Safety Analysis of the US Dual Coolant Lead Lithium ITER Test Blanket Module

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An attractive blanket concept for fusion reactor blankets that has been explored in both the United States (US) and European Union (EU) is the dual coolant liquid lead-lithium (DCLL) design concept. This blanket is constructed of reduced activation ferritic steel (RAFS). Helium is used to cool the blanket first wall and internal wall structures, and a self-cooled liquid breeder, Pb–17Li, is circulated for cooling the interior of the blanket and for tritium breeding and removal. A SiCf/SiC composite insert inside of the breeding zone is used as an electrical insulator to reduce the impact of the magnetohydrodynamic (MHD) pressure drop associated with the circulating Pb–17Li and as the thermal insulator to separate the high temperature Pb–17Li from the helium cooled RAFS structure.

In support of the International Thermonuclear Experimental Reactor (ITER) Test Blanket Module (TBM) program, the US is proposing a prototype of this DEMO blanket concept for testing in ITER as an ITER TBM. Because safety considerations are an integral part of the design process to ensure that this TBM does not adversely impact the safety of ITER, a safety assessment has been conducted for this DCLL TBM and its ancillary system as requested by the ITER project. This safety assessment must address a number of concerns that are directly caused by TBM system failures. These concerns stem from four basic loss-of-coolant accident events, such as coolant leaks from the TBM directly into the ITER vacuum vessel (VV). These events were selected to address specific ITER reactor safety concerns, such as VV pressurization, confinement building pressure build-up, TBM decay heat removal capability, tritium and activation products release from the TBM system, and hydrogen and heat production from chemical reactions. This paper presents the results of this safety assessment performed with the MELCOR computer code.