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ECAR-5021 Source Term Estimates for SCO Micro-Reactor Designs

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3. Safety SSC Determination Document ID	N/A	
4. SSC ID	N/A	
5. Project No.	33421	
6. Engineering Job (EJ) No.	N/A	
7. Building	N/A	
8. Site Area	N/A	

9. Objective / Purpose

The purpose of this document is to provide source term estimates and the associated technical basis for Strategic Capabilities Office (SCO) microreactor designs that use TRISO fuel. This estimate can be used for relevant environmental and safety analyses that will be done as part of the project for all of the selected suppliers for a safety design strategy. This is a combined quantitative and qualitative analysis.

10. If revision, please state the reason and list sections and/or page being affected.

Revision 1: Remove SCO markings per John Mendenhall.

11. Conclusion / Recommendations

A source term is developed for a 10 MWt microreactor based on conservative scaling from 95% confidence estimates for a 600 MWT NGNP high temperature gas reactor (HTGR).

For normal operation releases, only the noble gases would be released via leakage. The short-term noble gas releases from Table 7 of this ECAR are a good estimate of the inventories in the helium assuming no cleanup system. A simple conservative leak rate of ~0.1%/day could be applied to the short-term noble gas release to provide an upper bound of normal effluents if necessary. If the design incorporates a cleanup system, releases would be much smaller.

The accident source term timing in HTGRs is different than other reactor systems. There is a short term release that is associated with the depressurization of the reactor system and cooldown of the core (aka depressurized conduction cooldown) allowing fission products in the helium of the reactor coolant system and a fraction of the fission products plated out on the

reactor coolant system surfaces to be released. Following the depressurization, the reactor core heats up under decay heat, transferring its heat through the large amount of graphite to the surface of the reactor vessel and then radiating the heat to the ultimate heat sink. During this long slow heat-up, fission products are released and this is termed the 'long term' release. Both releases are shown in Table 7 of this ECAR. The long-term release is very conservative because the temperatures expected in a 10 MWT microreactor are much lower than those in the 600 MWT NGNP HTGR from which the source term values are scaled.

Contents

APPENDIXES

PROJECT ROLES AND RESPONSIBILITIES

Project Role	Name	Organization	Pages Covered (if applicable)
Performer	David Petti	C600	All
Checker[a]	Chris Long	H373	All
Independent Reviewer[b]	Jason Andrus	H350	All
CUI Reviewer[c]	INL/EXT-21-65072	-----	-----
Manager[d]	Justin Coleman	C100	All
Requestor[ef]	Justin Coleman	C100	All
Nuclear Safety[f]	Troy Reiss	H350	All
Document Owner[f]	Justin Coleman	C100	All

Responsibilities:

a. Confirmation of completeness, mathematical accuracy, and correctness of data and appropriateness of assumptions.

b. Concurrence of method or approach. See definition, LWP-10106.

c. Concurrence with the document's markings in accordance with LWP-11202.

d. Concurrence of procedure compliance. Concurrence with method/approach and conclusion.

e. Authorizes the commencement of work of the engineering deliverable. See Appendix A.

f. Concurrence with the document's assumptions and input information. See definition of Acceptance, LWP-10200.

NOTE: *Delete or mark "N/A" for project roles not engaged. Include ALL personnel and their roles listed above in the eCR system. The list of the roles above is not all inclusive. If needed, the list can be extended or reduced.*

BACKGROUND

The Next Generation Nuclear Plant (NGNP) project had an ultimate objective to design and build a modular high temperature gas-cooled reactor (HTGR). Significant effort was dedicated to qualifying a fuel form for that reactor design, specifically TRISO-coated uranium oxycarbide (UCO) fuel particles under the Advanced Gas Reactor (AGR) program. As part of the qualification effort, work has been done to experimentally quantify using a statistically significant population of particles (a) the initial level of defective particles produced during fuel manufacture, and (b) the anticipated failure fractions under irradiation and under postulated accident conditions. The testing bounded the service conditions (temperature, burnup, fast fluence) that the fuel would be exposed to in a modular HTGR. Fission product releases were also measured during these experiments. The excellent performance obtained thus far in AGR fuel qualification program has been documented and submitted to the Nuclear Regulatory Commission (NRC) as a Topical Report (Reference 1) that current and future reactor designers can use to take credit for in their designs and safety analyses. The topical report is currently under review at NRC.

