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Passive depressurization accident management strategy for boiling water reactors

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HIGHLIGHTS

- We proposed two passive depressurization systems for BWR severe accident management.
- Sensitivity analysis of the passive depressurization systems with different leakage area.
- Passive depressurization strategies can prevent direct containment heating.

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ABSTRACT

According to the current severe accident management guidance, operators are required to depressurize the reactor coolant system to prevent or mitigate the effects of direct containment heating using the safety/relief valves. During the course of a severe accident, the pressure boundary might fail prematurely, resulting in a rapid depressurization of the reactor cooling system before the startup of SRV operation. In this study, we demonstrated that a passive depressurization system could be used as a severe accident management tool under the severe accident conditions to depressurize the reactor coolant system and to prevent an additional devastating sequence of events and direct containment heating. The sensitivity analysis performed with SAMPSON code also demonstrated that the passive depressurization system with an optimized leakage area and failure condition is more efficient in managing a severe accident.

1. Introduction

The severe accident management guideline (SAMG) provides structured guidance to identify the appropriate strategy and subsequent actions needed to stabilize and return the plant to a controlled stable condition in case of a severe accident condition (Huh et al., 2009). Failure of the lower head under high pressure could eject molten core material at sufficiently high velocities such that the material would not collect in the reactor vessel cavity but would instead be dispersed into the containment atmosphere. The process of rapid energy transfer is generally called direct containment heating (DCH) (Hanson et al., 1990). Reactor vessel depressurization is an important measure before an accident sequence progresses to the point of vessel bottom head penetration failure. Because depressurization prevents high-pressure melt ejection and the threat of direct containment heating, it may potentially reduce the initial challenge to containment integrity (Hanson et al., 1990; Hodge and Petek, 1994). Thus, according to the current SAMG, operators are required to depressurize the reactor cooling system (RCS) as soon as possible using the SRVs (Hodge et al., 1992). RCS depressurization is normally accomplished rather simply by a manually induced actuation of the safety/relief valves (SRVs). As an alternative, for the plants equipped with the reactor core isolation cooling (RCIC) system, isolation condenser, and high-pressure coolant injection (HPCI) system, RCS is depressurized with the operation of a turbine (Hodge et al., 1992).

During the early phases of the accident, two viable options exist for accident mitigation: (1) SRVs are manually opened to depressurize RCS (if SRVs were operable) and (2) the SRVs are left to automatic actuation only. The sensitivity analysis by previous studies has proven that the first option is not efficient in the cases of station blackout (SBO) (Huh et al., 2009). Because the high rate of flow through the open SRVs would cause a rapid loss of reactor vessel water inventory, core plate dryout would occur soon after the operation (Ott, 1989). Heat-up of the totally uncovered core would then lead to significant structural melting and relocation. Because the core plate would be dry at this time, severe heat-up and local failures would occur immediately after debris relocation began.

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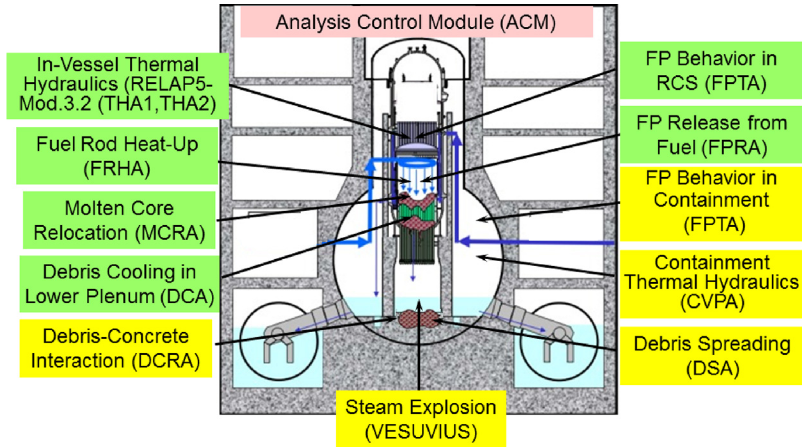

Fig. 1. Modules of SAMPSON (Ikeda et al., 2003b).

However, a slight delay (approximately several minutes) could be obtained if the SRVs were left to automatic actuation only.

