Annals of Nuclear Energy 113 (2018) 55–64

Contents lists available at ScienceDirect

Annals of Nuclear Energy

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

An improved simplified method of evaluating severe accident source term in the containment of AP1000

Xiaohui Sun [a], Xinrong Cao [a,*], Xingwei Shi [b]

[a] Fundamental Science on Nuclear Safety and Simulation Technology Laboratory, College of Nuclear Science and Technology, Harbin Engineering University, Harbin, Heilongjiang 150001, PR China

[b] Nuclear and Radiation Safety Center, Beijing 100082, PR China

ARTICLE INFO

Article history:
Received 22 March 2017
Received in revised form 27 September 2017
Accepted 1 October 2017

Keywords:
Source term
AP1000
ISST
MELCOR
NUREG-1465
SST

ABSTRACT

This paper focuses on the source term estimation in AP1000 Nuclear Power Plant (NPP) under severe accident condition. Methods that utilize MELCOR code can evaluate the source term, but the evaluation process is time-consuming. Also, the reference values in NUREG-1465 can be directly used as the severe accident source term; however, it was initially proposed for second–generation reactors and it has difficulty in simulating changes in release fraction along with accident process. In this paper, the Improved Simplified Source Term (ISST) code was developed based on the IAEA's Simplified Source Term model (SST) presented in IAEA-TECDOC-1127. In addition, the models for core uncovery stage and adiabatic heating-up stage were optimized in this work; taking into account the Accumulator (ACC) and the Core Makeup Tank (CMT) of the emergency core cooling system. Moreover, four fission products release evaluation models were integrated into ISST, making it applicable for the source term evaluation of AP1000 reactor under severe accident initiated by LBLOCA condition. To verify the developed ISST code, the IAEA's SST, MELCOR and the reference source term in NUREG-1465 were differently used to evaluate the source term released into the containment in a LBLOCA of the AP1000 nuclear power plant. The results obtained from the models in the ISST code were analyzed and compared with each other, and with those obtained from NUREG-1465, IAEA's SST and MELCOR. Compared with results calculated with the IAEA's SST, the results calculated with ISST are closer to those calculated using MELCOR. ISST is computationally faster than MELCOR, hence it gives the source term calculation results quicker. Compared with results calculated with MELCOR, the results calculated with ISST are more conservative for Cs and I while the result derived from NUREG-1465 for noble gases is more conservative.

1. Introduction

In severe accidents of nuclear power plant (NPP), large amounts of radioactive nuclides are released to the containment. If the containment fails or the containment is bypassed, radioactive fission products will be released to the environment, causing environmental radiation pollution; which is a threat to public health (Sun et al., 2016). During the period of peaceful use of nuclear energy by mankind, the Three Mile Island accident, Chernobyl accident and Fukushima accident had happened. Especially, Chernobyl accident and Fukushima accident had caused great damage to the environment and public health. Consequently, The source term research has also become a significant part of nuclear power plant safety and risk evaluation.

Currently, there are two kinds of research methods on source term in the world: experimental research and theoretical research. Experimental research began from the 1970s, and the major research includes the Germany SASCHA experiment (Albrecht et al., 1979), the United States HI/VI series experiments (Lorenz and Osborne, 1995), Canada CRL experiment (Lui et al., 1994), France HEVA/VERCORS series experiments (Pontillon et al., 2010), Japan VEGA experiment (A. Hidaka, 2011), the Russian QUENCH-VVER experiment (Goryachev et al., 2006), and Phebus FP experiment (Schwarz et al., 1999). Theoretical research mainly includes the method using MELCOR or other integral codes and the simplified source term evaluation method. The CORSOR series models in MELCOR (Suckow et al., 2007) and MAAP (Bachere and Duplat, 2005), semi-mechanism release model in ASTEC (Plumecocq et al., 2003) and mechanism release model in MFPR (Veshchunov et al., 2006) have been used in the integral codes calculation method. There are several simplified source term evaluation

* Corresponding author.
E-mail address: caoxinronghrbeu@163.com (X. Cao).

