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A recent version of MELCOR for fusion safety applications

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ABSTRACT

During the engineering design activity (EDA) of ITER, the MELCOR 1.8.2 code was selected as one of several codes to be used to perform ITER safety analyses. MELCOR was chosen because it has the capabilities of predicting thermal–hydraulic transients and self-consistently accounting for aerosol transport in nuclear facilities and reactor cooling systems. The Idaho National Laboratory (INL) Fusion Safety Program (FSP) organization made fusion specific modifications to the MELCOR 1.8.2 code that allows MELCOR to assess the thermal–hydraulic response of fusion reactor cooling systems and the transport of radionuclides as aerosols during accident conditions. The ITER International Organization (IO) used this version of MELCOR to perform accident analyses for ITER's "Rapport Préliminaire de Sûreté" (Report Preliminary on Safety – RPrS). Because MELCOR has undergone many improvements since version 1.8.2 was released, the INL FSP introduced these same fusion modifications into MELCOR 1.8.6, and thereby produced a version of MELCOR 1.8.6 with similar capabilities to the version of MELCOR used by the ITER IO for the ITER RPrS. We have applied this recent version of MELCOR to the analysis of a large in-vessel water leak event examined in the ITER Generic Site Safety Report (GSSR). This paper presents the results of this analysis and compares these results to those obtained from the MELCOR 1.8.2 code.

1. Introduction

During the engineering design activity of the ITER, the MELCOR 1.8.2 code was selected as one of several codes to be used to perform ITER safety analyses [1]. MELCOR was chosen because it has the capability of predicting coolant pressure, temperature, mass flow rate and radionuclide and aerosol transport in nuclear facilities and reactor cooling systems. MELCOR can also predict structural temperatures (e.g., first wall, blanket, divertor, and vacuum vessel) resulting from energy produced by radioactive decay heat and/or chemical reactions (oxidation). The Idaho National Laboratory (INL) Fusion Safety Program (FSP) organization made fusion specific modifications to the MELCOR 1.8.2 code [2–5] that allowed MELCOR to assess the thermal–hydraulic response of ITER cooling systems and the transport of radionuclides as aerosols during accident conditions. Recently, the ITER International Organization (IO) used a "pedigreed" version of MELCOR 1.8.2 [6] to perform accident analyses for ITER's "Rapport Préliminaire de Sûreté" (Report Preliminary on Safety – RPrS).

The MELCOR thermal–hydraulics code [7,8] is currently under development at Sandia National Laboratory (SNL) for the US Nuclear Regulatory Commission (NRC). The MELCOR code is used to model the progression of severe accidents in light water nuclear reactors. Several versions of the code have been released since the first version in 1989, with the latest official release being version 1.8.6.

Because MELCOR has undergone many improvements between version 1.8.2 and 1.8.6, the INL FSP has started the process of introducing fusion modifications into MELCOR 1.8.6, with the intent of producing a version of MELCOR 1.8.6 with capabilities similar to those of the version of MELCOR being used for the ITER RPrS. While it is our hope that MELCOR 1.8.6 for fusion will be used for future ITER safety studies, this version of MELCOR has not undergone the rigorous quality assurance (QA) process applied to MELCOR 1.8.2 and will not be ready for use in any ITER RPrS analyses. However, as a first step in this QA process, we have applied this version of MELCOR 1.8.6 to the analysis of a large in-vessel coolant leak event examined in the ITER Generic Site Safety Report (GSSR) [9], an earlier safety analysis report in the ITER project history. In this paper, we present the results of this analysis and compare these results to those obtained from the MELCOR 1.8.2 code.

The following section of this article describes the event used here as the basis for comparing these two versions of MELCOR. Section 3 discusses the methodology adopted for this comparison, while Section 4 describes the results obtained from this study. In the final section, we summarize our findings.

2. Large in-vessel coolant leak event description

As stated in Ref. [10], the plant-level response of the ITER design to off-normal events has been an important aspect of the safety

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assessment of ITER through the history of the ITER project. To demonstrate that the safety functions of the ITER design are more than adequate to achieve the overarching safety objectives of the ITER device, a set of representative accident scenarios, known as "Reference events", were identified [11] and analyzed to determine the overall consequences of these events. Of particular importance was the demonstration that the radioactive confinement safety functions of ITER are not compromised by these events, for example the confinement function provided by the ITER vacuum vessel (VV). A postulated event that challenges this VV safety function has been designated the "In-Vessel Multiple First Wall Pipe Break Event".