As the better than expected TRISO-coated particle fuel performance and fission product release data were being assessed as part of the AGR program, an effort was begun under the NGNP project to examine the impact of these results on the anticipated source term. The source term from the reactor core consists of three components: (1) releases from initially defective particles, (2) releases from incremental failures under irradiation and (3) releases associated with incremental failures under postulated accidents. The core graphite, the reactor coolant system and the reactor building (sometimes referred to as a confinement) all act as barriers to attenuate the source term from the particles. Graphite is assumed to not retain noble gases and halogen fission products. All other fission products will experience retention in the graphite and fuel matrix material under normal operation. Under accident conditions, less retention is expected as the reactor core heats up. The amount of retention (or the effectiveness of the barrier) depends on the specific fission product group, the temperature and time, based on historical HTGR fission product transport data. Some modest retention of fission products in the reactor coolant system and in the building prior to release to the environment is also accounted for in the analysis based on historical test data. A schematic of the different barriers is shown in Figure 1.

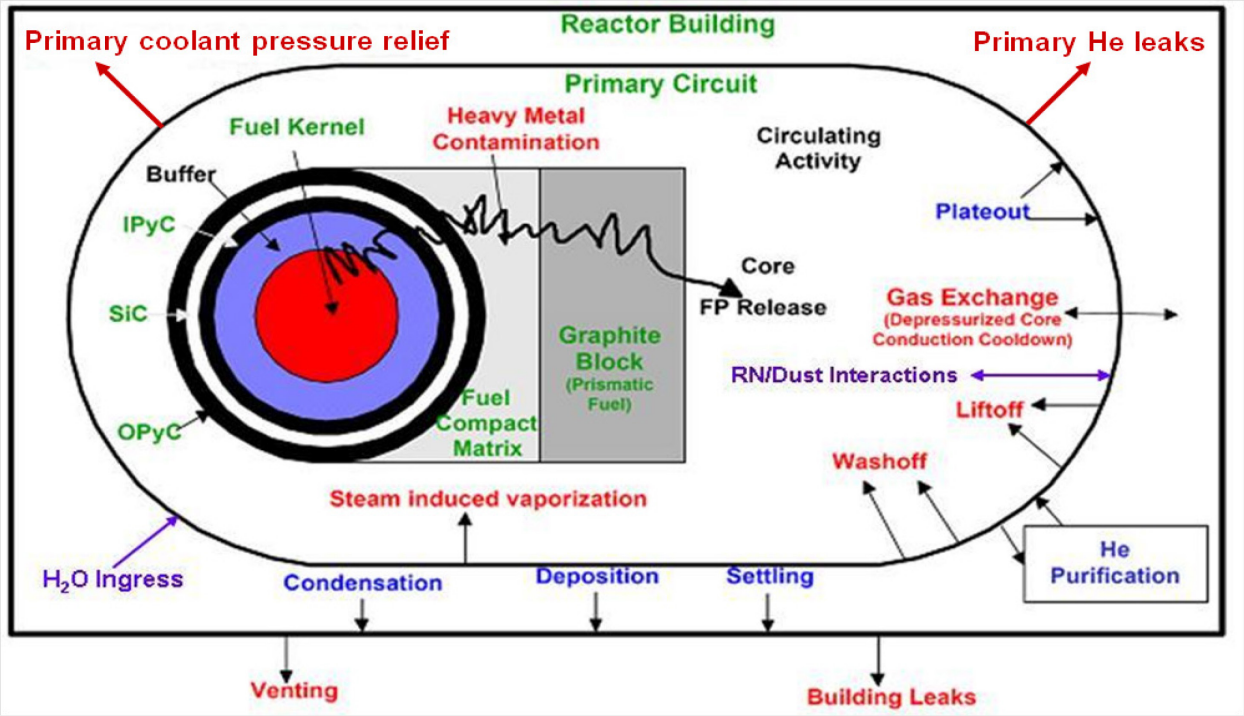

Figure 1. Schematic representing phenomena important in modeling fission product transport in an MHTGR.

The overall calculation of the source term is fairly complex as described in Reference 2.[1] Figure 2 shows the overall logic for establishing releases during normal operation and the logic for release under accident conditions is shown in Figure 3 (both figures are from Reference 2). The calculations start with establishing the level of defects and the level of assumed failures during normal operation and accident conditions (yellow boxes) as given by fuel fabrication and design specifications (purple boxes). Releases from the defective particles are calculated and for silver diffusion through intact particles is calculated (red boxes). Transport through the fuel matrix and graphite is calculated as well (red box). The retention of the barriers (core graphite, reactor coolant system, and reactor building (or confinement)) is accounted for (green boxes). The equations used to perform the calculation are given in Reference 2.

