Review of radiological source term evaluation methodologies to improve SIRIUS code in CINEMA

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1. Introduction

Present study was performed to set the goal of the task of "development of standard radioactive source term evaluation technology in containment buildings in case of serious accidents" within the "Core Technology Development Project for Improving Operational Nuclear Power Plant Safety" in 2022. We prepared present study to organize the essential technical elements of the radioactive source term evaluation to be performed in this project by reviewing the existing research results and regulatory conditions. We reviewed the evaluation methodology of radioactive source terms that have been developed so far, and summarized the essential technical issues in the evaluation of the radioactive source term established according to the current US NRC standard. Based on this, the current status of the fission product behavior analysis module, which is currently being dealt with in the domestic integrated severe accident analysis code (CINEMA), was analyzed, and finally, the points to be improved in the SIRIUS module were derived.

2. Review of source term evaluation methodology

2.1 TID-14844

The Technical Information Document (TID-14844) [2-1], which serves as the basis for the initial assessment of radiation hazards, was developed in 1962 for the purpose of securing a safe distance even in the event of the worst-case scenario, or maximum credible accident, for reactor sites. While TID-14844 is considered overly conservative due to its assumption of a maximum credible accident, it is still considered fundamental to the design and operation of most operating nuclear power plants. The regulatory framework for source term evaluation at domestic nuclear power plants is also based on TID-14844, with site boundary setting (10.CFR.100.11, NRC Regulatory Guide 2014-10) being applied, and numerous studies are being conducted to revise guidelines and notices to exclude excess conservatism. [2-2 ~ 2-4]

2.2 NUREG-1465

NUREG-1465 [2-5] is a report published in 1995, and the source term evaluation methodology presented in this report is called the Alternative Source Term (AST). Following the Three Mile Island Unit 2 (TMI-2) accident in 1979, the amount of iodine released was relatively small compared to the estimated value used for licensing calculations. The US NRC, based on a better understanding of substantial meltdowns, raised the need for the development of a more physically and realistically based radioactive source term that could be applied to design basis accidents. To this end, NUREG-1465 presents the release timing (gap release, early in-vessel, ex-vessel, late in-vessel) and release fractions of eight nuclide groups based on the calculation results of the initial version of the comprehensive accident analysis code, MELCOR. The methodology presented in NUREG-1465 became the basis for NRC Regulatory Guide 1.183 and is still widely used to this day.

Table 1. Release fraction of source term to the containment building (PWR) [2-5]

	Gap Release***	Early In-Vessel	Ex-Vessel	Late In-Vessel
Duration (Hours)	0.5	1.3	2.0	10.0
Noble Gases**	0.05	0.95	0	0
Halogens	0.05	0.35	0.25	0.1
Alkali Metals	0.05	0.25	0.35	0.1
Tellurium group	0	0.05	0.25	0.005
Barium, Strontium	0	0.02	0.1	0
Noble Metals	0	0.0025	0.0025	0
Cerium group	0	0.0005	0.005	0
Lanthanides	0	0.0002	0.005	0

2.3 SAND2011-0128

USNRC and US national laboratories such as SNL have further developed the alternative source term after NUREG-1465. The SAND2011-0128 report [2-6] examines the impact of the source term when high-burnup fuel and ice-condenser in the containment building are present in both BWRs and PWRs in the US. Since this report is based on MELCOR version 1.8.5, there are some differences from the models that the current version of MELCOR code can calculate. However, like in NUREG-1465, the report distinguishes the release timing of the source term (divided into 4 accident phases: gap release, in-vessel release, ex-vessel release, and late in-vessel release) and presents the release fraction for 9 chemical classes of radionuclides, thus advancing NUREG-1465.

Table 2 Revised chemical classes of radionuclides adopted for SAND2011-0128

Chemical Group Name	Elements in the Group
Noble Gases	Kr, Xe
Halogens	Br, I
Alkali Metals	Rb, Cs
Tellurium Group	Se, Sb, Te
Barium, Strontium Group (Alkaline Earths)	Sr, Ba
Molybdenum Group	Mo, Nb, Tc
Noble Metals	Ru, Rh, Pd, Co
Lanthanides	La, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
Cerium Group	Ce, Pu, Np, Zr

2.4 US NRC Regulatory Guide 1.183 rev. 1

As mentioned earlier, the US NRC Regulatory Guide 1.183[2-7] was first published in 2000 as a guideline for how the US NRC should review regulations using NUREG-1465. NUREG-1465 was developed for the purpose of providing information on the behavior of radioactive materials during the progression of a physical accident and to be applied to the design of advanced LWRs. These source terms were divided into different categories based on the composition and size of the radioactive material, as well as their chemical and physical characteristics, and the timing of their release to the containment building. NRC considered the applicability of NUREG-1465 source terms to operating reactors and mentioned that the current analysis approach based on TID-14844, which is also based on source terms, is still appropriate. Therefore, operating reactors licensed according to TID-14844 do not need to reanalyze accidents using NUREG-1465 source terms. However, the NRC has determined that it may request the use of an AST in the analysis to support cost-effective licensing actions for some operating reactors' licensees.

The first revised edition of this document was released in April 2022, after more than 20 years since its initial publication, incorporating new issues and major accident research results identified after the initial publication. The new contents included in the revised edition are (1) a clear definition of the maximum hypothetical accident (MHA) loss-of-coolant accident (LOCA) terminology, (2) the addition of transient release fractions resulting from experimental data and interpretation results based on various international research findings, (3) revision of steady-state release fractions for accidents other than LOCA, (4) the addition of information that the proposed regulatory guide can provide useful information to meet the radiation dose analysis requirements of 10 CFR Part 50 and 10 CFR Part 52 for advanced LWR design and site selection, (5) additional guidelines for modeling BWR main steam isolation valve (MSIV) leakage, (6) guidelines for source term related to accident-tolerant fuel (ATF), high burnup fuel, and increased enrichment, (7) modification of the transportation and contamination removal model for fuel handling DBA, (8) guidelines added to recognize MSIV leakage mitigation within the main steam line and condenser of BWR, and (9) additional guidance for weather assumption. Therefore, based on the content of this regulatory guide, the requirements for the development of a radiation source term evaluation methodology that we aim to develop in this research project will be selected.

2.5 Summary of source term evaluation methods

Following table 3 shows summarized characteristics of source term evaluation methods.

Table 3 Comparison of source term evaluation methods

Parameters	TID-14844	NUREG-1465	SANDI2011-0128
Regulatory use by NRC	RG 1.3, RG 1.4	RG 1.183	DG 1389 (Draft)
Target NPPs	Not specified	3 BWRs and 4 PWRs in US	2 BWRs and 2 PWRs in US
Accident sequences	Maximum credible accident	Those significantly impacting the early phase releases	Those covering the majority of CDF
Tool	-	STCP (+MELCOR)	MELCOR 1.8.5
Fuel	-	Low-burnup fuel	**High-burnup or MOX fuel**
Release phases	-	Gap release, early in-vessel, ex-vessel, late in-vessel	Same as NUREG-1465
Release time	Instantaneous	Step-wise for 1.8 hr	**Step-wise for 4.7 hr**
Estimate of representative values	-	Low volatile: 75% value Others: mean	**Median**
Release fractions	Noble gas: 100% Iodine: 50% Solid: 1%	Noble gas: 100% Iodine: 40%, Cs: 30% Te: 5%, Ba: 2%	Noble gas: 96% Iodine: 37%, Cs: 23% **Te: 30%, Ba: 0.4%**
Iodine form	I_2: 90% Organic: 4% Aerosol: 5%	Aerosol: 95% Inorganic vapor: 4.85% Organic vapor: 0.15%	Same as NUREG-1465

3. Future work to improve CINEMA

In the present study, we summarized future work to improve CINEMA[3-1] as follows.

1. It is necessary to add radionuclide species that are not modeled compared to NUREG-1465 in the initial inventory of nuclear fission products. Additionally, it is necessary to calculate the unique core inventory for OPR1000 and APR1400, which are the main goals of this study.

2. The release fraction of fission products needs to be user-optimized to allow conservative evaluation when applying NUREG-1465 or TID-14844 methodology.

3. Currently, the SIRIUS code cannot simulate the behavior of iodine gas and can only interpret it as CsI aerosol by reacting with cesium. Improvements are needed to simulate the behavior of gas iodine and organic iodide, which are emphasized in NUREG-1465.

4. It is urgently necessary to model important mechanisms such as re-suspension caused by turbulent flow in pipes in the aerosol condensation and removal models considered in the MELCOR RN package.

5. Additional models related to the condensation of steam and fission products on aerosol surfaces are necessary. Currently, SIRIUS reflects that steam can condense on the aerosol surface even if the ambient relative humidity is not 100% by applying only the hygroscopic model for hygroscopic aerosols. However, it does not simulate the phenomenon of steam condensation/evaporation according to the basic thermal-hydraulic environment used in MELCOR. In addition, a model for the condensation/evaporation of fission product steam is necessary. There is also an

urgent need to add models for safety equipment such as pool scrubbing and PCCS (passive containment cooling system) applicable to APR1400.

6. It is necessary to improve the two-way coupling method with CSPACE and SACAP, which are thermal-hydraulic analysis codes of CINEMA, to model the distribution of decay heat in RCS or containment buildings.

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