August 10, 2011

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NRC Project #0748

U.S. Nuclear Regulatory Commission
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SUBJECT: Contract No. DE-AC07-05ID14517 — Next Generation Nuclear Plant Project Submittal — Response to Nuclear Regulatory Commission Request for Additional Information Letter No. 002 Regarding Next Generation Nuclear Plant Project Fuel Qualification and Mechanistic Source Terms — NRC Project # 0748

Consistent with the actions identified in "NGNP Licensing Strategy – Report to Congress," dated August 2008, the purpose of this letter is to submit responses to the subject U.S. Nuclear Regulatory Commission (NRC) Request for Additional Information regarding Next Generation Nuclear Plant (NGNP) Project white papers. The attachment contains NGNP Project's responses for those Request for Additional Information (RAIs) received in NRC RAI Letter Number 002 (Request for Additional Information No. 5772 Revision 0), dated June 7, 2011. It should be noted that a single response is provided for those RAIs that are similar in nature for both the Fuel Qualification (FQ) and Mechanistic Source Terms (MST) white papers. The associated FQ and MST RAI numbers, as assigned by the NRC, are identified on headings of the RAI descriptions and responses.

The NRC licensing process encourages early interactions to identify and resolve policy, regulatory, and key technical issues related to the proposed facility. Conducting effective interactions with the NRC is a critical part of the NGNP licensing strategy because the early resolution of issues can significantly impact the preparation of an acceptable license application, the subsequent application review schedule, and the ultimate deployment of the NGNP. This NGNP response to the NRC's RAIs represents one in a series of submittals that address priority licensing topics related to establishing High Temperature Gas Reactor (HTGR) regulatory requirements using the process outlined in the Licensing Strategy.

Following NRC Staff review of these RAI responses, and pending resolution of associated follow-on questions, the NGNP Project requests that the NRC provide feedback and documentation of its review in a format that will facilitate resolution of key design, safety, and licensing issues on the topics of fuel qualification and mechanistic source terms that can be used as a firm basis for the preparation of future HTGR license application(s).

If you have any questions, please contact me at (208) 526-6063 or James Kinsey, Director, NGNP Regulatory Affairs at (208) 569-6751.

Sincerely,

David Petti for

Greg Gibbs, Project Director
Next Generation Nuclear Plant Project

JK:CN

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MRR

Attachment:

1. NGNP Response to NRC RAI No. 5772, Revision 0.

References:

a) "Next Generation Nuclear Plant – Fuel Qualification White Paper," July 21, 2010, CCN 221270
b) "Next Generation Nuclear Plant - Mechanistic Source Terms White Paper," July 21, 2010, CCN 221271
c) NRC RAI Letter Number 002 (Request for Additional Information No. 5772 Revision 0, June 7, 2011
d) Idaho National Laboratory, Next Generation Nuclear Plant Project Submittal, "Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification and Mechanistic Source Terms White Papers," May 3, 2011, CCN 223977

cc:

DOE-HQ
M. A. Feltus
T. J. O'Connor
C. J. Sink
J. Zamore

DOE-ID
M. L. Adams, (w/o Enc.)
B. Blythe
P. K. Bowers, (w/o Enc.)
C. P. Fineman
R. V. Furstenau
G. R. McClellan
S. M. Olson, (w/o Enc.)

INL
J. Alvarez, (w/o Enc.)
J. J. Grossenbacher, (w/o Enc.)
D. J. Hill, (w/o Enc.)
D. M. Storms, (w/o Enc.)
J. M. Welch, (w/o Enc.)

NRC
S. Basu
D. Carlson
M. E. Mayfield
T. A. Kevern
J. F. Williams

Attachment

NGNP Response to NRC RAI Letter No.002

Request for Additional Information No. 5772 Revision 0

Acronyms

AGR	Advanced Gas Reactor
ALARA	as low as reasonably achievable
AOO	Anticipated Operational Occurrences
ASME	American Society of Mechanical Engineers
ATR	Advanced Test Reactor
ATWS	anticipated transient without scram
AVR	Arbeitsgemeinschaft Versuchsreaktor
BDBE	beyond design basis event
BE	best estimate
BEAU	best estimate and uncertainty
CANDU	Canadian Deuterium Uranium
CFD	computational fluid dynamic
CFP	coated fuel particle
CSNI	Committee on the Safety of Nuclear Installations
CRADA	Cooperative Research and Development Agreement
DBA	design basis accident
DBE	design basis event
DIDO	di-deuterium oxide
DLOFC	depressurized loss of forced cooling
DOE	Department of Energy
DPA	displacements-per-target-atom
DTF	designed-to-fail
EAB	exclusion area boundary
EPRI	Electric Power Research Institute
F-C	frequency-consequence
FIMA	fissions per initial metal atom
FQ	fuel qualification
FRJ2	Forschungsreaktor Jülich-Reactor 2
FSAR	Final Safety Analysis Report
FSV	Fort St. Vrain
FZJ	Forschungszentrum Jülich

GA	General Atomics
HFR	High-Flux Reactor
HGC	helium gas circulator
HPB	helium pressure boundary
HPS	Helium Purification System
HTGR	high temperature gas-cooled reactor
HTR-PM	High Temperature Reactor Power Module
HTS	heat transport system
HTTR	High Temperature Engineering Test Reactor
HVAC	heating, ventilating, and air conditioning
IAEA	International Atomic Energy Agency
IMGA	irradiated microsphere gamma analyzer
INL	Idaho National Laboratory
IPS	Investment Protection System
IPyC	inner pyrolytic carbon
ISI	in-service inspection
ITAACs	inspections, tests, analyses, and acceptance criteria
ITP	initial test program
JAEA	Japan Atomic Energy Agency
LBE	licensing basis event
LCO	limiting condition of operation
LEU	low-enriched uranium
LOFC	loss of forced cooling
LWR	light water reactor
MHR	modular helium reactor
MHTGR	modular high temperature gas-cooled reactor
MST	mechanistic source term
MTR	materials test reactor
MTU	metric tons uranium
NDE	nondestructive examination
NGNP	Next Generation Nuclear Plant
NRC	Nuclear Regulatory Commission
OECD	Organization for Economic Co-operation and Development

OPyC	outer pyrolytic carbon
ORNL	Oak Ridge National Laboratory
P/F	power-to-flow
PAG	Protective Action Guide
PBMR	Pebble Bed Modular Reactor
PCRV	prestressed concrete reactor vessel
PHP	Process Heat Plant
PIE	post irradiation examination
PIRT	phenomena identification and ranking table
PRA	probabilistic risk assessment
PSID	Preliminary Safety Information Document
PyC	pyrolytic carbon
QC	quality control
QHO	quantitative health objective
R&D	research and development
R/B	release-to-birth
RAI	request for additional information
RCCS	reactivity cavity cooling system
RCS	reactor control system
RPS	reactor protection system
SC-MHR	steam cycle modular helium reactor
SCS	shutdown cooling system
SG	steam generator
SHE	simple homogenous experiment
SL	safety limit
SOARCA	state-of-the-art reactor consequence analyses
SR	surveillance requirement
SRM	staff requirements memorandum
SSCs	structures, systems, and components
STS	standard technical specifications
TAMU	Texas A&M University
TEDE	total effective dose equivalent
TEV	technical evaluation

THTR	thorium high temperature reactor
TLRC	top-level regulatory criteria
TRISO	tristructural–isotropic
UPS	uninterruptable power supply
VHTR	very high temperature reactor

RAI FQ-1/MST-1: For (a) representative prismatic-block *and pebble-bed* NGNP designs, as well as (b) the recently completed AGR-1 irradiation *and subsequent ongoing and planned fuel irradiations* in the Advanced Test Reactor (ATR) materials test reactor (MTR), *and (c) the past tri-structural isotropic (TRISO) fuel irradiations in the Forschungsreaktor Jülich 2 (FRJ2, Jülich Research Reactor 2, also called the Di-Deuterium Oxide (DIDO) reactor by its British designers), High-Flux Reactor (HFR), and Siloe MTRs*, please provide the following calculated quantities presented as functions of total burnup and irradiation time:

i. the changing inventories of fissionable nuclides that contribute significantly to total fuel burnup (i.e., U-235, Pu-239, Pu-241), associated nuclide-specific fission rates, and *plutonium burnup versus total burnup (expressed in fissions per initial metal atom, i.e., % FIMA)*, and

ii. the resulting production and inventories of chemical elements that can potentially affect TRISO-coated fuel particle (CFP) performance, including palladium, rare earths, and silver.

