

Evaluation on In-vessel Source Term in PGSFR (2015 Results)

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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 that of the Super-Safe, Small and Simple (4S) reactor [6], KAERI preliminarily evaluated the in-vessel ST using 4S 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 Assumptions of Radiological Consequence Analysis

In-vessel STs are estimated using a nonmechanistic and conservative methodology like that of 4S reactor. The fraction of fuel damaged is assumed 1 % (~ 243 fuel pins). The radioactivity inventory is assumed 110 % of the estimated inventory at the end of life that is very conservative but used to cover uncertainties in estimating the Fission Products (FPs) and activated primary sodium inventory. No fuel retention is assumed for the FPs group in the release into the primary sodium. The high primary sodium temperature (650 °C) is used for estimating the release fraction in the release into the cover gas region [6]. 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.

2.2 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 [6]. 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 as shown in the Fig. 1. The radiological inventory may include errors in excess of 10 % in magnitude by taking in account various uncertainties associated with fuel mass in the core. The nominal value of the radiological inventory is multiplied by a factor of 1.1 as an uncertainty margin to give the radiological inventory [6]. The inventory of the coolant is calculated by using the sodium mass (kg) and the specific activity (Ci/kg) [8].

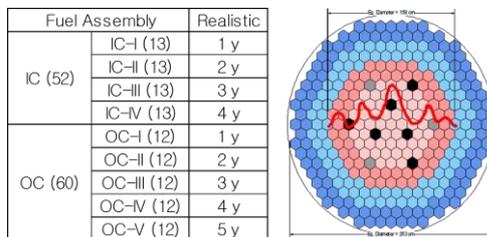


Fig. 1. The realistic burnup conditions in ORIGEN-2 code.

2.3 Release Path

Fig. 2 shows the radioactive release path used in the PGSFR ST evaluation. The release path includes transport of the fission products from the damaged fuel to the primary coolant, release of fission products and activated sodium from the primary coolant to the cover gas space, leakage from the cover gas space to the containment, and the leakage from the containment to the environment. The leak rate from the cover gas region will be 0.5 %/day. And, the leak rate from the containment will be 0.1 %/day as shown in Fig. 2. These design leak rates are demonstrable under the design pressure and temperature limits of these boundaries.

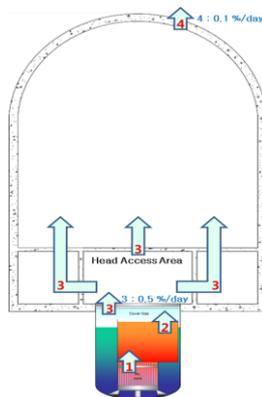


Fig. 2. Release path for ST evaluation in PGSFR.

2.4 Release from the Core to Primary Sodium

ST in the release from the core to primary sodium is calculated by using the assumption of 4S methodology [6]. Table I shows the release fraction from the core to primary sodium using for this calculation.

Table I: Release Fraction from the Core to Primary Sodium

Radionuclide Groups	Release Fraction (Core to Primary Sodium)
Noble Gases	0.01 (10^{-2})
Halogens	0.01 (10^{-2})
Alkali Metals	0.01 (10^{-2})
Te Group	0.01 (10^{-2})
Ba, Sr	0.01 (10^{-2})
Noble Metals	0.001 (10^{-3})
Ce Group	0.00001 (10^{-5})
Lanthanides	0.00001 (10^{-5})
Coolant	-

The main assumptions are as follows:

In noble gases group, the fission gas is retained mostly in the fuel for burnups less than 1~2 %. At higher burnups, passageways are created within the fuel allowing transport of the fission gas to the fuel pin gas plenum. At the end of life, the fraction of the fission gas residing in the gas plenum reaches 70 to 80 %. The remaining 20 to 30 % are retained in the fuel [9]. In this calculation, 100 % of the fission gas is assumed to be released instantaneously to the primary sodium on clad failure.