[1] Reference 2 is included as Appendix A to this document for the interested reader

Computer codes are usually used to calculate the source term for a prismatic HTGR as part of the safety analysis. A different approach using expert elicitation was used in Reference 2 based on a similar approach in light water reactor source term estimates.

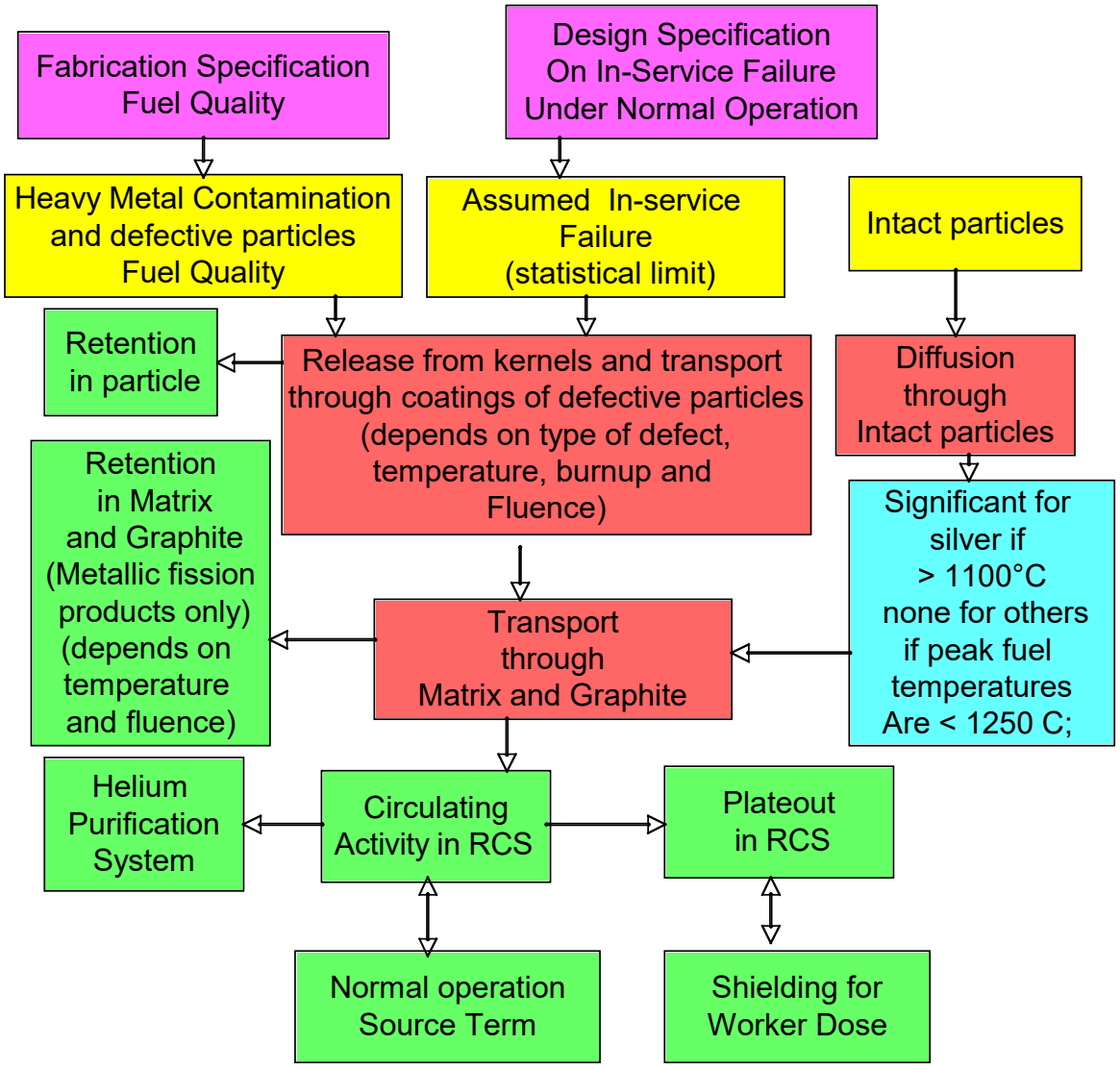

Figure 2. Logic diagram to describe MHTGR source term for normal operation

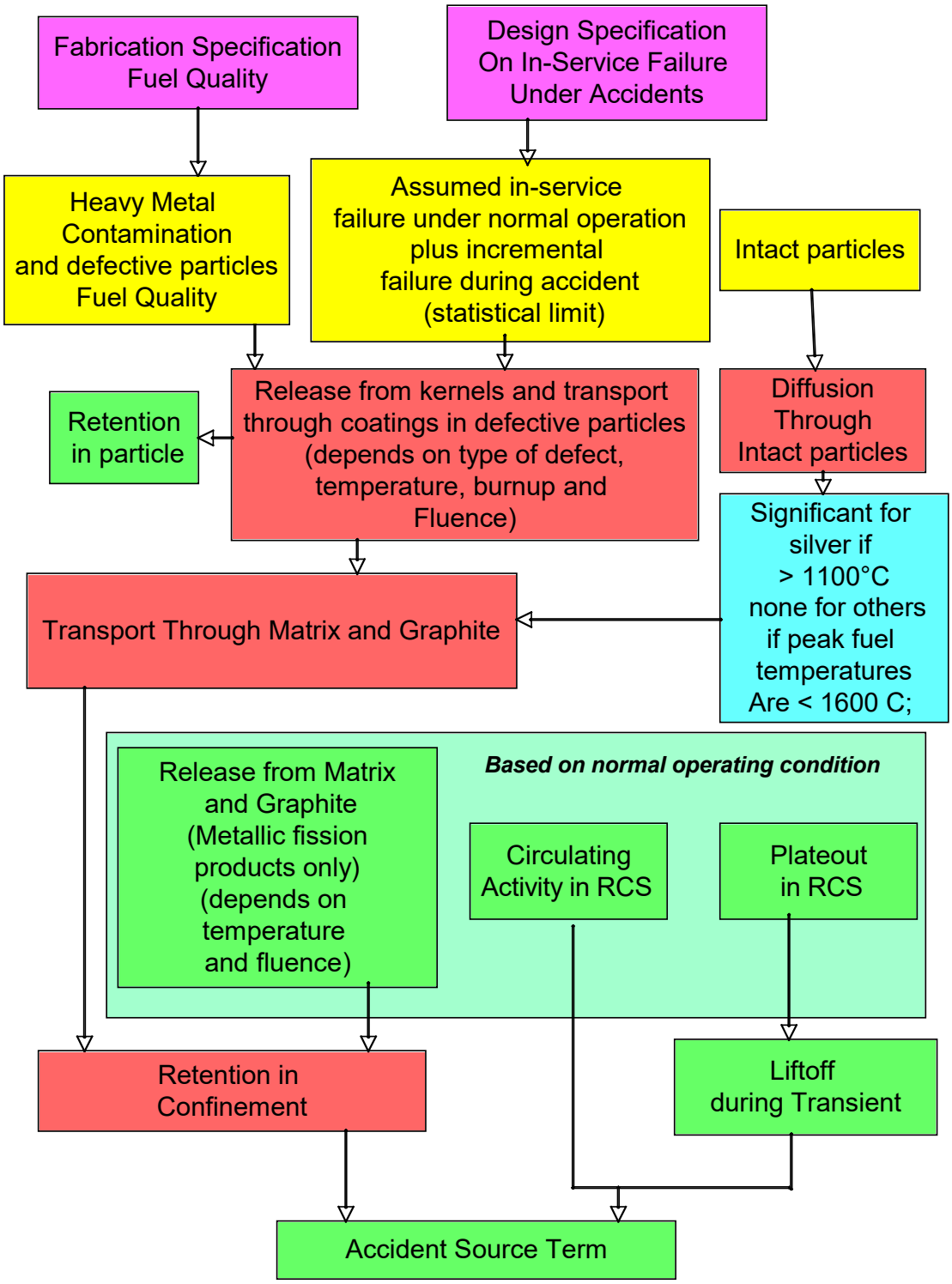

Figure 3 Logic Diagram for Accident Source Term for an MHTGR.

An expert panel was assembled to provide numerical estimates of the following technical parameters used to establish the source term:

- Initial levels of fuel defects: heavy metal contamination and SiC defects
- Failures during operation and accident conditions: incremental in-situ failures under irradiation and incremental in-situ failures under accident conditions
- Attenuation factors that reduce the transport (increase the holdup) of fission products in:
    - kernels of the coated particle,
    - intact particles,
    - graphite,
    - reactor coolant system and
    - reactor building.

  Those attenuation factors related to the reactor core and fuel are strong functions of temperature and time during normal operation and accidents. Attenuation in the reactor coolant system and building are a strong function of the environment (wet vs. dry). Thus, separate values were obtained for each of the relevant conditions.