Comments: This RAI incorporates and clarifies the first of two earlier information requests transmitted to INL in April 2010 and also adds item (c) regarding the past MTR irradiations of German TRISO fuel. The *italicized* items above are thus not addressed in TEV-1022, the INL technical report issued in partial response to those earlier requests. Note that certain pieces of the requested "item (c)" information may be found in two German documents[1, 2] known to the requester.

When high-temperature gas-cooled reactor (HTGR) fuel qualification irradiations are performed in MTRs, consideration must be given to how differences between the HTGR and MTR neutron energy spectra could lead to differences in fuel integrity and retentiveness. Such considerations generally include ensuring that the HTGR fuel design values of fast-neutron fluence and total burnup are enveloped by those achieved in the MTR irradiations. However, especially for low-enriched uranium (LEU, i.e., < 20% U-235) fuels, it is also important to evaluate how the neutron spectral differences affect uranium-to-plutonium conversion factors, nuclide-specific (U-235/Pu-239/Pu-241) fission rates and burnup, and the produced inventories of chemical elements that can affect fuel performance. The following observations bear noting in this context:

- The different fissionable nuclides (mainly U-235, Pu-239, and Pu-241) that undergo fission in LEU fuel have very different yields of certain fission products that can deleteriously affect the integrity and retentiveness of TRISO fuel particles. In particular, the fission yields of silver and palladium and various rare earth elements are many times higher from plutonium fission than from U-235 fission. Therefore, the total production of these fission products may be more a function of plutonium burnup than total burnup.

- Plutonium fission generally accounts for a large and variable fraction of the total burnup in high-burnup LEU fuels. For a given initial uranium enrichment and total fuel burnup, the magnitude of the plutonium fission fraction will vary with changes in the neutron energy spectrum. An HTGR spectrum may convert more uranium to plutonium than the softer spectra in water-cooled MTRs like

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[1] W. Kühnlein, *HFR-K5 und HFR-K6 Spaltproduktinventare und Abbrand (HFR-K5 and HFR-K6 Fission Product Inventories and Burnup)*, FZJ Hausmitteilung (Memorandum) 13.03.2003.

[2] R. Schröder, W. Kühnlein, H. Dahmen, *FRJ2-K15 Gammaspektrometrie an Brennelementen und Kalotten: Nachrechnung des Spaltproduktinventars (FRJ2-K15 Gamma Spectrometry on Fuel Elements and Capsules: Fission Product Inventory Calculations)*, FZJ Technische Notiz IWE-TN-15/94, April 1994.

the ATR and FRJ2 (DIDO). Furthermore, for a given content of plutonium in relation to U-235, the hotter thermal neutron spectrum in an HTGR, which typically peaks near the 0.3 eV fission resonances of Pu-239 and Pu-241, will more strongly favor plutonium fission over U-235 fission.

- It is widely noted that palladium and various rare earth fission products can have deleterious effects on particle coating integrity.[3] The effects of palladium have been summarized as follows: "Fission product palladium is known to attack SiC at localized reaction sites. These interactions have been the subject of extensive study. In high burnup LEU fuels, 25 to 50 times more Pd is produced than in either high burnup HEU fuels or LEU low burnup fuels because of the large fraction of fissions from Pu that are expected at high burnup. As a result, the potential for Pd attack of the SiC could be higher in LEU high burnup fuels like that proposed for NGNP. A review of the international database shows no strong dependence on burnup or the composition of the kernel, although theoretically this could be important."[4]

- It is also widely noted that silver diffuses readily through SiC at moderately high fuel operating temperatures. In the past, researchers have hypothesized that the cumulative effects of silver diffusion could alter the SiC grain boundaries. For example: "In the part played by silver it is not clear whether the release is determined by an independent diffusion process or whether silver and palladium first widen the SiC grain boundaries and can be regarded as precursors of SiC damage."[5] One could further hypothesize that the effects of silver diffusion on SiC grain boundaries could also increase grain boundary diffusion of Cs.

- Initial information needed for evaluating the effects of different neutron energy spectra in MTRs versus HTGRs would include the following calculated or measured quantities as functions of total burnup and irradiation time: (a) plutonium burnup and (b) inventories of palladium, selected rare earth fission products, and silver.

- To achieve more representative (or conservative) fission product compositions in future high-burnup irradiations of TRISO fuel in MTRs, one could increase the plutonium burnup fractions by doing some combination of the following:

  - Reducing the TRISO fuel's initial enrichment

  - Hardening the MTR's thermal neutron spectrum

  - Increasing the MTR's epithermal neutron spectrum

  - Replacing some UO_2/UCO in the fuel kernels with PuO_2/PuCO.

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[3] *R. Morris, D. Petti, D. Powers, B. Boyack, TRISO Coated Particle Fuel Phenomena Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents, NUREG/CR-6844, Volumes 1-3, July 2004.*

[4] *D. Petti, J. Maki, The Challenges Associated with High Burnup and High Temperature for UO_2 TRISO Coated Particle Fuel, MIT NGNP Symposium, INL/CON-05-00038, February 2005.*

[5] *W. Schenk, D. Pitzer, H. Nabielek, Fission Product Release Profiles from Spherical HTR Fuel Elements at Accident Temperatures, Jül-2234 (quoting from page 118), September 1988.*

It is noted that the NRC staff first described this technical issue in the June 2002 edition of the Draft NRC Advanced Reactor Research Plan[6] and again in later editions of that and related NRC documents. Quoting from page 48:

> *"Physics of TRISO fuel irradiation in test reactors versus HTGRs:*
>
> *The extensive use of various test reactors for the irradiation testing of HTGR TRISO fuels raises questions about the nonprototypicality of the neutron energy spectra, accelerated fuel burnup rates, and fuel temperature histories in the test reactors. Reactor-specific calculations of neutron fluxes and nuclide generation, depletion, and decay should therefore be performed to provide a basis for analyzing the sensitivity of computed fluences and fuel nuclide inventories to the neutronic differences between the test reactors and HTGRs. Of interest are the potential effects of such differences on TRISO fuel performance (i.e., fission product retention) under normal and accident conditions. Such differences therefore include the differences in irradiation temperature histories, burnup rates, and neutron energy spectra that result in different neutron fluences, different rates of plutonium production and plutonium fission versus uranium fission, and, thus, different yields of important fission products. It is known, for example, that U-235 and Pu-239 give substantially different yields of various fission products that potentially affect TRISO fuel performance. (Note: This nuclear analysis issue relates directly to fuel analysis issues described in Section IV.3.2.)"*

Response FQ-1/MST-1:

The information requested in item (i) has been provided in TEV-1022 [1] for (a) a representative prismatic design and a pebble-bed design and (b) the AGR-1 irradiation in ATR. Similar information will be provided when available for subsequent AGR irradiations. In TEV-1022, burnup is given in GWD/MTU. For AGR-1, the conversion factor to % FIMA is 0.10542% FIMA/(GWD/MTU). For a representative prismatic design, the conversion factor is 0.10828% FIMA/(GWD/MTU), and for a pebble bed design the conversion factor is 0.10235% FIMA/(GWD/MTU). It is not germane to provide item (i) for (c) past irradiations in Europe as fuel behavior data from those irradiations will not be utilized in the qualification of NGNP fuel. The qualification of Next Generation Nuclear Plant (NGNP) fuel will be based on irradiations carried out in the AGR program.

Requested item (ii.) has been provided for Pd and Ag in TEV-1022. The inventories for several rare-earth elements (Ce, La, Nd, and Eu) in AGR-1 and a representative prismatic design are provided in Table 1 below, along with a discussion of the effects of rare earths on tristructural–isotropic (TRISO)-coated fuel particle performance. The effects of Pd and Ag on TRISO-coated fuel particle performance are discussed in response to RAI FQ-4/MST-5.