In halogens and alkali metals groups, while the formation of CsI is possible for both types of fuel, the possibility of having elemental I in the PGSFR fuel is made extremely remote by the presence of uranium metal and sodium (to form UI_3 and NaI). Results of the Run Beyond Cladding Breach (RBCB) experiments for sodium-bonded metal fuel show no measurable amount of I release [10]. CsI is less volatile than Cs and I. Therefore, its formation reduces the release fraction of both Cs and I. In this calculation, I is not retained in the fuel as UI_3 , 100 % of the Cs inventory is released from the fuel to the primary sodium as elemental Cs (No CsI is formed), and 100 % of the I is released from the fuel to the primary sodium.

In Te group, elements in this group interact with the fuel bond sodium to form Na_2X compounds, e.g., Na_2Te for Te. In this calculation, 100 % of the inventory in this group is involved in this reaction with the fuel bond sodium. The release of the bond sodium to the primary sodium occurs instantaneously at the time of fuel failure.

In Ba and Sr group, the melting points of Ba and Sr are higher than the peak fuel temperature estimated for Design Basis Accidents (DBAs) and most of the inventory will likely be retained in the fuel. In this calculation, 100 % of the inventory of this group is dissolved in the bond sodium and released.

In noble metals group, noble metals have melting points that are significantly higher than that of the metallic fuel. Elements in the group do not react substantially with sodium and have low solubility. In this calculation, a release fraction of 0.1 % is assumed in this analysis.

In Ce and lanthanides groups, the elements in these groups hardly react with sodium and have low solubility. Amount of plutonium dissolved into bond sodium during the life time (i.e., about 360 months) is less than 0.1 % of total plutonium amount. In this calculation, a release fraction at 1 % pin failure is estimated to be less than 0.001 % at the fuel end of life.

2.5 Release from the Primary Sodium to Cover Gas Space

ST in the release from the primary sodium to cover gas space is calculated by using the assumption of 4S methodology [6]. Table II shows the release fraction from the primary sodium to cover gas space using for this calculation.

Table II: Release Fraction from the Primary Sodium to Cover Gas Space

Radionuclide Groups	Release Fraction (Primary Sodium to Cover Gas Space)
Noble Gases	1
Halogens	$7.9 \cdot 10^{-6}$
Alkali Metals	$1.4 \cdot 10^{-4}$
Te Group	$3.7 \cdot 10^{-6}$
Ba, Sr	$3.7 \cdot 10^{-6}$
Noble Metals	$3.7 \cdot 10^{-6}$
Ce Group	$3.7 \cdot 10^{-6}$
Lanthanides	$3.7 \cdot 10^{-6}$
Coolant	$3.7 \cdot 10^{-6}$

The main assumptions and equations are as follows:

In noble gases group, noble gases hardly react with sodium. In this calculation, 100 % of the noble gases are assumed to be released to the cover gas region. In the primary sodium, the fraction of primary sodium that resides as vapor in the cover gas space (F_{Na}) is estimated using the sodium partial pressure at the 650 °C, the volume of the cover gas space, and the mass of the liquid sodium in the reactor vessel. F_{Na} is derived

using the following equation for the molar ratio of the sodium gas to sodium liquid.

$$F_{Na} = \frac{n_i^g}{n_i^l} = \frac{\frac{P_i^0}{RT} V_g}{\frac{\rho}{A_{Na}} V_l} \quad (1)$$

where, n_i^g is the sodium mole number in the cover gas (mol), n_i^l is the sodium mole number in the sodium (mol), P_i^0 is the saturated vapor pressure of the sodium (Pa), V_g is the cover gas volume (m^3), R is the ideal gas constant, T is the temperature (K), ρ is the sodium density (g/m^3), A_{Na} is the sodium atomic mass (g/mol), and V_l is the sodium volume (m^3). In this calculation, the constant values are as follows: V_g is $87.233 m^3$ [11], R is $8,314 J/kg \cdot kmol$ [12], A_{Na} is $23 kg/kmol$ [12], and V_l is $635.35 m^3$ [11].