Both best estimate (50% confidence) and conservative estimates (95% confidence) were provided for each of these technical parameters for each fission product of interest. Some values were informed by testing from the AGR fuel qualification program, and other values were obtained from historical HTGR testing data and safety analyses done on prismatic HTGR designs. The best estimate and conservative bounds were used to establish normal distributions of the technical parameters. Instead of stacking each of the 95% confidence values on top of each other, these distributions were sampled in a Monte Carlo simulation to establish 50% confidence, mean and 95% confidence estimates of the release to the environment under the postulated accident. The details of the source term produced by the expert panel using this approach are documented in Reference 2. Relevant results are reproduced here to compare directly to the source term for a microreactor.

As is commonly done in source term analysis in light water reactors, the fission products were grouped into classes (see Table 1) based on their similar chemical volatility in the gas reactor system. (These grouping are a little different than the traditional LWR groupings because of the difference in chemistry in gas reactors).

The characteristics isotopes in Table 1 conservatively bound the isotopes expected to contribute to off-site dose based on previous reactor designs and safety analyses performed by General Atomics (Reference 3).

Two different reactor designs and two different accident scenarios were evaluated. The designs were: (a) 600 MWt prismatic MHTGR with an outlet temperature of 700°C, and (b) 600 MWt prismatic MHTGR with an outlet temperature of 900°C. The accidents were: (a) a depressurized conduction cooldown (loss of helium from the reactor coolant system) and (b) moisture ingress event from a steam

generator tube rupture. This study provided an understanding of the effects of reactor outlet temperature and accident atmosphere (wet versus dry) on source terms. For the analysis here, the results from the 900°C outlet design and the depressurized conduction cooldown scenario will be used as the starting point. The designs of the microreactor by the different vendor teams are not final and the use of water in the reactor is not finalized. Thus, at this point, a moisture ingress event, such as a steam generator tube rupture, will not be considered. The original source term analysis (Reference 2) does include a source term for a moisture ingress event. The source term is not that different than the one used here. However, this can be revisited once the designs mature. Reactivity events are not risk dominant for modular HTGRs (Reference 3).

Table 1. Fission product classes.

Fission Product Class	Characteristic Nuclides
Noble Gases	Kr-85, Kr-88, Xe-133
I, Br, Te, Se	I-131, I-133, Te-132
Cs, Rb	Cs-134, Cs-137
Sr, Ba, Eu	Sr-90
Ag, Pd	Ag-110m, Ag-111
Sb	Sb-125
Mo, Ru, Rh, Tc	Ru-103
La, Ce	La-140, Ce-144
Pu, actinides	Pu-239

DESIGN OR TECHNICAL PARAMETER INPUT AND SOURCES

1. *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP).* EPRI, Palo Alto, CA: 2019. 3002015750.
2. David A. Petti, Richard R. Hobbins, Peter Lowry, Hans Gougar, “Representative Source Terms and The Influence of Reactor Attributes on Functional Containment in Modular High Temperature Gas-cooled Reactors,” *Nuclear Technology*, Vol. 184, p. 181-197, Nov. 2013.
3. Inamati, S., Parme, L. and Silady, F., “Probabilistic risk assessment of the modular high-temperature gas-cooled reactor,” GA-A-18930; CONF-870820-71993, July 1987.

RESULTS OF LITERATURE SEARCHES AND OTHER BACKGROUND DATA

N/A

ASSUMPTIONS

1. The source term for a 10 MWt microreactor is conservatively scaled from a 95% confidence source term for a 600 MWt HTGR that was performed for the NGNP project.

2. The peak temperatures of the fuel in the 600 MWt HTGR (1200-1300 C) and the associated irradiation testing in the AGR program (< 1400 C) used to develop the source term under normal operation bound that expected in the 10 MWt microreactor.
3. The peak temperatures of the fuel under accident conditions (1600 C) bound that expected in the 10 MWt microreactor, where peak temperatures are not expected to exceed those in normal operation.

COMPUTER CODE VALIDATION:

This section is not applicable. This analysis is a summary of scoping calculations for the core inventory and source term at the proposed microreactor, but does not derive controls for these operations. Validation of the computer code and of the models used for these scoping calculations is not necessary. When controls are eventually derived for these operations, a separate source term evaluation will be created and validation will be included there. The calculations uses by this ECAR were done as scoping calculations using non-quality assured computer codes to develop the original core inventory and source term for the Next Generation Nuclear Power (NGNP) program. They are suitable for conceptual design[4].