The inventory for many of the rare earth elements is quite large, as shown by the data for Ce, La, and Nd in Table 1, in which inventories of rare earths calculated for the AGR-1 experiment irradiation are compared with a representative prismatic design, the General Atomics 350 MW_{th} Modular Helium Reactor (MHR).

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[6] *Draft NRC Advanced Reactor Research Plan*, June 2002 (ML021760135, publicly available).

Table 1. Total number of fission product atoms per initial mass of uranium (atoms/g U initial) produced over the course of burnup (approximately 190 GWD/MTU) (approximately 20% FIMA).

Fuel	La	Ce	Nd	Eu
MHR	6.31E+19	1.14E+20	1.11E+20	2.44E+18
AGR-1	6.22E+19	1.13E+20	1.10E+20	1.95E+18
Ratio*	1.01	1.01	1.00	1.25

* Ratio=MHR/AGR-1

Note in Table 1 that the inventories of the rare earth elements with large inventories (La, Ce, and Nd) are basically the same for the AGR-1 irradiation and the MHR. For Eu, which has a much lower inventory, the inventory in the MHR is 25% larger than that from the AGR-1 irradiation, but is of no significance, as discussed below. The results in Table 1 bound the inventories in the pebble bed design because of the roughly factor of two lower burnup in the pebble bed.

For TRISO-coated fuel particles with uranium carbide kernels (for conceptual ease UC_2 is used here although some UC is also usually present) it is well known that rare earth elements such as Ce, Nd, La, Y, and Pr attack the SiC layer [2, 3]. The SiC layer is not attacked by these rare earths in fuel with UO_2 kernels because these elements are in the form of oxides, which are stable within the kernel. The chemistry of the rare earths in UCO fuels depends on the initial UC_2 content of the kernels and the burnup. As long as the oxygen potential within the kernel is controlled by the UC_2-UO_2 equilibrium, the high-yield rare earths listed above are in oxide form and do not attack the SiC layer. The UCO TRISO-coated particle fuel selected for qualification in the AGR program has an initial composition in the range of 15% to 35% UC_2. For the target 17% FIMA maximum burnup for a prismatic NGNP with a uranium enrichment of 14%, the oxygen potential in the kernel will be governed by the UC_2-UO_2 equilibrium during the entire fuel irradiation, and no rare earth attack of the SiC layer should occur.

Europium, which has a relatively small yield compared with Ce, La, and Nd, as shown in Table 1, is expected to be in the form of EuC_2 during irradiation of UCO fuel [2] but no evidence of attack on SiC has been recorded. Its mobility through intact SiC under long-term post irradiation annealing (10,000 hours = 417 days) of UCO fuel particles at temperatures in the range 1200-1500°C has been demonstrated [4]. Evidence of the release of Eu during AGR Program irradiations will be sought in post irradiation examination (PIE) by examining the fuel matrix and components of irradiation capsules for deposited Eu. The release of Eu will be measured in safety tests and holdup in graphite will be measured in the PIE of the AGR-3/4 test, which will contain designed-to-fail fuel particles and rings of fuel matrix and fuel element graphite at a variety of temperatures. Some evidence of the mobility of europium through intact SiC under irradiation has been measured in the on-going PIE of the AGR-1 irradiation.

References:

1. J. T. Maki and J. W. Sterbentz, "Response to Questions about the Applicability of the AGR Test Results to NGNP Fuel," TEV-1022, Idaho National Laboratory, September 2010.

2. F. J. Homan, T. B. Lindemer, E. L. Long, Jr., T. N. Tiegs, and R. L. Beatty, "Stoichiometric Effects on Performance of High-Temperature Gas-Cooled Reactor Fuels from the U-C-O System," Nucl. Technol., 35, 428 (1977).

3. T. N. Tiegs, T. B. Lindemer and T. J. Henson, "Fission Product Behavior in UC_xO_y Fissile Particles Made From Weak-Acid Resins," J. Nucl. Mater., 99, 222 (1981).

4. R. E. Bullock, "Fission-Product Release During Postirradiation Annealing of Several Types of Coated Fuel Particles," J. Nucl. Mater., 125, 304 (1984).

RAI FQ-2/MST-2: Describe in detail how the temporary increase in Cs diffusivity through SiC that is expected to be present during irradiation and absent afterward will be evaluated from the results of past and planned tests and experiments.

Comments: This RAI incorporates the second of two earlier information requests transmitted to INL in April 2010 and partially addressed by INL in TEV-1022.

Metallic fission product (e.g., Cs) release data obtained from accident heat-up simulation tests are used for predicting metallic fission product diffusion coefficients for the fuel temperatures associated with a core heat-up accident. Because these heat-up tests are conducted as part of post-irradiation testing, they do not address any diffusion-related phenomena that are present during irradiation and absent afterward. The additional use of such post-irradiation heat-up data as "margin data" for predicting fission product diffusion during irradiation at operating temperatures above those addressed by the fuel qualification irradiations could therefore be non-conservative.

For example, recent experiments and atomistic simulations have suggested that lattice vacancies play an important role in both the solubility and the diffusion of Cs in SiC.[7] It is well known that neutron irradiation produces not only extended defects such as dislocation loops and voids but also temporary lattice vacancies and interstitials that disappear soon after irradiation stops.[8] These non-equilibrium vacancies and interstitials would likely increase solubility of Cs in SiC and accelerate Cs diffusion during irradiation.

Post-irradiation measurements would miss this effect and thus potentially under-predict Cs diffusion during irradiation. In general, the evaluation of diffusion effects during irradiation should consider how the concentration of lattice vacancies increases with both irradiation intensity and temperature.

Response FQ-2/MST-2:

As a point of clarification, the historical database for fission product transport data draws on both irradiation and post-irradiation heating tests to establish a best estimate of diffusion through the SiC layer.

An evaluation of experimental data resulted in a model that uses fast-neutron fluence as a parameter in the Arrhenius temperature dependence of cesium diffusion through SiC under irradiation (See Figure A-4 in Reference [1]). The influence of fast fluence in the model becomes insignificant at 1600°C.

In the AGR Program, the diffusion of cesium through SiC under irradiation will be evaluated by measuring Cs-134 and Cs-137 released from fuel particles and retained in the matrix of compacts and

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[7] T. Allen, I. Szlufarska, D. Morgan, K. Sridharan, M. Anderson, L. Tan, *Semi-annual report to the Nuclear Regulatory Commission on the Cooperative Agreement for Research on Advanced VHTRs*, University of Wisconsin, January 2010.

[8] W. Schilling, *Properties of Frenkel Defects*; and L. Hobbs, F. Clinard Jr., S. Zinkle, R. Ewing, *Radiation Effects in Ceramics*, Journal of Nuclear Materials, Volume 216 (1994), Pages 45-48 and 291-321.

other locations within the capsules during post-irradiation examinations. Capsules with no exposed kernels, as identified by fission gas release-to-birth ratios (R/Bs), will be used for these measurements. The PIE (e.g., leach-burn-leach) will determine if there are any fuel particles with defective or failed SiC. Data from the hot capsule (irradiated at 1400°C) in AGR-2 and hot capsules in the AGR-7 irradiation margin test will be of special interest. If cesium outside fuel particles is measured (in excess of that contributed by heavy metal contamination and defective/failed SiC), an analytic method based on a transient diffusion solution will be used to establish the diffusion coefficient of cesium in SiC under irradiation.

Based on the current PIE of AGR-1 fuel, no cesium above that expected because of contamination has been found, suggesting that enhanced diffusion via vacancies under irradiation is likely a very small effect.

Reference:

1. IAEA-TECDOC-978, Fuel Performance and Fission Product Behaviour in Gas Cooled Reactors, International Atomic Energy Agency, November 1997.

RAI FQ-3/MST-3: Describe how the NGNP prototype plant will use startup and operational tests, operational monitoring, and periodic confirmatory measurements and inspections, in conjunction with license conditions and other regulatory controls, to confirm and monitor that initial and evolving NGNP operating conditions and performance are consistent with those predicted and considered for licensing.