https://doi.org/10.1016/j.anucene.2017.10.001
0306-4549/

Nomenclature

t	time (s)
Q'	decay heat (W)
q	volumetric heat generation rate of Zr oxidation (W/m^3)
H	latent heat of water (J/kg)
T	temperature (K)
C	specific heat capacity of water (J/kg K)
f	fraction of the reactor full thermal power
r	fuel pin radius (m)
c	constant
D	diffusion coefficient (m^2/s)
W	activation energy (j/mol)
R	ideal gas constant
Greek	
ω	weight (kg)
Δ	change
χ	fraction of core uncovered
σ	the quantity of coolant boiled off in a time step
Subscripts	
bd	blowdown
u	uncovery
s	start
e	end
dh	decay heat
wcs	water in coolant system
wac	water in ACC and CMT
wcr	water in core region
fs	fuel assemblies and support structures in core region
oxi	Zr oxidation

methods. The typical one is the reference source term in NUREG-1465 (Soffer et al., 1995). This reference source term can be used in the research on fission product removal by natural processes (Powers et al., 1995) and source term rapid evaluation system for nuclear emergency (Ramsdell et al., 2012). Other two simplified methods for source term evaluation are the method in IAEA-TECDOC-1127 (International Atomic Energy Agency, 1999) and the method developed by Huang et al. (2014). The advantages and disadvantages of these methods are obvious. The method using MELCOR code can accurately simulate the source term of a severe accident in NPP, but the modeling process is complicated and time-consuming. Also, this method is not consistent with cost-benefit assessment (Huang et al., 2014). The simplified evaluation method utilizing reference source term in NUREG-1465 and IAEA's Simplified Source Term method (SST) can simulate the source term in a severe accident in nuclear power plant faster. However, reference values in NUREG-1465 were initially proposed for the second-generation reactor, and it has difficulty in simulating changes in release fraction along with accident process. Also, there are several weaknesses in SST which can be improved. Actually, the simplified method of source term evaluation needs to take into account both time and accuracy requirements.

In fact, the fission products release models in MELCOR and MAAP are functions of reactor core temperature. However, most of the time they are used in simulating the thermal hydraulics process in the primary circuit system after the accident; and during the process of model establishment and program calculation. Therefore, simplifying the thermal hydraulics process in the primary circuit system after the accident will significantly shorten the modeling process and calculation process, and improve the computation efficiency. A new calculation method, the Simplified Source Term method (SST) is proposed in the IAEA-TECDOC-1127, which simplifies the thermal hydraulics process during an accident. The released model is an improvement on the Kress/Booth RelVol model (Kress et al., 1987). This method has several weaknesses: it does not consider the new features of the primary circuit system of AP1000 nuclear power plant, such as Accumulators (ACCs) and Core Makeup Tanks (CMTs), etc. Also, the models for core uncovery and core adiabatic heating-up stage are rough, and the fission product release model is relatively outdated. In this paper, a new code, the Improved Simplified Source Term (ISST) is developed based on the IAEA's SST model presented in the IAEA-TECDOC-1127.

Large Break Loss of Coolant Accident (LBLOCA) was selected as a reasonable scenario upon which to base the timing of initial fission product release into the containment with the reason that Loss of Coolant Accidents (LOCAs) are a substantial contributor to Core Damage Frequency (CDF) for PWR according to NUREG-1465 (Soffer et al., 1995). LBLOCA was selected because LBLOCAs occurs for a shorter time duration and results in a larger source term according to IAEA-TECDOC-1127 (International Atomic Energy Agency, 1999). In this paper, the source term in a LBLOCA was also selected as the reference source term in a severe accident. It should also be noted that ISST can be used only for source term evaluation in the severe accident initiated by LBLOCA. Four methods, namely the IAEA's simplified source term method, the ISST procedure, MELCOR, and the reference values in NUREG-1465 are used to calculate the source term released into the containment, respectively. Then, the applicability of the ISST program used for simplified source term evaluation will be verified by comparative analysis.

2. The improved simplified source term method

2.1. Main differences between ISST and SST

The Improved Simplified Source Term method (ISST) was developed based on the IAEA's Simplified Source Term model (SST) in IAEA-TECDOC-1127. The main differences between ISST and SST are as shown in Figs. 1 and 2. Core uncovery phase and core adiabatic heat-up phase were improved in ISST. Moreover, four fission products release models were integrated into ISST.

2.2. Introduction of ISST code

As illustrated in Fig. 3, thermal hydraulics and core degradation models were simplified in IAEA-TECDOC-1127. The models were postulated based on the experience with the application of core melt thermal hydraulics codes and with limited in-pile experimental data. Although the temperature profile of the core is not uniform, it is assumed that the whole core behaves in the same way as the fuel pin. This can affect the timing as well as the magnitude of the overall source term released into the containment. However, treating all parts of the core as behaving in this way simultaneously is conservative with respect to the timing and magnitude of the source term (International Atomic Energy Agency, 1999). To convert the generalized thermal transient into fission product release, the following input is required: