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Evaluation of an IVR-ERVC strategy for a high power reactor using MELCOR 2.1

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ABSTRACT

The IVR-ERVC (In-Vessel Retention of molten corium through External Reactor Vessel Cooling) is an effective severe accident management strategy for reducing the possibility of a reactor containment failure by terminating the severe accident progress inside a reactor. However, the technical applicability and feasibility of the IVR-ERVC design for an advanced high-power reactor should still be validated considering the uncertainties of physical models, the initial conditions and assessment methodologies. In this paper, the severe accident progress of the APR1400 for a large break loss-of-coolant accident is analyzed using MELCOR 2.1 when the reactor cavity is fully flooded. The chronology of events, the thermal hydraulic behaviors and the core degradation behaviors are analyzed. As a result of the MELCOR calculation, a relatively large portion of particulate debris is relocated to the bottom of the lower head at the end of the debris-quench mode, preventing effective heat transfer to the ex-vessel wall. Because the lower head wall cannot be ablated by melting in the MELCOR, the in-vessel wall temperature is increased as compared to the melting point of the lower head. The heat flux is maximized at approximately 3.5e4 s and it is compared to the results from the lumped parameter method.

1. Introduction

If core materials start to be relocated and are not adequately cooled, they may be ejected through a reactor vessel, with subsequent phenomena, i.e., MCCI, FCI and DCH, possibly threatening the containment structural integrity. The IVR concept is considered in order to guarantee the structural integrity of the containment by holding the damaged core materials in the reactor and this is usually realized by the ERVC strategy. In order to utilize the IVR-ERVC as a severe accident management strategy, complicated physical phenomena in a reactor and effective heat removal methods at the external vessel wall should be considered simultaneously. ECCM, CFD, LPM and severe accident integral codes have been used to analyze the thermal hydraulic behaviors of reactors, adopting the ERVC strategy (Theofanous et al., 1995; Esmaili et al., 2004; Tran, 2007; Hong et al., 2011). However, the analysis results by those methods should be used complementary as the levels of physical phenomena to investigate differ from one another. The LPM, which assumes conservative bounding conditions of thermal loading, was used for regulatory audit calculations (Theofanous et al., 1995; Esmaili et al., 2004) and thermal loading conditions of the mass of the core materials and the decay heat were derived from other engineering assumptions or from the analysis results of the severe accident integral codes.

SCDAP/RELAP5, ASTEC, MAAP5 and MELCOR, which are the typical integral codes for severe accident analyses, are capable analyzing the damage of to the core materials and the thermal behavior of the lower head. Analyses using these codes appear more realistic because they can model more systems of nuclear power plants as compared to the CFD at the same time, while also analyzing transient accident progression as compared to the LPM. Among them,

Abbreviations: CFD, computational fluid dynamics; CHF, critical heat flux; CRP, control rod poison; DCH, direct containment heating; DOE, Department of Energy in the United States; DVI, direct vessel injection; ECCM, effective convectivity-conductivity model; ERVC, external reactor vessel cooling; FCI, fuelcoolant interaction; INEEL, Idaho National Engineering and Environment Laboratory; HS, heat structure in MELCOR; HT, heat transfer; ICI, in-core instrument; IVR, in-vessel retention; IVR-ERVC, in-vessel retention of molten corium through external reactor vessel cooling; LB, large break; LOCA, loss-of-coolant accident; LPM, lumped parameter method; MAAP, modular accident analysis program; MCCI, molten corium-concrete interaction; MELCOR, method of estimation of leakages and consequences of releases; MP1, oxidic molten pool in MELCOR; MP2, metallic molten pool in MELCOR; RCP, reactor coolant pump; PRA, probabilistic risk analysis; RCS, reactor coolant system; SBO, station black out; SI, safety injection; SS, stainless steel; SSOX, oxide of stainless steel; TLOFW, total loss of feed-water.

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Nomenclature

β	thermal expansion coefficient (1/K)
θ	inclination angle of the lower head (degree)
v	kinematic viscosity (m^2/s)
ρ_g	density of steam (kg/m^3)
ρ_f	density of water (kg/m^3)
σ	interfacial surface tension between the steam and water (N/m)
g	acceleration of gravity (m/s^2)
H	height of the molten pool (m)
h_{debris_wall}	heat transfer coefficient from particulate debris to the lower head vessel wall ($W/m^2 \cdot K$)
h_{mpw}	heat transfer coefficient from the molten pool to the lower head vessel wall ($W/m^2 \cdot K$)
h_{NB}	nucleate boiling heat transfer coefficient ($W/m^2 \cdot K$)
i_{fg}	latent heat of the vaporization of water (J/kg)
k	thermal conductivity ($W/m \cdot K$)
Nu	Nusselt number
$Nu_{average}$	average Nusselt number at the oxidic molten pool
$Nu(\theta)$	Nusselt number at the oxidic molten pool according to the inclination angle
P	pressure of the reactor cavity pool (Pa)
Pr	Prandtl number
Q	volumetric heat generation (W/m^3)
q''_{debris_wall}	heat flux from particulate debris to the lower head vessel wall (W/m^2)
q''_{mpw}	heat flux from molten pool to the lower head vessel wall (W/m^2)
q''_{NB}	nucleate boiling heat flux (W/m^2)
R	radius of the lower head
Ra	Rayleigh number
Ra_e	external Rayleigh number
Ra_i	internal Rayleigh number
T_{debris}	temperature of the particulate debris (K)
T_{mp}	temperature of the molten pool (K)
$T_{Wall,ex-vessel}$	temperature at the ex-vessel wall of the lower head (K)
T_{sat}	saturation temperature of the reactor cavity pool (K)
$T_{wall,in-vessel}$	temperature of the in-vessel wall of the lower head (K)
ΔT	temperature difference between the bottom and upper surface (K)
ΔT_{ws}	wall superheating (K)