Comments: This topic has particular ramifications for NGNP in view of the inherent technical challenges that make HTGR core operating conditions both difficult to measure and difficult to reliably predict. For both pebble-bed and prismatic-block HTGRs, the ability to perform on-line measurements is inherently limited by the high and highly variable temperatures themselves and associated challenges to sensor performance and the placement of sensor leads and structures in an otherwise all-ceramic refractory core. Given these challenges and the overarching importance of prototype confirmatory testing, monitoring, and inspections to the NGNP licensing approach, it is suggested that a white paper on this topic be developed to enable more productive discussions and feedback on NGNP licensing and technical issues.

The NGNP white papers on FQ, MST, and HTM seek NRC agreement that the specified technical approaches to qualification, analysis, and validation are acceptable. In particular, qualification and validation adequacy cannot be judged without considering what tests and periodic confirmatory measurements and inspections will be performed on the NGNP prototype plant. As a basic principle of performance-based regulation, it is generally true that less extensive operational confirmation calls for more extensive prior validation and qualification of the predicted operating conditions and performance elements that affect safety.

The acceptability of the licensing approach for the NGNP prototype plant is keyed to understanding how startup and operational tests, monitoring, and periodic measurements and inspections will be used to confirm and ensure that normal operating conditions and performance in the prototype pant are consistent with those predicted for licensing. Such tests and measurements are also needed for detecting and monitoring the advent of credible anomalous or off-normal operating conditions (e.g., blocked coolant holes, local pebble flow anomalies). Of particular concern is the potential for either inaccurately predicted normal conditions or undetected off-normal operating conditions to exceed those addressed in the licensing safety evaluation and the qualification, analysis, and validation that support it. Depending on their likelihood and difficulty of detection, the potentially undetected presence of certain anomalous or off-normal operating conditions may have to be considered in establishing operating limits and factored

into both the long-term and immediate pre-accident NGNP operating histories assumed in licensing safety analysis.

It is thus clear that information on startup tests, operational monitoring, and periodic confirmatory measurements and inspections must be considered in determining the ranges of operating service conditions (e.g., temperature histories, fuel burnups, neutron fluences, chemical environments, temperature gradients, cyclic loads, etc.) to be addressed by qualification testing, analysis, and validation. Briefly discussed below are a few of the areas where potential needs and opportunities are seen for performance-based monitoring and periodic confirmatory measurements in the NGNP prototype plant:

(a) Specific nuclear measurements will likely be needed to confirm predicted core power shapes, including the engineered power shapes in a prismatic-block NGNP, and more generally to detect plausible core irregularities such as fuel misloadings, pebble flow anomalies, block-stack motions, etc. Given the technical obstacles to deploying conventional in-core detectors in both pebble-bed and prismatic-block HTGRs, core monitoring and confirmation will likely have to place significant reliance on ex-core and ex-vessel detectors as well as post-irradiation examination (PIE) of discharged fuel compacts or pebbles and near-core or in-core activation probes.

(b) Periodic confirmatory PIE and accident heatup testing on fuel compacts or pebbles sampled from the NGNP discharge or circulation streams may be needed to confirm CFP integrity and retentiveness. Such tests would help address any outstanding fuel performance uncertainties such as those potentially associated with (i) the adequacy and reliability of fuel quality controls, (ii) the potential for fuel operating conditions (e.g., temperatures in undetected core hot spots) to exceed those addressed by qualification testing and analysis, and (iii) particular fuel-weakening phenomena in the NGNP core exceeding those in the MTR-based test irradiations used for fuel qualification (e.g., palladium time-at-temperature, as shown in TEV-1022).

(c) Monitoring or periodic sampling of dust and plate-out activity at selected locations may be needed, in addition to routine activity monitoring of circulating helium and the helium purification system, to resolve outstanding uncertainties related to fuel qualification and mechanistic source terms.

(d) Temperature distributions in the core and/or at the core outlet will likely have to be periodically measured to confirm and monitor that operating temperature profiles and histories remain consistent those for which the fuel and other materials and components have been designed, qualified, and analyzed. Such temperature measurements in past HTGRs have employed a variety of sensor technologies including "NATE" sensors,[9] thermocouples, noise thermometers, and melt-wires. The following observations bear noting in this context:

    (i) It has been reported that indirect measurements and analyses showed bypass helium flows, and the resulting core operating temperatures, to have been significantly higher than predicted in the Arbeitsgemeinschaft Versuchsreaktor (AVR, Consortium Test Reactor, Ltd.), Thorium High Temperature Reactor (THTR), and Fort Saint Vrain (FSV) HTGRs. In particular:

        - Recent analyses of the AVR's melt-wire experiment have shown that the neglect of core bypass flows was likely a major contributor to the >250 °C under-prediction of peak core

[9] Note: Please provide a reference that describes the THTR's "NATE" sensor system and its use to monitor temperature profiles in the bottom reflector, as mentioned in [H. Kalinowski, July 2001].

operating temperatures by the computational models used for AVR design and safety analysis.[10, 11]

- In the THTR, the predicted and actual core bypass flows were reported as 7% and 18%, respectively.[12] However, because the THTR ran for only 1.2 full-power years, no data were obtained on the increases in bypass flows that generally occur with the irradiation-induced shrinkage of HTGR reflector blocks.
- At FSV, measurements in the core outlet plenum showed radial temperature gradients significantly larger than expected. These were attributed to higher-than-predicted bypass flows caused in part by excessive pressure drops across the helium inlet orifices above each fuel block stack.[13]

(ii) The inherent long-term instability of the radial profiles of pebble-flow rate, power, and temperature (caused largely by high pebble friction in helium that decreases with increasing temperature – see related comments under RAI MST-7) gave rise in the THTR to a large inner-core hot spot[14] that tended to intensify with operating time. Despite effective monitoring of evolving outlet temperature radial profiles in the bottom reflector and the use of adaptive zonal loading strategies to counteract the pebble flow profile instability, the evolution of increasingly excessive helium outlet temperature gradients in the THTR appears to have contributed to the early failure of 35 of the ~2,600 hot-duct insulation attachment bolts in just 1.2 full-power years of operation.

(iii) Although perhaps less widely discussed, the potential for core hot spots during normal operation is of significant concern also in prismatic-block HTGRs. This is because prismatic-block HTGRs:

- have highly variable and uncertain bypass flows between blocks within the core in addition to those in the reflectors,
- may be susceptible to various local "closed-core" under-cooling effects including undetected local blockages of fuel coolant holes,
- may keep fuel in potential hot spots for many months at a time, and

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[10] C. F. Viljoen, R. S. Sen, F. Reitsma, U. Ubbink, P. Pohl, H. Barnert, *The Re-Evaluation of the AVR Melt-Wire Experiment Using Modern Methods with Specific Focus on Bounding the Bypass Flow Effects,* HTR-2008 Topical Meeting, Washington, DC.

[11] C. F. Viljoen, R. S. Sen, *The Re-Evaluation of the AVR Melt-Wire Experiment with Specific Focus on Different Modelling Strategies and Simplifications,* HTR-2010 Topical Meeting, Prague.

[12] R. Bäumer, I. Kalinowski, *THTR Commissioning and Operating Experience*, 11th International Conference on the HTGR, June 1989 (paper included in handouts and discussed by NRC staff in: *Safety Aspects of HTR-technology - NRC visit in Germany – 23-26 July 2001*, GRS, ML092250104).

[13] Note: Please provide a reference that describes such FSV observations (as recalled by the requester with perhaps questionable accuracy).

[14] Note: The term "hot spot" is defined here as a core region that runs significantly hotter than intended during ostensibly normal operation.

- rely on engineered power shaping achieved through fuel shuffling and complex zoning of fuel and burnable poison, yet, unlike light water reactors, do so with:
  - highly variable and uncertain local moderator temperatures,
  - incomplete bound thermal neutron scattering data (i.e., little or no fluence-damage dependent graphite S(alpha, beta) data),
  - little fully applicable validation benchmark data, and
  - little or no real-time confirmation or calibration from in-core measurements.

It bears noting here that local decay power shapes directly affect heatup accident temperatures, whereas effects of localized coolant flows before the accident are effectively smeared by conductive and radiative heat transfer as the accident evolves over many hours.

Response FQ-3/MST-3:

It is early, with design still underway and the license applicant having not yet been identified, to discuss specific commitments for the content of the various testing programs for the NGNP Plant.