In the late phases of a severe accident where core melting has already occurred, the DC power and control air are necessary for vessel pressure control by remote-manual SRV actuation (or HPCI/RCIC steam turbine operation). Based on the information currently available, depressurization of RCS using SRVs would be sufficient to mitigate DCH. However, there are human factors and equipment issues that could affect the capability to successfully implement intentional depressurization (Hanson et al., 1990). A human factors evaluation identified two important factors that could affect the performance of plant personnel in implementing depressurization strategies: (1) the capability of plant personnel to recognize that a depressurization strategy should be implemented and (2) the capability of personnel to implement the necessary depressurization actions in the available time (Hanson et al., 1990). If the depressurization is too late, the core intermittently loses water inventory through the SRV(s) and might be damaged under the high-pressure condition. Hence, there is a threat of early containment failure because of DCH. If the depressurization is too early, on the other hand, before preparing the measures of alternative water injection, the core is kept uncovered after the depressurization and might be damaged under low pressure (Kawahara et al., 2013). These two factors troubled plant personnel of the Fukushima Dai-ichi nuclear power plant Unit 1 (1F1) during the Fukushima accident. Thus, there is a potential need for another (backup and last-resort) method to keep the reactor vessel depressurized in case the normal methods are no longer be available for this purpose (Hodge et al., 1992). A lesson learned from the Fukushima accident showed that it is necessary to have an alternate means of reactor vessel venting once SRVs become inoperable because of a loss of control air or DC power. Probabilistic risk assessments based upon the existing BWR facilities consistently include accident sequences involving the loss of DC power and control air among the dominant sequences leading to core melt for BWRs (Hodge and Petek, 1994). And the situation in 1F1 provided an insight to find a potential strategy to depressurize the reactor. For 1F1, even though the records do not show any deliberate attempt to depressurize the RPV, by 2:45 a.m. on March 12, the RPV pressure was found to be 0.8 MPa (Nuclear Emergency Response Headquarters, 2011). It is not clear whether RPV depressurization occurred because of damage to RPV by the molten core, by a break in an attached low-elevation pipe, or by SRVs that had stuck open (Klein and Corradini, 2012). Thus, it may have been caused by the damaged parts from which steam/gas was released from the RPV or its peripherals directly into the PCV (Koshizuka, 2012).

The depressurization caused by the leakage through the pressure boundary was not included in the SAMG at present. Therefore, we describe an alternative system in this study that can be performed to provide guidance for late-phase operator actions. The new method may significantly improve the existing BWR emergency procedure strategies to limit the extent of on-going damage to the in-vessel structures and to terminate the accident (Hodge and Petek, 1994). For that aim, we propose two passive depressurization systems for BWR severe accident management that may prevent DCH and give operators a chance to inject water into the RPV before the failure. As a result, with the achievement of earlier debris cooling, the source term can be reduced even if the RPV eventually fails. All sensitivity analyzes addressing the depressurization events are performed with SAMPSON severe accident analysis code.

2. SAMPSON code overview

SAMPSON is an integral code for a detailed accident analyses with modular structure developed in the IMPACT project of Japan. It is composed of twelve interrelated modules. A schematic description is illustrated in Fig. 1. SAMPSON integrates various analysis modules into a single code and calculates physical and chemical behaviors for various areas in a hypothesized core melt event based upon computer simulation by using fundamental physics principles and sophisticated modeling technologies. Each module can run independently and can communicate with multiple analysis modules supervised by the analysis control module that makes an integral analysis possible, as shown in Figs. 1 and 2 (Ikeda et al., 2003a,b).

Each analysis module of the SAMPSON code has been developed and validated against separate-effect tests with good agreement (Ujita et al., 1999). The thermal-hydraulics calculation of the SAMPSON code is based on RELAP5 code. The in-core molten core relocation analysis module was originally developed based on SIMMER code. It implements a detailed and mechanistic analysis, solving the flow (mass, momentum and energy for multi-phase and multi-component) in three-dimensional or quasi-three-dimensional configurations without overemployment of correlations (Cadiou et al., 2002; Maschek et al., 2005; Morita et al., 1999; Naitoh et al., 2005). The debris coolability analysis module was developed for the thermal hydraulics analysis of debris relocated in the lower head (Hidaka et al., 2002). The simulation results using this module showed good agreement with the experimental data for predicting of the flow spearhead location (Hidaka et al., 2002). The containment vessel phenomena analysis module analyzes the behavior of pressure and temperature in a containment