This event is the double-ended rupture of multiple FW coolant pipes during plasma burn. All FW/Blanket modules around the inboard and outboard toroidal circumference of the machine are postulated to be damaged in this event. Coolant will be discharged at a high flow rate directly into the vacuum vessel (VV). The rapid coolant ingress is assumed to terminate the plasma by inducing a plasma disruption. To test the robustness of the ITER design, an aggravating failure of loss of electrical power is also assumed. Some relevant safety systems and design parameters that assure the integrity of the VV during such events are:

- A VV design pressure of 0.2 MPa;
- The VV Pressure Suppression System (VVPSS) opening at a VV pressure of 0.15 MPa;
- A bleed line into VVPSS and drain lines into the drain tank open at a VV pressure of 90 kPa;
- A VV cooling loop operating in natural circulation heat rejection mode; and
- Detritiation systems supported by emergency power.

The total FW break area considered feasible for this scenario in RPrS calculations is 0.02 m^2. However, for purpose of comparing our results with a previous analysis found in the GSSR, we have adopted for this paper the GSSR break area of 0.2 m^2.

3. Methodology

Two versions of the MELCOR code were selected for this comparison study. The first code is a verified and validated version of MELCOR 1.8.2 modified for fusion applications [6]. This version of the code is being used in analyzing reference events for ITER's RPrS [10]. The second code is a version of MELCOR 1.8.6 that contains identical modifications, as closely as possible, to those of MELCOR 1.8.2 for fusion. Both code versions were applied to the same accident scenario, the In-Vessel Multiple First Wall Pipe Break Event, and used the same input parameters and boundary conditions.

The input deck used for this analysis is that used by previous ITER safety studies [9,10]. A schematic presentation of this deck appears in Fig. 1. Illustrated in Fig. 1 are the computation volumes and flow paths of the MELCOR input model for this analysis. This model simulates the components of the ITER design that fail during this event, such as the FW/Blanket/Shield modules and the cooling system for these components, and systems designed to mitigate the consequences of this failure, in particular the VVPSS. The major components of the VVPSS are the suppression tank, that receives and condenses steam emanating from the VV by way of relief pipes connected to ITER's neutral beam ducts, and the drain tank, which receives excess water from the bottom of the ITER's VV, thereby eliminating the possibility of this water continuing to flash by contacting the hot structures within this vessel. The relief pipes to the suppression tank have two operating modes. The first mode of pressure relief is by way of a bypass line, a line that bypasses

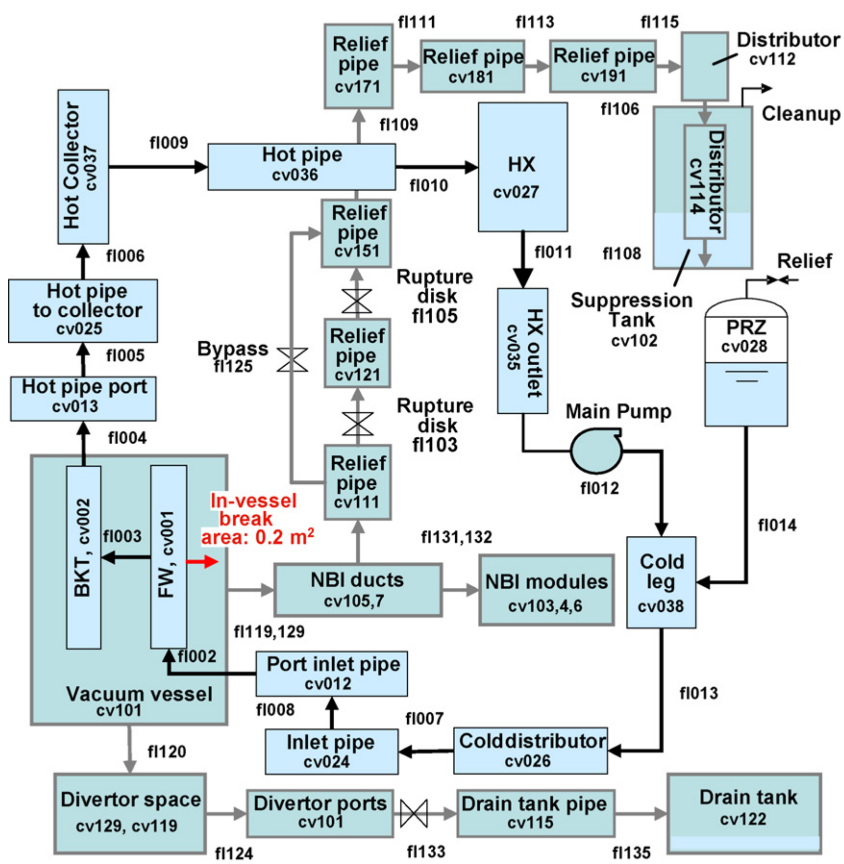

Fig. 1. Schematic presentation of MELCOR thermal–hydraulic input model used for this comparison study.