This response provides brief conceptual summaries of the following programs currently envisioned for the single-module HTGR demonstration plant: (1) a startup-testing program, (2) a demonstration-testing program supporting Design Certification of a Standard Reactor Module, and (3) operational surveillance programs. As the design matures and the licensing interactions progress, it is anticipated that several Licensing Topical Reports will be prepared on these topics by the license applicant. However, specific commitments at this time would be premature.

Although the pre-commissioning effort is not formally considered to be part of the startup schedule, it is addressed here because it serves as a transition between construction and commissioning. A substantial portion of the pre-commissioning effort will be to fulfill the required portion of the Inspections, Tests, Analyses, and Acceptance Criteria (ITAACs) prior to the beginning of the Initial Test Program (ITP). Pre-commissioning activities must be completed prior to the commencing ITP so that the required systems can fulfill their functions. This typically includes initial instrument and equipment calibration, flushing, cleaning, wiring continuity checks, insulation resistance tests, cable separation checks, rotational direction, pressure testing, and functional testing of components and systems. It is during these individual equipment and system commissioning tests that the basic functional capabilities of instrumentation and systems will be confirmed. Many of these activities must be completed prior to initial fuel loading. Special instrumentation in addition to the normal plant instrumentation, to support validation of analytical methods and confirm performance of first-of-a-kind components, will likely be needed to support the subsequent commissioning effort.

Startup Testing Program

The startup testing (commissioning) program will be based on regulatory and customer requirements, as well as reactor supplier requirements. It will address customer requirements such as demonstrating overall output, efficiency claims, load demand response capability, operability and maintainability, initial reliability, and investment protection. It will also address the needs and requirements of the reactor supplier for further confirmation of the design and validation of analysis tools. The program will be performed in a manner that meets the safety and licensing requirements of the NRC. The specifics of the ITP will confirm elements of the safety analyses and will help to achieve regulatory acceptance of the

modular HTGR demonstration plant by satisfying many of the ITAACs. The insights from regulatory guidance for startup and testing of nuclear power plants (e.g., Regulatory Guide 1.68 for Light Water Reactors [LWRs]) will be reviewed and reflected in this plan. The plan will also be based on the commissioning and operating experience from earlier HTGRs, especially the two licensed U.S. HTGRs (Peach Bottom Unit 1 and FSV).

The commissioning tests will be divided into several major phases, and within each phase specific tests will be conducted. The tests chosen for each of the phases will be based on experience gained from past HTGR commissioning and will be modified as appropriate for the industrial applications. The program steps within each phase and the associated tests will be sequenced such that the test program can be carried out in a logical and efficient order. The following commissioning phases are currently and tentatively envisioned:

- Integrated Hot Flow Test
- Initial Fuel Loading and Zero Power Testing
- Low Power Testing
- Power Ascension Testing

Demonstration Testing

The purpose of the demonstration-testing program is to provide operational data to confirm the analytical bases for Design Certification of a standard reactor module, which will be obtained to facilitate licensing of additional modules for the demonstration plant and/or follow-on commercial modular HTGR plants. The tests to be performed as part of this program will ultimately be defined based on the need to fill any data gaps in the safety case but will also depend on the availability of the plant for such testing and the contractual relationships between the plant owner, the reactor supplier, and the Government. Those caveats aside, it is currently envisioned that the following tests may be considered by the reactor supplier, owner-operator, and other stakeholders as part of the demonstration-testing program. These tests would be primarily designed to confirm the safety margins and are fundamental to demonstrating the safety capabilities of the modular HTGR. This preliminary list is not intended to be all-inclusive, and tests may be added, modified, or deleted to accommodate plant design specifics.

- Trip of the Heat Transport System (HTS) and reactor trip with various delay times prior to resumption of forced cooling with the HTS or the Shutdown Cooling System.
- Simulation of a Pressurized Loss of Forced Circulation and reactor trip, with passive core residual heat removal using the Reactor Cavity Cooling System (RCCS).
- Demonstration of a Depressurized Loss of Forced Circulation and reactor trip, while simultaneously depressurizing the primary coolant to atmospheric pressure utilizing the RCCS as the passive means for the core residual heat removal.
- Demonstration of partial RCCS performance degradation during normal operation.
- Demonstration of time available during normal operation without helium purification.

- Simulation of a reactor shutdown with only the inherent negative temperature coefficient in the same manner as was demonstrated by Arbeitsgemeinschaft Versuchsreaktor (AVR) and HTR-10, i.e., tripping the HTS while not allowing the Reactor Control System to insert the control rods.

In some cases, these events may be initiated from less than full power and at several power levels in a graded manner to ensure there will be no damage to the plant equipment or a reduction of plant life. This demonstration-testing program is expected to last approximately one year.

Operational Surveillance Program

A comprehensive operational surveillance program will be formulated by the license applicant and approved by the NRC. This surveillance program will be initiated as applicable during the pre-commissioning phase and will continue throughout the operating lifetime of the plant (although the frequency of some activities may be reduced once the plant operating characteristics have been established). There will be a strong emphasis on safety-related structures, systems, and components (SSCs), and the surveillance program will confirm that these SSCs continue to meet design requirements through their design lifetimes.

Some of these surveillance activities will be required to demonstrate compliance with the plant Technical Specifications. The demonstration of fuel performance as a function of temperature, fluence, and burnup will be accomplished primarily by measuring the circulating activity in the primary coolant (primarily noble gases). See the response to RAI FQ-23/MST-28 for a discussion of possible NGNP Technical Specifications related to confirmation of fuel performance through measurements of primary and secondary radionuclide content. See the response to RAI FQ-26/MST-31 for a discussion of fuel surveillance.

Formulation of this surveillance program will be strongly influenced by the surveillance programs conducted for prior HTGRs, and the lessons learned from them will be incorporated in the program. The surveillance program will be approved early in the final design phase to ensure that all of the requisite instrumentation is included in the plant design and that adequate provisions are made in the plant design for In-Service Inspection (ISI) access, including compliance with as-low-as-reasonably-achievable (ALARA) requirements.

It is anticipated that the operational surveillance program could include, but would not be limited to, the following elements:

- Measurement of the circulating and plateout activities in the primary circuit
- Measurement of an overall plant tritium mass balance
- Sampling and characterization of the particulate matter ("dust") in the primary coolant circuit
- Inspection, including metrology, of components removed from the primary coolant circuit for repair or replacement
- Post irradiation examination of fuel elements and reflector elements discharged from the core
- Measurements and ISI to monitor the performance of the RCCS
- Provisions to monitor the performance of the primary pressure boundary (e.g., reactor vessel Charpy specimens, nondestructive ISI examination measurements, etc.)

- Measurements of the leak rates of primary and secondary coolants (inferred from makeup rates)
- Measurements and ISI to monitor the corrosion of structural materials in the primary circuit.

Again, it must be noted that it is early, with design still underway and the license applicant having not yet been identified, to discuss specific commitments for the content of the various testing programs for the NGNP Plant. This response only provides brief conceptual summaries of the testing programs currently envisioned for the single-module HTGR demonstration plant. Specific commitments will be made at the appropriate time by the facility license applicant.

THTR NATE Sensors

Among the footnotes to the comments that accompany this RAI was a request that the NGNP Project provide a reference that describes the Thorium High Temperature Reactor (THTR) "NATE" sensor system and its use to monitor temperature profiles in the bottom reflector, as mentioned in [H. Kalinowski, July 2001]. Prior to receipt of this RAI, the NGNP Project had no information on the NATE sensor system. Upon further investigation, it was learned that [H. Kalinowski, July 2001] is a handout attached to an NRC trip report that is filed under the NRC's Agency wide Documents Access and Management System (ADAMS) under accession number ML092250104. The report is for a trip to Germany taken by NRC staff in 2001 to review various topics related to pebble bed HTGR safety. The reference cited is a handout for a presentation by Dr. Helga Kalinowski on startup testing at THTR. The handout includes a handwritten drawing that shows a plan view of the locations of the NATE sensors at the bottom of the reactor.

To respond to this information request, the NGNP Project staff contacted a former GA and INL staff member who was assigned to THTR for over four years. The following information regarding the NATE sensors was obtained.