DISCUSSION/ANALYSIS

The approach for this source term estimate is to scale the 95% confidence results for the MHTGR to the SCO micro-reactor designs. The major conservatisms to this approach are:

(1) AGR testing performed after the work in Reference 2 was complete is showing even larger safety margin in terms of the level of failure anticipated under both irradiation and accident conditions and in terms of the actual fission product release.
(2) The peak temperature of the fuel under normal operation in the 600 MWt HTGR (1200-1300 C) and that used in the AGR testing (1400 C max) bounds that expected in the microreactor where peak temperatures may be in the range of 1000-1200 C.
(3) The accident thermal response of a microreactor will be much more benign than in the larger MHTGRs assumed in the analysis (Analyses of a microreactor show a cooldown upon loss of the helium because the decay heat is easily dissipated in the high surface- to-volume ratio core, whereas in MHTGRs peak temperatures slowly increase as the decay heat is absorbed in the large amount of graphite in the core. Peak fuel temperatures for the MHTGR can reach ~1600°C about 24 to 48 hours after initiation of the event depending on the details of the design. By contrast, microreactor peak temperatures in the accident may be no higher than peak temperatures under normal operation).
(4) Burnup in the microreactor is much less than in the MHTGR. Simple scaling by power and not accounting for the reduced burnup will conservatively overpredict the inventory of long-lived safety-significant isotopes such as Cs-137 and Sr-90.

The much lower power of the microreactor (10 MW(t) vs. 600 MW(t)) results in a much smaller fission product inventory in the reactor. The resultant fission product inventories are a factor of 60 smaller in the microreactor as shown in Table 2.

MHTGR TRISO-coated particle fuel has strict limits on the level of fabrication defects to limit I-131 release (the risk dominant isotope) under accidents to meet the EPA protection action guides and thus not require public evacuation after an accident. The values used in the NGNP[4] analysis are shown in Table 3. Heavy metal contamination, defined as the level of uranium outside of intact SiC, is a key fabrication defect. For the microreactor using TRISO fuel with a much smaller fission product inventory and much more benign accidents, the value of heavy metal contamination was selected to be 1E-04 instead of the 2E-05 95% confidence value in Table 3. The value of 1E-04 is 5x larger than the value used in the NGNP analysis. Scaling on power would enable an increase of 60x, however the site boundary is much closer to the reactor (300 m in NGNP to perhaps 50 m for the microreactor which would decrease dispersion of the radioactive plume. This would imply an overall increase of 10x (=60/6) could be appropriate. Instead a value of 5x was selected.) It is considered preliminary and will have to be validated with actual dose calculations later. This value is completely adequate for microreactor applications and can enable more economic fuel fabrication. Reducing the amount of fuel that must be destructively examined in quality control tests to demonstrate the level of heavy metal contamination will reduce fuel fabrication costs.

Table 2. Modular HTGR and Microreactor Initial core fission product inventories, in Curies.

Fission Product Class	Characteristic Nuclide	Inventory – 600 MW(t) Prismatic	Inventory – 10 MW(t) Microreactor
Noble Gases	Xe-133	3.63E+07	6.05E+05
	Kr-85	1.90E+05	3.17E+03
	Kr-88	1.85E+07	3.08E+05
I, Br, Te, Se	I-131	2.00E+07	3.33E+05
	I-133	3.60E+07	6.00E+05
	Te-132	2.71E+07	4.52E+05
Cs, Rb	Cs-137	1.69E+06	2.82E+04
	Cs-134	1.90E+06	3.17E+04
Sr, Ba, Eu	Sr-90	1.69E+06	2.82E+04
Ag, Pd	Ag-110m	2.81E+04	4.68E+02
	Ag-111	2.96E+06	4.93E+04
Sb	Sb-125	2.35E+05	3.92E+03
Mo, Ru, Rh, Tc	Ru-103	3.61E+07	6.02E+05
La, Ce groups	Ce-144	2.33E+07	3.88E+05
	La-140	3.27E+07	5.45E+05
Pu, actinides	Pu-239	4.66E+03	7.77E+01

The impact of this factor of 5 increase in heavy metal contamination was evaluated through the equations used to calculate the source term delineated in Reference 2. A straight scaling of 5x on the releases was conservative for all fission product groups except Cs, Sr, Sb and Ag. For Cs, Sr and Sb, no increase in overall release was calculated because the release was dominated by particles with SiC defects and the strong attenuation of these fission products in graphite in the core; heavy metal contamination was a small part of the overall source term for these fission product groups. It is also well known that silver is released from intact TRISO fuel particles. Changes in the level of heavy metal contamination do not affect the overall release of silver. (Silver is not expected to be a radiological concern but is included for completeness.)

