Nuclear Engineering and Design 315 (2017) 77–92

Contents lists available at ScienceDirect

Nuclear Engineering and Design

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

Quantification of LOCA core damage frequency based on thermal-hydraulics analysis

Jaehyun Cho *, Jin Hee Park, Dong-San Kim, Ho-Gon Lim

Korea Atomic Energy Research Institute, 1405 Daedeok-daero, Yuseong-gu, Daejeon 305-353, Republic of Korea

HIGHLIGHTS

- We quantified the LOCA core damage frequency based on the best-estimated success criteria analysis.
- The thermal-hydraulic analysis using MARS code has been applied to Korea Standard Nuclear Power Plants.
- Five new event trees with new break size boundaries and new success criteria were developed.
- The core damage frequency is 5.80E−07 (/y), which is 12% less than the conventional PSA event trees.

ARTICLE INFO

Article history:
Received 4 March 2016
Received in revised form 17 February 2017
Accepted 18 February 2017

Keywords:
LOCA
Probabilistic safety assessment
Core damage frequency
Safety injection
Normal secondary cooling

ABSTRACT

A loss-of-coolant accident (LOCA) has always been significantly considered one of the most important initiating events. However, most probabilistic safety assessment models, up to now, have undoubtedly adopted the three groups of LOCA, and even an exact break size boundary that used in WASH-1400 reports was published in 1975. With an awareness of the importance of a realistic PSA for a risk-informed application, several studies have tried to find the realistic thermal-hydraulic behavior of a LOCA, and improve the PSA model. The purpose of this research is to obtain realistic results of the LOCA core damage frequency based on a success criteria analysis using the best-estimate thermal-hydraulics code. To do so, the Korea Standard Nuclear Power Plant (KSNP) was selected for this study. The MARS code was used for a thermal hydraulics analysis and the AIMS code was used for the core damage quantification. One of the major findings in the thermal hydraulics analysis was that the decay power is well removed by only a normal secondary cooling in LOCAs of below 1.4 in and by only a high pressure safety injection in LOCAs of 0.8–9.4 in. Based on the thermal hydraulics results regarding new break size boundaries and new success criteria, five new event trees (ETs) were developed. The core damage frequency of new LOCA ETs is 5.80E−07 (/y), which is 12% less than the conventional PSA ETs. In this research, we obtained not only thermal-hydraulics characteristics for the entire break size of a LOCA in view of the deterministic safety assessment, but also a more realistic core damage frequency of the LOCAs using updated information. The difference between the results of the present study and the conventional knowledge can be used to modify the current emergency operator procedures and to design another nuclear power plant type.

1. Introduction

At the beginning stage of nuclear safety, nuclear safety functions, such as a safety injection, were developed based on hypothetical accidents of nuclear power plants (NPPs). Among these accidents, a loss-of-coolant accident (LOCA) has been predominantly considered the most severe. In a probabilistic safety assessment (PSA), a LOCA is also considered one of the most important initiating events. Since the publication of WASH-1400 in 1975 and the division of the LOCAs into three groups in the report (USNRC, 1975), most PSA models, up to now, have undoubtedly adopted the three groups of LOCA and even an exact break size boundary. Against this background, for a pressurized water reactor (PWR), the conventional PSA model divides the LOCAs into three groups along the break sizes: a small break size (below 2.0 in), medium break size (from 2.0 to 6.0 in), and large break size (above 6.0 in).

With an awareness of the importance for a realistic PSA for a risk-informed application, several studies related to a LOCA have

* Corresponding author.
E-mail address: chojh@kaeri.re.kr (J. Cho).

http://dx.doi.org/10.1016/j.nucengdes.2017.02.023
0029-5493/

been conducted. Han and co-authors (2007) assessed the feasibility of a method to estimate an operator's action for a small break LOCA without a high pressure safety injection (HPSI) for the Korea Standard Nuclear Power Plant (KSNP). This shows that normal secondary cooling can be a success criterion in a small LOCA without a safety injection, although it is in contrast with a conventional PSA. Furthermore, another study conducted thermal-hydraulics calculations for a LOCA to improve the accident sequences and success criteria of the event tree (Lee et al., 2014). However, in this study, the number of groups and break size boundary in a conventional PSA were used, and the core damage frequency was not quantified. In addition, the Technical University of Madrid suggested an integrated safety assessment for a LOCA; the group of authors concluded that the present emergency operating procedures (EOPs) are adequate for managing accidents (Gonzalez-Cadelo et al., 2014). However, this study did not focus on the PSA model or quantification of the core damage frequency.

The purpose of the present research is to obtain realistic results of the LOCA core damage frequency based on the success criteria analysis using the best-estimated thermal-hydraulics code. For this purpose, we first identify the characteristics of NPP response for all break sizes of a cold-leg LOCA (Chapters 2 and 3). Next, in Chapter 4, a quantification of the LOCA core damage frequency is obtained by restructuring all event trees based on the thermal-hydraulics results. In Chapter 5, several insights obtained from the present research are summarized.