"NATE" is a German acronym for "Nicht-Autauschbare Thermo-Elemente", which in English translates approximately to "nonreplaceable temperature sensor element". These were hard-wired sensors that were located in THTR throughout the reflector (side and bottom) and at the inlet to the SG bundles. They were thermocouples that were designed as special commissioning instruments. Should they fail to function during the plant life, they would be left in place and would not be replaced since they were not needed for the operation of the plant. These sensors could be read into a computer at a press of a button to record the temperatures. This was done during the start up at 40, 60, 80 and 100% power.

Figure 1 is a page from the "THTR Shift Supervisor Training Manual – Metallic and Graphite Reactor Internals - Document Number CU0025". It includes a brief description of the NATE sensor system. The document is in German. Figure 2 is another drawing that shows the location of the THTR NATE sensors.

This is all of the information on these sensors that the NGNP Project has obtained.

Fort St. Vrain Core Outlet Temperatures

The comments that accompany this RAI include the following observation regarding FSV core outlet temperatures:

> *"At FSV, measurements in the core outlet plenum showed radial temperature gradients significantly larger than expected. These were attributed to higher-than-predicted bypass flows caused in part by excessive pressure drops across the helium inlet orifices above each fuel block stack."*

An accompanying footnote requests "a reference that describes such FSV observations (as recalled by the requester with perhaps questionable accuracy)."

The NGNP Project staff is not familiar with the conditions described in the footnote. To obtain additional information, discussions were held with a former member of the GA staff who had lead responsibility for thermal/hydraulic analysis of FSV operating data. That individual was also not familiar with the conditions described in the footnote. It was noted in these discussions that the thermocouple guide tubes, through which the region outlet thermocouples were wired from the bottom reflector to the side of the prestressed concrete reactor vessel, experienced some bypass helium flow leakage at their entrance. This resulted in depressed region outlet temperature measurements in regions at the southeast perimeter of the core. Methods were developed to correct these temperature readings to ensure proper balance of the helium inlet orifices. Modern modular HTGRs do not have helium inlet orifice valves above the core.

In-Core Instrumentation

The comments that accompany this RAI refer to historical thermal anomalies in certain HTGRs and "the technical obstacles to deploying conventional in-core detectors in both pebble-bed and prismatic-block HTGRs". RAI FQ-52/MST-56 also addresses this issue, and the response provides information regarding such anomalies and the status of current technology development efforts for in-core instrumentation for HTGRs.

1.6 Sicherheitstechnische Anforderungen

Die sicherheitstechnische Bedeutung des thermischen Schildes ist insofern gegeben als bei einem Erdbeben mit einer Horizontalbeschleunigung von 0,1 g die NWA gesichert sein muß, d. h., die Funktion der Gasführung im Boden und Decke muß gewährleistet sein.

1.7 Instrumentierung

1.7.1 Aufgabenstellung

Überprüfung der rechnerisch ermittelten Auslegungswerte während des bestimmungsgemäßen Betriebes für die ersten Betriebsjahre.

1.7.2 Meßwertaufnehmer

Für die Überprüfung der Auslegungswerte sind insgesamt 86 Temperatur-Meßwertgeber installiert. Ihre Aufteilung zeigt Bild-05. Sämtliche Temperatur-Meßwertgeber sind nicht auswechselbar, abgekürzt "NATE" geschrieben.

1.7.3 Meßwertverarbeitung

Entsprechend der Aufgabenstellung für die Instrumentierung wird der zeitliche Verlauf der Temperaturmeßwerte zuerst in einem Kernsicher (Platte) erfaßt, der bei normalem Meßwertanfall eine Speicherkapazität für 24 Betriebsstunden besitzt.

Figure 1. Description of the THTR NATE Sensor System

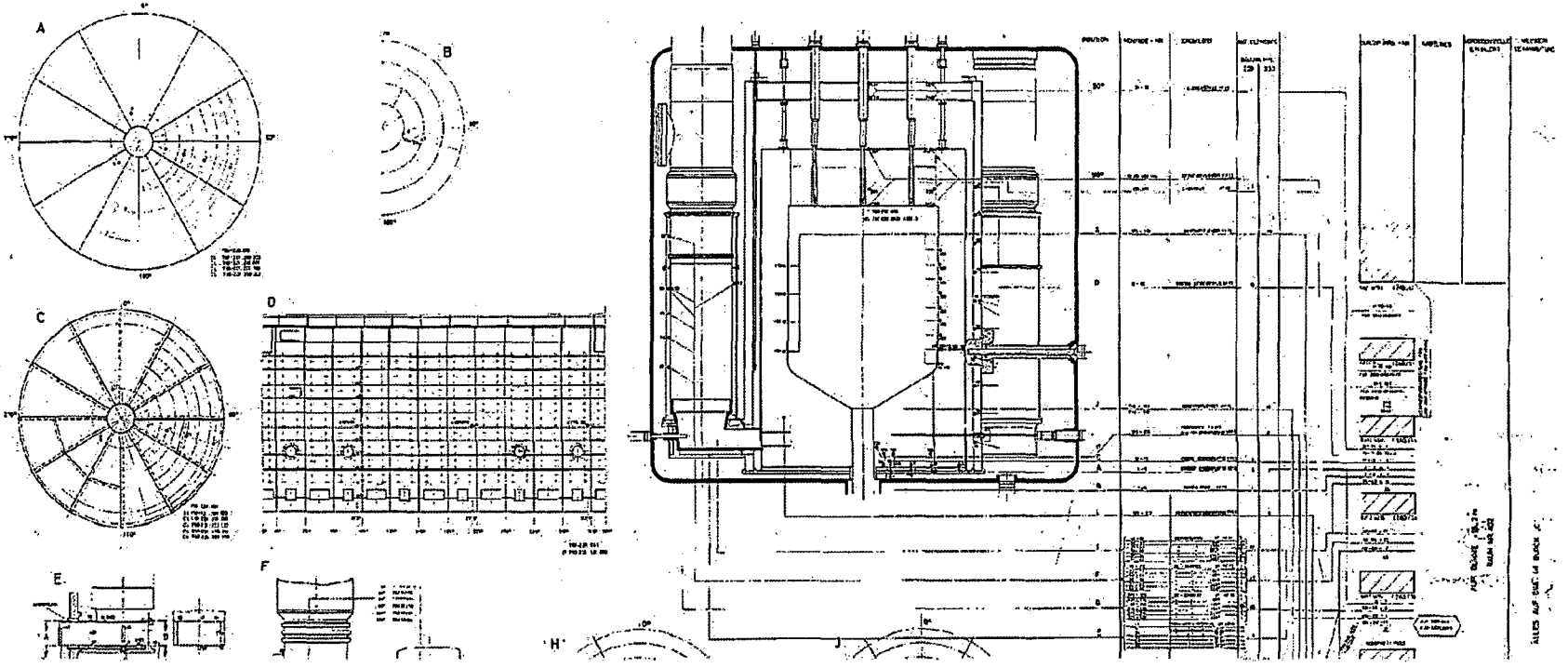

Figure 2. THTR NATE Sensor Locations

RAI MST-4: Provide a more definitive discussion of "event-specific mechanistic source terms" that considers the comments provided below.

