2015; Pellegrini et al., 2016;

Abbreviations: AWI, Alternative Water Injection; BAF, Bottom of Active Fuel; BSAF, Benchmark Study of the Accident at the Fukushima Daiichi Nuclear Power Station; CST, Condensate Storage Tank; COR, Core package in MELCOR; CV, Control Volume (i.e. used for hydrodynamic nodalization in MELCOR); FP, Fission product; HPCI, High Pressure Coolant Injection; LH, Lower Head; MCCI, Molten Core Concrete Interaction; PCV, Primary Containment Vessel; RCIC, Reactor Core Isolation Cooling; RPV, Reactor Pressure Vessel; ROV, remote Operated Vehicle; SRV, Safety Relief Valves; TAF, Top of Active Fuel; U3, Unit 3; U4, Unit 4.

* Corresponding author.
E-mail address: Leticia.Fernandez-Moguel@psi.ch (L. Fernandez-Moguel).

https://doi.org/10.1016/j.anucene.2018.09.008
0306-4549/

and Luxat et al., 2016) the RPV lower head failure prediction was predicted between 45 and 96 h.

The inspection inside the U3 containment in October 2015 showed that the water level in the drywell is ca. 6.3 m above the containment floor, pointing out that there may be no significant leakage at the bottom of the drywell and the possibility of containment liner failure in U3 would seem unlikely (TEPCO, 2015). In July 2017 inspections inside the pedestal of U3 were made with a remotely operated underwater vehicle (ROV). The images captured by the ROV identified damage to multiple structures, and solidified melted structures from the RPV and/or the containment which relocated from their original position in the area inside the pedestal. Some of the observed structures are believed to be reactor internals (TEPCO, 2017a). Additionally, Muon measurements have been carried out at U3 from May to September 2017 and an attempt to quantify the debris distribution in the RPV has been made (TEPCO, 2017b). The large uncertainties in the measurement make the estimation only tentative, nonetheless the results point out that there are some debris remaining in the core area as part of the fallen debris remained in the lower head of the RPV and a fraction of the debris was expelled to the pedestal.

Based on the observations above, it is confirmed that some amount of debris was expelled from the lower head to the pedestal. However, the amount of debris discharged from the RPV to the pedestal is uncertain. Even more uncertain is the amount of debris which may have reached the floor of the pedestal in molten form and has participated in molten core concrete interaction. The possibility of water present in U3 pedestal at the time of RPV failure makes the estimation of MCCI uncertain and a challenge for the severe accident codes used in the estimation of the accident progression.

The RPV lower head failure time as well as the transfer mode of the melt from RPV to the pedestal are among of the main uncertainties of the sequence. Therefore, especial attention was given in the present study to identify the most likely time of its occurrence and which mechanisms could have caused it.

As described in (Sehgal and Bechta, 2016) most of the current knowledge on severe accident progression in the reactor vessel lower plenum is based on the knowledge from a pressurized water reactor (PWR), to a large extent from the accident in Three Mile Island (TMI-2). In a PWR the melt would likely be expelled from the RPV with a global lower head failure and all the melt would leave the RPV relatively fast. BWR is different from PWR since the reactor vessel lower plenum has a large number of penetrations for control rod guide tubes (CRGT) and for instrument guide tubes (IGTs). CRGTs and IGTs traverse the thick vessel bottom wall and are welded to the reactor vessel. These structures would have a large impact on the accident progression. The first mode of failure of a BWR is believed to be one or several of the penetrations. The seals of the penetration would fail first causing a leakage of water from the reactor vessel (Sehgal and Bechta, 2016). Release of debris through the failed penetration seal should be small.

The existing experiments which have addressed the mechanism failure of a BWR lower head are limited. One example of such experiments is a melting test of a full scale Short Range Monitor/Intermediate Range Monitor (SRM/IRM) guide tube used in a BWR, reported in (Naitoh et al., 2018), similar to the those present in the Fukushima units. The experiment showed that only a small amount of corium was relocated from the bottom opening of the tube into the pedestal as particles, and such outflow stopped soon due to the re-solidification of the stainless steel melt in the tube which caused the flow path to be blocked. In the test, the time of induction heating to the lower crucible was limited to only about 23 min. At the actual plant, the high temperature corium might stay longer in the lower plenum resulting in heating the RPV bottom wall. There was not internal heating of the debris particles; on the other hand the experiments were performed in the absence of water. Water could contribute even further to the re-solidification of the melt as may have been the case in Fukushima U3 where water was being injected in several occasions to the RPV.

After the lower head of an RPV fails, it is expected that the melt would drop to the bottom of the containment and Molten core-concrete interaction (MCCI) would start. However, this process may be stopped or completely suppressed if water is present in the containment, or if water is injected to the melt. The experimental data for the effect of water on MCCI progression are limited.

The OECD/MCCI-2 project (Lomperski et al., 2010) studied the effect of crack formation on the upper crust of a molten corium on the coolability. In particular they studied the cooling effect by the water ingression on the top of the crusted melt. Water injection from bottom of the melt layer is a more direct and probably more effective measure to reach coolability. This has been studied on the COMET concept for ex-vessel core melt retention (Widmann et al., 2006). In those experiments it was observed that the water injection at the bottom would promote rapid fragmentation of the corium, porosity formation and thus coolability.

Another strategy for melt coolability is to have a pre-existing flooded cavity. The SARNET2 European project on ex-vessel debris formation and coolability (Pohlner et al., 2014) studied the coolability of melt released into a water-filled cavity. Analyses of KTH Stockholm on vessel failure modes and timing for BWR point to a most likely failure of the control rod and instrumentation guide tubes located in the vessel bottom. This may lead to a dripping outflow mode for the melt falling from the vessel into the water filled cavity, with good chances to form there a coolable debris bed. For deeper water pools and not too high melt mass outflow rates, the formation of a fragmented debris bed could be demonstrated. The coolability of such debris beds was dependent on several parameters, including particle diameters, the porosity of the bed, geometry, decay heat, etc. On the one hand, if scenarios have to be envisioned with relatively fast melt release with large jet diameters and large mass flow rates, there is also a dependence of the bed cooling on the initial conditions, i.e. initial temperature and solidification state of the settling particles. If melt is released in dripping mode which can last for few thousand seconds, initial temperature of the debris is less important due to rather rapid quenching of initially dry debris bed. Even though, having a pre-existing flooded cavity may have not been intended in Fukushima U3, it is possible that water was present in the cavity before the melt reached the floor of the pedestal. However, It is uncertain when and how much water where present at the moment of the melt ejection. Some of the mentioned processes may have taken place during the accident of U3. In Section 4.2 additional discussions will be provided as well as the possible implications for additional hydrogen generation which will be described in Section 5.

This paper aims to present plausible accident sequences for U3. The input deck from (Fernandez-Moguel and Birchley, 2015) was updated with new available information from TEPCO and the BSAF project. The sequence was extended to 15 days after the SCRAM. An estimation of the source term was performed. The paper is divided in two parts. Part 1 is focused on the thermal-hydraulic behaviour of the accident transient with the presentation of the most probable accident sequence, and part 2 will be dedicated to the fission product release analysis. For this first part, a description of the main changes made to the model with Melcor 2.1 as well as the boundary conditions is provided in Section 2. Section 3 presents the selected plausible sequences and identifies areas of uncertainty in the transient. Section 4 addresses the uncertainties in the sequence as well as code limitations: Section 4.1 is devoted to identify the most likely time when the RPV lower head failure took place; Section 4.2 is devoted to the possibility of MCCI;