2. Materials and method

2.1. MARS KS model for KSNP

To obtain the general characteristics of the thermal-hydraulics behavior in the LOCA, the Korea Standard Nuclear Power Plant (KSNP), whose name originates from the Optimized Power Reactor 1000 (OPR 1000), was selected. The KSNP is one of the NPP types developed in South Korea. Among 32 units of NPPs (including constructing) in South Korea, 10 are KSNPs. This is a pressurized water reactor with 1000 MWe of electricity generation. There are two coolant loops, one steam generator (SG), and two reactor coolant pumps (RCPs) for each loop [Website].

The MARS code was developed by KAERI based on the RELAP5/MOD3 and COBRA-TF codes (Jeong et al., 1999; Lee et al., 2002). The governing equations developed in the MARS code are one-dimensional conservation equations for mass, momentum, and energy of the flow. It adopts a non-homogeneous, non-equilibrium two-fluid model for a two-phase flow. The reactor kinetics model uses the point kinetics with reactivity feedback coefficients and the decay power model. For the reasons outlined above, the MARS code is being considered a best-estimate thermal-hydraulic code to conduct the success criteria analysis of the PSA for NPPs of South Korea.

KAERI has developed the MARS input model of a KSNP for a realistic analysis of small- and large-break LOCAs (Jeong et al.,

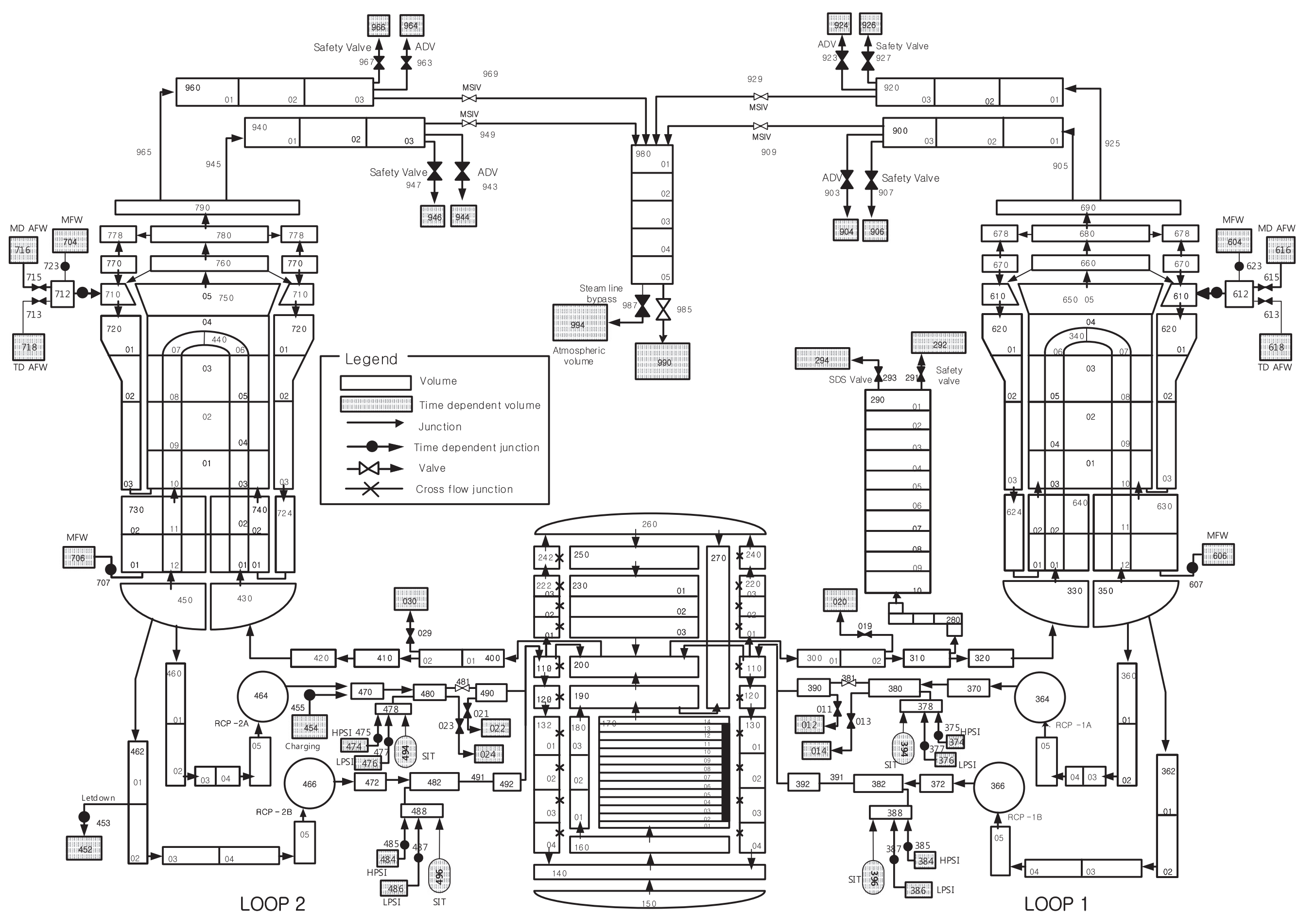

Fig. 1. Nodalization scheme of the MARS input model for the LOCA calculations of the KSNP.