Comments: The white paper solicits NRC agreement that the definition of event specific mechanistic source terms for the HTGR is acceptable. To that end, the definition and supporting discussions would benefit from clarification with regard to the following:

a) The white paper uses the word "deterministic" in unusual ways that tend to obscure or confuse the intended meaning of event-specific mechanistic source terms. "Deterministic" is conventionally used elsewhere to mean non-probabilistic or nonstochastic (e.g., deterministic versus probabilistic risk assessment (PRA)-based approaches to LBE selection, deterministic versus Monte Carlo computational methods). The white paper, on the other hand, repeatedly uses the word "deterministic" to describe traditional light water reactor (LWR) source terms where other characterizations, such as "prescriptive," "non-event-specific," "nonmechanistic," or "bounding conservative," would seem more appropriate. Different yet likewise unclear meanings of "deterministic" seem to be intended where the term "deterministic analysis" appears in Table 3-1 and Section 4.3.

b) The white paper's discussion of event-specific mechanistic source terms could be made more definitive by expanding and improving the description of how the proposed source term approaches align or contrast with the variety of approaches used to determine LWR radiologic source terms in such contexts as level-3 PRA, siting analysis, emergency planning and response, security, the NRC's State-of-the-Art Reactor Consequence Analyses (SOARCA) program, etc. It would also be helpful to describe the respective source term approaches in terms of their uses of best estimate versus bounding or conservatively biased methods and assumptions and the evaluation and treatment of uncertainties.

c) The white paper does not clearly establish the scope of application of the proposed event-specific mechanistic source terms. Clarification is needed on whether the proposed source term approach is to be applied in such contexts as normal operations and maintenance, defined licensing basis events, siting analysis, level-3 PRA, emergency planning and response, security, etc. Where not applied, the use of alternate source term approaches should be identified.

d) The white paper acknowledges that the term "source term" can have different meanings in different contexts. Largely for that reason, it may be easier to agree on an acceptable definition of "event-specific mechanistic consequence analyses" rather than "event-specific mechanistic source terms." The revised terminology would accurately describe the proposed approach while avoiding potential confusion over the various meanings of "source term." Because the approach described in the white paper does not alter the traditionally accepted mechanistic methods for calculating dose consequences from a given release, switching to this alternate terminology would have little impact on the contents of the white paper yet would clarify the intent and its conceptual nexus with the LWR SOARCA program.

Response MST-4:

The response to RAI MST-84 provides further discussion of the definition of the mechanistic source term that responds to some of the issues raised in the comments that accompany RAI MST-4.

a) As used in the white paper, "deterministic" was intended to refer to analyses and analysis results containing prescriptive elements (models, assumptions, data, etc.) that may not be based on

detailed, mechanistic phenomena. Hence, it was implied, as noted in the NRC staff comments that accompany RAI MST-84 that deterministic analyses are, at least to some degree, not mechanistic to the extent intended for the calculation of mechanistic HTGR source terms. The use of the word "mechanistic" is intended to denote use of models that use first principle methods supported as needed by empirical confirmation to represent mechanisms (phenomena) that affect the generation and transport of radionuclides in the HTGR facility.

b) Questions regarding use of best estimate versus conservative analyses in source term calculations are addressed in the responses to RAIs MST-17, FQ-29/MST-34, FQ-30/MST-35, MST-67, MST-87, MST-88, and MST-101. It was not the purpose of the Mechanistic Source Terms White Paper to delineate approaches to calculating source terms that are used under a variety of situations by other reactor technologies. The white paper seeks NRC concurrence with the approach to be taken for the modular HTGR.

c) The proposed approach to calculating mechanistic source terms is intended to be used in all of the applications listed in part c) of the comments that accompany this RAI. No alternate source term approaches are being proposed.

d) The NGNP Project would like to receive a determination from NRC regarding the acceptability of the approach summarized in Section 1.3 and Section 6 of the Mechanistic Source Terms White Paper, including associated terminology.

RAI FQ-4/MST-5: In reference to RAI FQ-1 and the calculated palladium and silver inventories in MTR test irradiations and representative NGNP designs (including those presented in TEV-1022), please provide the following information:

a) Provide a summary of the current state of knowledge about the behavior of silver and palladium in TRISO fuel and how silver and palladium time-at-temperature can affect TRISO coating integrity and retentiveness under (i) operating conditions, including in postulated operations with unintended core hot spots at temperatures up to 1300 and 1400°C, and (ii) heatup accident conditions at temperatures up to 1600 and 1700°C.

b) Building on the summarized knowledge, provide discussions on (i) the importance of including plutonium burnup, or silver or palladium time-at-temperature, as one of the parameters to be enveloped by TRISO irradiation testing and accident heatup testing, and (ii) proposed approaches for assessing and mitigating fuel qualification and MST uncertainties associated with the potential for silver and palladium time-at-temperature being significantly higher in prismatic-block or pebble-bed NGNP designs than in the TRISO fuel irradiation tests performed in MTRs.

c) Identify potential test design approaches for increasing plutonium fission and palladium time-at-temperature in MTR-based fuel irradiation tests and assess their feasibility and effectiveness both individually and in combination. Such approaches should include but not be limited to: (i) reducing the TRISO fuel's initial enrichment, (ii) hardening the MTR's local thermal neutron spectrum, (iii) increasing the MTR's local epithermal neutron spectrum, and (iv) replacing a small fraction of UO_2/UCO in the fuel kernels with PuO_2/PuCO.

Comments: The requested knowledge summary should draw on the current body of relevant technical literature, including the TRISO fuel phenomena identification and ranking table (PIRT) results

documented in NUREG/CR-6844 (2004) and references cited therein and related information such as[15, 16, 17] as appropriate. The discussion should address related aspects of RAIs FQ-1 and FQ-2 and specifically describe the technical basis for INL's following statement in TEV-1022:

"TRISO fuel performance is also strongly affected by the concentrations of fission products such as silver and palladium. These concentrations are dependent upon the fuel isotopes, thus the rate of formation changes with burnup as plutonium builds in and uranium burns out. The rate of plutonium build-up in the fuel is a strong function of the spectrum. As the AGR-1 test spectrum may differ from the NGNP spectrum, it is important that the final plutonium and fission concentrations in the AGR-1 test are comparable to those anticipated in NGNP fuel."

Response FQ-4/MST-5:

It is well known that silver is released from intact TRISO-coated particle fuel during irradiation at elevated temperatures [1-5], and evaluations of these data suggest that SiC may not be an effective barrier for silver during irradiation above about 1100°C [3] or 1200°C [1, 2, 4, 5]. Under accident conditions, silver is released as a function of time at temperature over the range 1500-1800°C [2, 6, 7, 8], and heating of individual fuel particles has documented a wide variation in silver release from particle to particle [7, 8].

The mechanism of silver transport through intact SiC is an area of active research, with recent results identifying grain boundaries as a likely pathway for the silver diffusion [9-12] in contrast to other work invoking nanocracks as a likely pathway [4]. Experimental and theoretical evidence has been presented that grain boundary morphology (equiaxed vs. columnar vs. laminar) and type (high angle vs. low angle) can affect silver mobility through SiC [9, 12, 13]. In one out-of-pile experiment, silver transport via grain boundaries in SiC was found to take place only in the presence of palladium [11].

While there has been some speculation that the diffusion of silver through SiC may be a precursor of SiC damage [14], that hypothesis has not been confirmed. Silver in very high concentrations (100 times greater than those expected in high burnup low-enriched uranium (LEU) fuel in a modular HTGR) will, at temperatures in the range 1200 to 1500°C attack SiC. However, at concentrations more typical of modular HTGRs, silver passes through intact SiC under normal irradiation conditions without evidence of attack [15]. No influence of burnup or fast fluence was found on the diffusion of silver through silicon carbide over the range 2.5 to 12.1% FIMA and 0.5 to 5.9 $\times 10^{25}$ n/m^2, respectively [16]. As shown in TEV-1022 [17], the inventory of silver in a modular HTGR is calculated to be up to 1.66 times the inventory of silver in the AGR-1 experiment at the same burnup.

Palladium is known to interact with SiC under irradiation forming metallic nodules at sites of attack [15, 18]. The attack is localized and highly variable and does not cause thinning of the SiC. The rate of attack was faster in out-of-pile tests in which a temperature gradient of 278°C/cm was applied than in irradiated coated fuel particles and faster in accelerated irradiations than in real-time irradiation tests [18]. While

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[15] P. E. Brown, A. J. Inns, R. J. Pateman, B. A. Phillips, B. M. Sharpe, *Post-Irradiation Examination of HTR Fuel Elements R2-K13/1, HFR-K3/1&3, FRJ2-K13/2&4, AVR 71/22, 74/11, 76/18, and 76/20*, Harwell Laboratory, Didcot, Oxon, England, AERE-G4740, May 1988.

[16] K. Minato, T. Ogawa, S. Kashimura, K. Fukuda, M. Shimizu, Y. Tayama, I. Takahashi, *Fission Product Palladium-Silicon Carbide Interaction in HTGR Fuel Particles*, JNM 172 (1990) 184-196.

