

Evaluation Plan on In-vessel Source Term in PGSFR

Seung Won Lee *, Kwi-Seok Ha, Sang June Ahn, Kwi Lim Lee, Taekyeong Jeong

SFR Reactor Design Division

Korea Atomic Energy Research Institute (KAERI)

Daedeok-daero 989-111, Yuseong-gu, Daejeon, 305-353, Republic of Korea

**Corresponding author: swonlee@kaeri.re.kr*

1. Introduction

The potential for accidents to cause the release of radionuclides into the public environment is the source of safety concern with the use of nuclear reactors for power generation, research and actinide transformation. Safety concerns with nuclear power plants are sufficient that a conservative safety strategy termed “defense in depth” has been adopted essentially universally. This strategy requires nuclear plants to have features that prevent radionuclide release and multiple barriers to the escape from the plants of any radionuclides that are released despite preventive measures. Considerations of the ability to prevent and mitigate release of radionuclides arise at numerous places in the safety regulations of nuclear plants. The effectiveness of mitigative capabilities in nuclear plants is subject to quantitative analysis. The radionuclide input to these quantitative analyses of effectiveness is the Source Term (ST). All features of the composition, magnitude, timing, chemical form and physical form of accidental radionuclide release constitute the ST [1]. Also, ST is defined as the release of radionuclides from the fuel and coolant into the containment, and subsequently to the environment [2].

Since the TMI accident in 1979, extensive experimental and analytical information has been accumulated on the accident ST for LWRs. Such mechanistic models and computer codes as the MELCOR and MAAP have been developed [3]. The results of extensive calculations and experiments have been used to formulate an alternative to the simple TID-14844 ST [4] for regulatory purpose. This Alternative ST (AST), NUREG-1465 ST categorizes radionuclides into eight chemical classes based on chemical and physical similarity [5]. AST specifies the release fractions of each class of radionuclides into the containment during each of the four accident phases: gap release, in-vessel release, ex-vessel release and late in-vessel release. Use of AST is optional for existing LWRs. Future LWRs are required to use AST. AST is certainly not applicable to SFR. AST provides a valuable insights and framework for the development of a mechanistic ST model for SFR to be used in licensing as well as risk analysis.

There are not much experimental data or experience about the source term of metal fuel in SFR. Since one example of the source term about metal fuel comes from ANL-ART-38 report [6], KAERI preliminarily will

evaluate the in-vessel ST using ANL methodology in additional to 4S (Super-Safe, Small and Simple) methodology in the Prototype Gen-IV Sodium-cooled Fast Reactor (PGSFR). This paper shows the matters of progress of the preliminary evaluation on the in-vessel ST.

2. Calculation of In-vessel Source Term

2.1 Radionuclide Groups and Inventory

The elements to be evaluated and the radionuclide groups were specified based on NUREG-1465 ST [5] and Regulatory Guide (RG) 1.183 [7]. Because uranium is not defined in RG 1.183, it was included in the cerium group. A separate group was added, because the operation of liquid metal-cooled reactors results in the activation of the sodium coolant. Radionuclides with a half-life of more than 1 minute are considered. The radionuclide groups and the elements are as follows:

1. Nobles Gases: Xe, Kr
2. Halogens: I, Br
3. Alkali Metals: Cs, Rb
4. Tellurium Group: Te, Sb, Se
5. Barium, Strontium: Ba, Sr
6. Noble Metals: Ru, Rh, Pd, Mo, Tc, Co
7. Lanthanides: La, Zr, Nd, Eu, Nb, Pm, Pr, Sm, Y, Cm, Am
8. Cerium Group: Ce, Pu, Np, U
9. Coolant: Na

The radiological inventory is proportional to the thermal power and is gradually accumulated depending on the radionuclide. The inventory of each radionuclide is calculated by ORIGEN-2 code using the realistic burnup conditions. The inventory of the coolant is calculated by using the sodium mass (kg) and the specific activity (Ci/kg) [8].

2.2 Release from the Core to Primary Sodium

ST in the release from the core to primary sodium will be calculated by using the assumption (700 °C ~ 1,000 °C) of ANL-ART-38 report [6]. Table I shows the release fraction from the core to primary sodium using for this calculation.

The main assumptions are as follows [6]:

In noble gases group, the increase in fuel temperature, and small amount of fuel melting in eutectic regions of the fuel matrix, should only increase the amount of noble gases released when compared to fuel failure near normal operating temperatures, as fuel melting liberates noble gases that remain trapped within the fuel matrix. Therefore, the release of noble gases remains highly burnup dependent, but may be slightly higher than the maximum of 85 % seen at lower temperatures. A release percentage of up to 100 % of the noble gas inventory is noted.