Table 3. Fuel defect fractions.

Fission Product Class	Fabrication			
	Fraction Heavy Metal Contamination		Fraction SiC Coating Defects	
Confidence Limit	50%	95%	50%	95%
Noble Gases	1E-5	2E-5	NA	NA
I, Br, Se, Te	1E-5	2E-5	NA	NA
Cs, Rb	1E-5	2E-5	1E-5	3E-5
Sr, Ba, Eu	1E-5	2E-5	1E-5	3E-5
Ag, Pd	1E-5	2E-5	1E-5	3E-5
Sb,	1E-5	2E-5	1E-5	3E-5
Mo, Ru, Rh, Tc	1E-5	2E-5	1E-5	3E-5
La, Ce	1E-5	2E-5	1E-5	3E-5
Pu, actinides	1E-5	2E-5	1E-5	3E-5

All other assumed failure fractions under irradiation and accident conditions remain the same as in the original analysis as shown in Tables 4 and 5. As noted earlier, the values for failures under normal operation are conservative relative to current testing in the AGR program by a factor of 10 to 40. The incremental failures under accidents are representative of the data from the AGR program generated to date. No failures have actually been seen in the accident condition testing; the values developed by the AGR program are a statistical estimate at 95% confidence for zero failures in the particle population that has been tested to date.

Table 4. Prismatic in-service fuel failure fractions under normal operations.

Fission Product Class	In-Service Failure Fraction for 900°C ROT	
Confidence Limit	50%	95%
Noble Gases	1.4E-05	7.0E-05
I, Br, Se, Te	1.4E-05	7.0E-05
Cs, Rb	2.1E-04	1.05E-03
Sr, Ba, Eu	2.1E-04	1.05E-03
Ag, Pd	2.1E-04	1.05E-03
Sb,	2.1E-04	1.05E-03
Mo, Ru, Rh, Tc	2.1E-04	1.05E-03
La, Ce	2.1E-04	1.05E-03
Pu, actinides	2.1E-04	1.05E-03

Table 5. Prismatic in-service fuel failure fractions under depressurized conduction cooldown accident.

Fission Product Class	Accident Release	
	Incremental Failure-Accident	
Confidence Limit	50%	95%
Noble Gases	3E-5	8E-5
I, Br, Se, Te	3E-5	8E-5
Cs, Rb	3E-5	8E-5
Sr, Ba, Eu	3E-5	8E-5
Ag, Pd	3E-5	8E-5
Sb	3E-5	8E-5
Mo, Ru, Rh, Tc	3E-5	8E-5
La, Ce	3E-5	8E-5
Pu, actinides	3E-5	8E-5

The inventory of risk significant isotopes in key parts of the reactor system is shown in Table 6 for both the 600 MWt NGNP prismatic design and a 10 MW(t) microreactor scaled appropriately. The cesium and strontium (and other lower volatility fission products) are bound to the graphite and metal surfaces of the helium pressure boundary. Iodine is plated out in the coolest parts of the helium pressure boundary. The helium coolant is expected to contain only noble gases (e.g., Kr-85, Kr-88, Xe-133, and other shorter-lived noble gas isotopes). Halogen and other metallic fission products will also plate out on cooler parts of the reactor coolant system.

Table 6. Mean values for I-131, Cs-137, and Sr-90 inventories (curies) released to the helium pressure boundary and retained in the fuel matrix and graphite for the NGNP 600 MWt design and the 10 MWt microreactor.

Reactor Design Configuration	I-131, Curies		Cs-137, Curies		Sr-90, Curies	
	In Fuel Matrix and Graphite	In Helium Pressure Boundary	In Fuel Matrix and Graphite	In Helium Pressure Boundary	In Fuel Matrix and Graphite	In Helium Pressure Boundary
600 MW(t) Prismatic 900°C ROT	Nil	74	226	254	5680	31
10 MW(t) microreactor	Nil	6.2	3.76	0.42	94.7	0.52

The accident source term timing in HTGRs is different than other reactor systems. There is a short term release that is associated with the depressurization of the reactor system and cooldown of the core (aka depressurized conduction cooldown) allowing fission products in the helium of the reactor coolant system and a fraction of the fission products plated out on the reactor coolant system surfaces to be

released. Following the depressurization, the reactor core in the MHTGR heats up under decay heat, transferring its heat through the large amount of graphite to the surface of the reactor vessel and then radiating the heat to the ultimate heat sink. During this long slow heatup fission products are released and this is termed the ‘long term’ release. For the long-term release, the core is assumed to heat up for 50 hours, after which time a release of nuclides is assumed for the next 40 hours. The short term release is assumed to be prompt following the helium release from the depressurization of MHTGR piping.