The saturated vapor pressure of the sodium was estimated using the following equation [13].

$$P_i^0 = 1.01329 \times 10^5 \times e^{\left(18.832 - \frac{13113}{T} - 1.0948 \ln T + 1.9777 \times 10^{-4} T\right)} \quad (2)$$

The sodium density was estimated using the following equation [8].

$$\begin{aligned} \rho &= 1014 - 0.2357 T & T \leq 1,155 K \\ \rho &= 1016.8329 - 0.233937 T - 0.305 \cdot 10^{-5} T^2 & T > 1,155 K \end{aligned} \quad (3)$$

In halogens and alkali metals groups, the fraction of halogens and alkali metals released to the cover gas (F) is estimated using the following equation, where K_d is the gas-liquid equilibrium constant for halogens and alkali metals:

$$F = K_d \cdot F_{Na} \quad (4)$$

The release fractions of I and Cs are representative of the halogens and alkali metals groups. The gas-liquid equilibrium constant, K_d , for I and Cs, is based on the maximum value found in existing literature [14, 15]. I exists in the primary sodium as NaI, thus the gas-liquid equilibrium constant of NaI is used. Cs remains in its elemental form in the primary sodium [6]. The gas-liquid equilibrium constants of NaI and Cs are shown in equations (5) and (6) below:

$$NaI : K_d = 10^{\frac{592}{T} + 0.5 \log T - 1.79} \quad (5)$$

$$Cs : K_d = 10^{\frac{987}{T} + 0.490} \quad (6)$$

Fig. 3 shows the temperature dependency of the sodium mass fraction in the cover gas space using equation (1) and the release fraction for NaI and Cs in

the PGSFR reactor system as calculated using equations (4), (5), and (6). For the assumed primary sodium temperature of 650 °C, the release fractions from primary sodium to the cover gas space are $3.7 \cdot 10^{-6}$, $7.9 \cdot 10^{-6}$, and $1.4 \cdot 10^{-4}$ for sodium, NaI, and Cs.

In Te, Ba and Sr, noble metals, Ce, and lanthanides groups, elements in these groups have very small saturated vapor pressure compared with the sodium, halogens and alkali metals. In this calculation, the release fraction of these elements should be as low as that of sodium or lower ($3.7 \cdot 10^{-6}$).

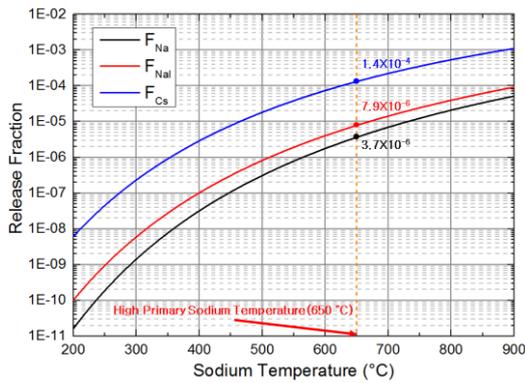


Fig. 3. Release fraction of sodium, NaI and Cs from coolant to cover gas.

Table III: Result of the In-vessel ST in PGSFR

Radionuclide Group	Elements	Mass (g)
Noble Gases	Xe	4.39496E+02
	Kr	4.62520E+01
Halogens	I	2.39730E-04
	Br	2.07996E-05
Alkali Metals	Cs	5.25273E-02
	Rb	5.90118E-03
Tellurium Group	Te	2.07916E-04
	Sb	2.35175E-05
	Se	2.61019E-05
Barium, Strontium	Ba	5.24106E-04
	Sr	4.23429E-04
Noble Metals	Ru	8.03944E-05
	Rh	1.89944E-05
	Pd	2.25723E-05
	Mo	1.25720E-04
	Tc	3.03901E-05
Lanthanides	Co	0.00000E+00
	La	4.77959E-07
	Zr	3.21490E-05
	Nd	1.40441E-06
	Eu	2.56746E-08
	Nb	2.06224E-08
	Pm	1.14719E-07

	Pr	4.33195E-07
	Sm	2.53532E-07
	Y	2.15475E-07
	Cm	3.07875E-12
	Am	1.13605E-10
Cerium Group	Ce	1.05652E-06
	Pu	5.78650E-06
	Np	1.38801E-07
	U	2.87778E-04
Coolant	Na	2.67460E-06

Table III shows the result of the in-vessel ST in PGSFR by using the assumption of 4S methodology.