Table I: Release Fraction from the Core to Primary Sodium

Radionuclide Groups		Release Fraction (Core to Primary Sodium)
Noble Gases		100 %
Halogens		20 %
Alkali Metals		60 %
Te Group		1 %
Ba, Sr	Ba	10 %
	Sr	5 %
Noble Metals		1 %
Ce Group	Ce	5 %
	U&Pu	0.1 %
Lanthanides		1 %

In halogens group, the low release rate of iodine seen at SRE (Sodium Reactor Experiment) was likely a result of the very low fuel burnup, which limited initial iodine migration to the NaK bond region. However, the SRE results do demonstrate that iodine release from the fuel matrix at this temperature range is small, even with significant eutectic melt regions forming. Therefore, the release fractions for iodine with fuel pin failure in this temperature range will likely be only slightly higher than the fraction of iodine contained within the bond sodium region. Based on the assumptions of the previous temperature category, a release percentage of up to 20 % is assumed.

In alkali metals group, the SRE incident appears to indicate that cesium release from the fuel matrix is small, even with significant eutectic melt formation. Thus, the determining factor is the quantity of cesium that has migrated to the bond sodium and plenum regions. Therefore, cesium release percentage is set at up to 60 %.

In Te group, The SRE incident appears to indicate that significant eutectic melting does not cause the liberation of a significant amount of tellurium. It should also be noted that many of the damaged SRE pins did not use a uranium zirconium alloy, which would aid tellurium retention through zirconium tellurides. With little additional data, the release percentage for

tellurium in this temperature range is set at up to 1 % of the total inventory.

In Ba, SRE appears to indicate that barium release in this temperature range is small, but the low burnup of the fuel limits its range of applicability, as barium migration to the NaK bond may not have occurred. It is assumed that the barium release noted at SRE was from barium within the fuel matrix, not the bond. Therefore, a release percentage estimate for barium of up to 10 % is assumed.

In Sr, It appears that a small release of strontium is possible in this temperature range, likely due to exceeding the melting point of elemental Sr (768 °C) and the formation of eutectic melt regions in the fuel, which may allow strontium migration to the sodium, where it is soluble. While the SRE experiment was at very low burnup, based on the preceding temperature category, it does not appear that increasing burnup would necessary lead to greater fractional strontium releases. Therefore, the strontium release percentage is set at less than 5 % of the strontium inventory, with no apparent dependencies.

In noble metals group, the SRE incident provides another data point indicating that ruthenium release is apparently very small when below fuel melting temperatures. The eutectic melt regions may account for the liberation of some ruthenium from the fuel matrix. Therefore, the release percentage for ruthenium is set at less than 1 % of the ruthenium inventory.

In Ce, the fact that cerium was released from the damaged fuel pins at SRE (although in a small amount) is consistent with evidence from DFR (Dounreay Fast Reactor) and the RBCB (Run Beyond Cladding Breach) tests, and the FFTF (Fast Flux Test Facility) PIE (Post-Irradiation Examination). If cerium has migrated to the fuel/clad interface, eutectic melting in this temperature range may increase the release fraction. Therefore, the cerium release percentage is estimated to be up to 5 %.

In U & Pu, it is unlikely that the release of uranium or plutonium will increase in this temperature range, as melting will only occur in eutectic regions. Also, the following temperature ranges (1,100 °C ~ 1,300 °C) will show that uranium/plutonium releases at higher temperatures is small, which bound the estimate for this temperature range. Therefore, the release percentage is maintained at up to 0.1 %, with a dependence on burnup.

In lanthanides group, the SRE accident appears to confirm the findings of the RBCB tests at EBR-II (Experimental Breeder Reactor-II), where Zr was also found released from the failed fuel pins. It is difficult to assess how the low burnup of the SRE fuel affected the release of Zr/Nb. With higher burnup, more of the Zr inventory may have migrated, but lanthanide interaction with the cladding may have increased, resulting in less mobile lanthanides. Therefore, a likely conservative release percentage of up to 1 % of the lanthanide inventory is assigned. Even though lanthanide solubility

in sodium is generally low, the SRE data does indicate that some Zr/Nb can escape the fuel matrix.

3. Conclusions

The in-vessel STs of PGSFR will be estimated using the methodology of ANL-ART-38 report in addition to 4S methodology. The in-vessel STs are calculated through several phases: The inventory of each radionuclide is calculated by ORIGEN-2 code using the realistic burnup conditions. ST in the release from the core to primary sodium is calculated by using the assumption of ANL methodology. Lastly, ST in the release from the primary sodium to cover gas space is calculated by using equation and experimental materials. The leak rates from the cover gas region (0.5 %/day) and containment (0.1 %/day) will be used the design leak rates that are demonstrable under the design pressure and temperature limits of these boundaries.

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