The resultant accident source terms for a 600 MWt NGNP reactor and a 10 MWt microreactor with the appropriate scaling factors are shown in Table 7. Most of the fission product classes are scaled down by a factor of 60 based on the power ratio between two types and the linear relationship between power and fission product inventory. The fission product classes of iodine, noble gases and tellurium and the lower volatile fission product groups are scaled by a factor of 12 (a factor of 60 reduction in power level and a factor of 5 increase in allowable heavy metal contamination compared to that used in the NGNP analysis). For normal operation releases, only the noble gases would be released via leakage. The short-term noble gas releases in Table 7 are a good estimate of the inventories in the helium assuming no cleanup system. A simple conservative leak rate of ~0.1%/day could be applied to the short-term noble gas releases to provide an upper bound of normal effluents if necessary. If a design uses a clean- up system, releases would be much smaller.

Table 7. Accident source term for 600 MWt NGNP reactor design and 10 MWt microreactor appropriately scaled

	600 MWt, 900°C reactor outlet temperature; 2E-05 heavy metal contamination; 95% confidence source term		**10 MWt microreactor; 1E-04 heavy metal contamination; source term**	
	Short Term (Ci)	**Long Term (Ci)**	**Short Term (Ci)**	**Long Term (Ci)**
Xe-133	1.04E+02	1.94E+02	8.67E+00	1.62E+01
Kr-85	5.34E-01	1.40E+00	4.45E-02	1.17E-01
Kr-88	5.32E+01	5.51E-04	4.43E+00	4.59E-05
I-131	1.57E+00	2.46E+01	1.31E-01	2.05E+00
I-133	2.61E+00	1.08E+01	2.18E-01	9.00E-01
Te-132	2.05E+00	2.66E+01	1.71E-01	2.22E+00
Cs-137	4.57E+01	7.93E+00	7.62E-01	1.32E-01
Cs-134	8.04E+00	9.34E+00	1.34E-01	1.56E-01
Sr-90	4.09E+00	4.33E+00	6.82E-02	7.22E-02
Ag-110m	4.61E+00	7.75E+00	7.68E-02	1.29E-01
Ag-111	1.14E+02	6.33E+02	1.90E+00	1.06E+01
Sb-125	3.94E-01	1.25E-01	6.57E-03	2.08E-03
Ru-103	5.50E-02	8.78E+00	4.58E-03	7.32E-01
Ce-144	6.56E-01	6.39E-01	5.47E-02	5.33E-02
La-140	5.03E-02	3.81E-01	4.19E-03	3.18E-02
Pu-239	4.90E-06	1.06E-05	4.08E-07	8.83E-07

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1. *Uranium Oxycarbide (UCO) Tristructural Isotropic (TRISO) Coated Particle Fuel Performance: Topical Report EPRI-AR-1(NP).* EPRI, Palo Alto, CA: 2019. 3002015750.
2. David A. Petti, Richard R. Hobbins, Peter Lowry, Hans Gougar, “Representative Source Terms and The Influence of Reactor Attributes on Functional Containment in Modular High Temperature Gas-cooled Reactors,” *Nuclear Technology*, Vol. 184, p. 181-197, Nov. 2013.
3. Inamati, S., Parme, L. and Silady, F., “Probabilistic risk assessment of the modular high-temperature gas-cooled reactor,” GA-A-18930; CONF-870820-71993, July 1987.
4. *Engineering Services For The Next Generation Nuclear Plant (NGNP) With Hydrogen Production NGNP Contamination Control Study 911117 Revision 0*. General Atomics, April 2008.

Appendix A

Embedded is copy of the following journal article:

David A. Petti, Richard R. Hobbins, Peter Lowry, Hans Gougar, “Representative Source Terms and The Influence of Reactor Attributes on Functional Containment in Modular High Temperature Gas-cooled Reactors,” *Nuclear Technology*, Vol. 184, p. 181-197, Nov. 2013.

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Representative Source Terms and the Influence of Reactor Attributes on Functional Containment in Modular High-Temperature Gas-Cooled Reactors

D. A. Petti, R. R. Hobbins, P. Lowry & H. Gougar

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