MELCOR is a fully integrated, engineering-level computer code which models the progression of severe accidents in light-water reactor nuclear power plants. It was developed at Sandia National Laboratory for the U.S. Nuclear Regulatory Commission (Humphries et al., 2005). The original purpose of the MELCOR code is a PRA analysis considering simple models based on experiments and several sensitivity variables. However, it is also used at present to analyze severe accidents, as numerical schemes and detailed models based on realistic physical phenomena have been included and fine nodalization has become possible. Core degradation, including the thermal response and the relocation of core materials during melting, slumping and formation of molten pools and debris, can be analyzed in the MELCOR COR package.

Several studies were performed to analyze the IVR-ERVC designs for different reactors using severe accident codes in recent years. These analyses using severe accidents codes are applied to evaluate the safety margins of the strategy from the initiating event to the relocation of core material to the lower plenum transiently or to obtain the input data of the LPM analyses. Core degradation analyses of the APR1400 under the ERVC condition were performed by INEEL and KAERI (Rempe et al., 2005; Kim et al., 2005). The accident scenarios of TLOFW, SBO and LOCA without SI were considered in INEEL analyses using the SCDAP/RELAP-3D code and the limiting case was a LB LOCA with early lower head vessel failure by creep rupture. More than 100 tons of core materials were relocated in the lower plenum and the maximum heat flux at the external vessel wall was 1.64 MW/m^2, which exceeded the CHF value according to a SBLB experiment with a condition including a plain surface and no insulation structure (Cheung et al., 2003). Based on the SCDAP/RELAP-3D result, the heat flux at the external vessel wall was analyzed using VESTA (Rempe and Knudson, 2004), which is a LPM code assuming a two-layer molten pool configuration, the maximum heat flux was about 2 MW/m^2 in that case. KAERI performed a nearly identical analysis using the SCDAP/RELAP/Mod3.3 and LILAC-LP code, where the predicted maximum heat flux was 3.2 MW/m^2. And the ASTEC V1.2 code was applied to evaluate the IVR strategy for VVER-440/213 (Tarabelli et al., 2009). The corium mass, composition and decay heat level are assumed and only the lower plenum behavior was analyzed. The evaluated heat flux at the ex-vessel was lower than the MVITA prediction (Sehgal et al., 2003). The possible IVR phenomena for VVER-1000 was analyzed using ASTEC 2.0 rev2 and the compared the results of the SOCRAT code (Zvonarev et al., 2014). The two codes predicted the location of the maximum heat load differently.

Duspiva analyzed the ERVC design which is applied to the VVER-1000/320 using MELCOR 1.8.6 for a LB LOCA (Duspiva, 2015). More than 160 tons of core materials were relocated to the lower plenum and the maximum heat flux was observed from 15000 s, reaching approximately 1.4 MW/m^2. In addition, nearly 20% of the Zircaloy in use was oxidized. The typical focusing effect at the metallic molten pool was not observed as the location of the highest heat flux is at the elevation of the top of the oxidic pool. The core materials were initially relocated in the form of particulate debris, after which they became oxidic and metallic molten pools. When the maximum heat flux was observed, nearly 60 tons of particulate debris remained in the lower plenum, continuously melting into molten pools.

Jin et al. analyzed the IVR-ERVC design of a three-loop, 5000-MWt-scale pressurized water reactor using MELCOR for a LB LOCA (Jin et al., 2015). The steady-state molten pool state was observed at 55,000 s, and the maximum heat flux was about 1.4 MW/m^2 at 70–80°. The authors found that no lower head failure occurred for more than 1.5e5 s long-term calculations without mentioning any assumptions pertaining to the lower head failure conditions.

The APR1400 is a generation III+ nuclear power reactor which adopts IVR-ERVC as a severe accident management strategy, and it was constructed as Shin-Kori 3&4 initially (IAEA, 2011). When IVR-ERVC is applied to high power reactors, heat flux at the ex-vessel wall becomes large and this results in low thermal margins with respect to the critical heat flux at ex-vessel wall, the amount of the ablated vessel wall for discharging high heat flux and the mechanical strength of the remained wall. Therefore, additional measures are considered such as water injection into the reactor pressure vessel or application of an internal core catcher. (Sehgal et al., 2012; Aquaro et al., 2016). Actually, the in-vessel water injec-