In the leakage from the cover gas space to the containment, the design leakage rate of 0.5 %/day will be used. This leakage rate is conservative, because the used leak rate does not account for the possibility of blockage. The in-containment ST will be calculated by using the in-vessel ST and the design leakage rate. In the leakage from the containment to the environment, the design leakage rate of 0.1 %/day will be used. This leakage rate is also conservative.

3. Conclusions

The in-vessel STs of PGSFR are estimated using a nonmechanistic and conservative methodology like that of 4S reactor. The many assumptions and equations evaluated in 4S are used. The in-vessel STs are calculated through several phases: The inventory of each radionuclide is calculated by ORIGEN-2 code using the peak burnup conditions. The nominal value of the radiological inventory is multiplied by a factor of 1.1 as an uncertainty margin to give the radiological inventory. ST in the release from the core to primary sodium is calculated by using the assumption of 4S methodology. Lastly, ST in the release from the primary sodium to cover gas space is calculated by using the assumption of 4S methodology. 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.

REFERENCES

- [1] D. A. Powers, B. Clément, R. Denning, S. Ohno, and R. Zeyen, Advanced Sodium Fast Reactor Accident Source Terms: Research Needs, SAND2010-5506, 2010.
- [2] B. D. Middleton, E. J. Parma, T. J. Olivier, J. Phillips, and J. L. LaChance, The Development of a Realistic Source Term for Sodium-Cooled Fast Reactors: Assessment of Current Status and Future Needs, SAND2011-3404, 2011.
- [3] S. D. Suk and H. Y. Jeong, Review of SFR Severe Accident Source Term Studies, KAERI/AR-879/2011.
- [4] J. J. DiNunno, F. D. Anderson, R. E. Baker, and R. L. Waterfield, Calculation of Distance Factors for Power and

- Test Reactor Sites, Technical Information Document (TID)-14844, U. S. Atomic Energy Commission, 1962.
- [5] L. Soffer, S. B. Burson, C.M. Ferrell, R. Y. Lee, and J. N. Ridgely, Accident Source Terms for Light-Water Nuclear Power Plants, NUREG-1465, U. S. Nuclear Regulatory Commission, 1995.
- [6] 4S Safety Analysis, Toshiba Corporation, AFT-2009-000155 Rev. 000(0), 2009.
- [7] U. S. Nuclear Regulatory Commission, Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors, Regulatory Guide (RG) 1.183, 2000.
- [8] T. Fei and T. K. Kim, PGSFR Radiation Assessment, ANL-KAERI-SFR-14-25 Revision 0, January 26, 2015.
- [9] R. G. Pahl, et al., Steady-state Irradiation Testing of U-Pu-Zr Fuel to >18 at% Burnup, Proc. Int. Fast Reactor Safety Meeting, Snowbird, 1990, Vol. 4, American Nuclear Society, 1990.
- [10] R. G. Pahl, et al., The Characterization and Monitoring of metallic Fuel Breaches in EBR-II, International Conference on Fast Reactor Systems and Fuel Cycles, Kyoto (Japan), 1991.
- [11] 한인수, 2015 년도 PGSFR 안전해석용 기계분야 입력자료, SFR-IOC-M/S-15-003, April, 2015.
- [12] B. Doup, J. Lim, Q. Zhou, N. Karancevic, P. McMinn, S. J. Lee, and C. Y. Paik, Interim Report on ISFRA Code Development, FAI/15-0028, February, 2015.
- [13] J. K. Fink, et al., Thermophysical Properties of Sodium, ANL-CEN-RSD-1, 1971.
- [14] C. G. Allan, et al., Solubility and Deposition Behavior of Sodium Bromide and Sodium Iodine in Sodium/Stainless Steel Systems, TRG Report 2458(D), 1973.
- [15] B. D. Pollock, et al., Vaporization of Fission Product from Sodium, ANL-7520 Part-1, 1968.