[17] J. H. Neethling, J. H. O'Connell, E. J. Olivier, *Palladium Assisted Silver Transport in Polycrystalline SiC*, HTR-2010 Topical Meeting, Prague.

there are reports of the buildup of palladium at the IPyC-SiC interface in German high temperature safety tests [2] there are no reports of SiC corrosion by palladium in German irradiation tests which were generally conducted at an acceleration factor relative to real-time HTGR irradiation of 3 or less.

An analysis of the depth of SiC corrosion by palladium in irradiated fuel particles under the assumption that the reaction was controlled by the release of palladium from the fuel kernel concluded that the maximum reaction depth depends on the amount of palladium released from the kernel to the one-third power [18]. In these irradiations burnup varied from 1.4 to 9% FIMA, and average temperatures ranged from 1065 to 1510°C.

In another study, evidence indicated no threshold concentration necessary for palladium penetration of SiC, and increasing amounts of palladium increased only the number of nodules in the SiC [19]. In these irradiations the amount of palladium per fuel particle varied by an order of magnitude and the average temperature ranged from 1000 to 1425°C. Temperature was the major factor affecting penetration of the SiC and was found to fit an Arrhenius relationship. Penetration rate data plotted in Reference 19 include results from the FTE-13 experiment [20] in which TRISO-coated PuO_2 kernels were irradiated to 70% FIMA. Included in this experiment are fuel particles with palladium concentrations per unit geometric surface area of the kernel (a metric for mass flux across a surface) about four times, or per unit geometric surface area of the inside of the SiC layer (a metric for reaction with the SiC) about three times, compared with values typical of LEU TRISO-coated fuel particles considered for use in an HTGR.

New measurements of the distribution of palladium in fuel particles irradiated over a wide range of burnup and temperature will be made in the post-irradiation examinations of the AGR Program to provide insight on the processes controlling Pd – SiC interaction. In TEV-1022 [17], the inventory of palladium in an HTGR is calculated to be 1.49 times the inventory of palladium in the AGR-1 experiment. The cube root dependence of palladium penetration of the SiC from [18] indicates an effect of $(1.49)^{1/3} = 1.14$ on penetration because of the greater inventory of palladium in the NGNP relative to that in the AGR-1 experiment, whereas no effect of palladium inventory would be expected based on Reference [19].

In conclusion, in contrast to the NRC statement cited within TEV-1022 (Reference 17) that "TRISO fuel performance is also strongly affected by concentrations of fission products such as silver and palladium", this review of the literature on the behavior of silver and palladium suggests no strong influence of the inventory of these materials on the behavior of TRISO-coated particle fuel in the NGNP (inventories roughly 50% higher than in AGR irradiations) at either normal operation or accident conditions. In the AGR Program, the release behavior (fraction released and locations within the irradiation capsule) and influence of silver and palladium on SiC microstructure after normal operation over a wide range of burnup, fast fluence and operating temperature will be measured during PIE. The release behavior and influence of these metals will be measured during heatup tests at 1600 and 1700°C, and their influence on SiC microstructure under accident conditions will be measured by examination (such as metallography and SEM) after the heating tests. The conduct of a fuel surveillance program in the NGNP demonstration plant, such as discussed in the response to RAI FQ-3/MST-3, will confirm whether fuel performance is consistent with the expected performance based on the results of the NGNP/AGR fuel qualification program.

a. Given the summary above, it is (i) not particularly important to include plutonium burnup, or silver and palladium time-at-temperature as one of the parameters to be enveloped by TRISO irradiation testing and accident testing. However, in each AGR Program irradiation there is a considerable range of TRISO fuel burnup and silver and palladium inventories. The inventories of these fission products will be documented for each irradiation, any influence of them on fuel behavior observed in the PIE, and in the safety testing will be noted. Because the effects of silver

and palladium time-at-temperature in NGNP designs are not expected to be significantly greater than those in the AGR irradiations, no special approaches (ii) are required to assess or mitigate uncertainties in these effects for fuel qualification or mechanical source terms (MSTs).

b. Given the result of the summary in (a) above that no great effect of silver or palladium inventory is expected on TRISO coating integrity or retentiveness, there is no need to design approaches to increase the inventories of silver and palladium in the AGR irradiation tests. However, for completeness, all four of the approaches suggested in the RAI have been evaluated. Hardening of the thermal neutron spectrum for the AGR-1 and AGR-2 tests has been done through the combination of a hafnium shield and borated graphite, but the other three approaches suggested are not feasible or would preclude meeting other important test objectives.

References:

1. H. Nabielek, P. E. Brown, and P. Offermann, "Silver Release from Coated Particle Fuel," Nucl. Technol. 35, 483 (1977).

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3. U. S. NRC, "TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables (PIRTs) for Fission Product Transport Due to Manufacturing, Operations, and Accidents," NUREG/CR-6844, Vol. 1, 2004.

4. H. J. Maclean, "Silver Transport in CVD Silicon Carbide, Ph.D. Thesis, MIT, 2004.

5. J. J. van der Merwe, "Evaluation of Silver Transport Through SiC During the German HTR Fuel Program," J. Nucl. Mater., 395, 99 (2009).

6. W. Schenk and H. Nabielek, "High-Temperature Reactor Fuel Fission Product Release and Distribution at 1600 to 1800°C," Nucl. Technol. 96, 323 (1991).

7. K. Minato, K, Sawa, T. Koya, T. Tomita, A. Ishikawa, C. A. Baldwin, W. A. Gabbard, and C. M. Malone, "Fission Product Release Behavior of Individual Coated Fuel Particles for High-Temperature Gas-Cooled Reactors," Nucl. Technol. 131, 36 (2000).

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11. J. H. Neethling, J. H. O'Connell, and E. J. Olivier, "Palladium Assisted Silver Transport in Polycrystalline SiC," Paper 196, Proceedings of HTR 2010, Prague, Czech Republic, October 18-20, 2010.

12. I. Szlufarska, "Ag Transport through SiC: Modeling and Experiment," VHTR R&D FY10 Technical Review Meeting, Denver Colorado, April 27-29, 2010.

13. D. A. Petti, J. Buongiorno, J. T. Maki, R. R. Hobbins, and G. K. Miller, "Key Differences in the Fabrication, Irradiation, and High Temperature Accident Testing of US and German TRISO-Coated Particle Fuel, and Their Implications on Fuel Performance," Nucl. Eng. Design, 222, 281 (2003).

14. W. Schenk, D. Pitzer, and H. Nabielek, "Fission Product Release Profiles from Spherical HTR Fuel Elements at Accident Temperatures," Kernforschungsanlage Jülich, GmbH, Jül-2234, September 1988.

15. R. J. Lauf, T. B. Lindemer, and R. L. Pearson, "Out-of-Reactor Studies of Fission Product-Silicon Carbide Interactions in HTGR Fuel Particles," J. Nucl. Mater. 120, 6 (1984).

16. W. Amian and D. Stover, "Diffusion of Silver and Cesium in Silicon-Carbide Coatings of Fuel Particles for High-Temperature Gas-Cooled Reactors," Nucl. Technol. 61, 475 (1983).

17. J. T. Maki and J. W. Sterbentz, "Response to Questions about the Applicability of the AGR Test Results to NGNP Fuel," TEV-1022, Idaho National Laboratory, September 2010.

18. K. Minato, T. Ogawa, S. Kashimura, K. Fukuda, M. Shimizu, Y. Tayama, and I. Takahashi, "Fission Product Palladium-Silicon Carbide Interaction in HTGR Fuel Particles," J. Nucl. Mater. 172, 184 (1990).

19. T. N. Tiegs, "Fission Product Pd-SiC Interaction in Irradiated Coated-Particle Fuels," Nucl. Technol. 57, 389 (1982).

20. C. M. Miller and W. J. Scheffel, "Postirradiation Examination and Evaluation of Peach Bottom FTE-13," GA Technologies, Document No. 906939, November 1985.

RAI FQ-5/MST-6: In reference to RAI FQ-1 and TEV-1022, provide the detailed MTR design and test information needed to perform independent modeling of (a) the completed, ongoing, and planned AGR TRISO fuel and material irradiations in the ATR MTR and (b) the past TRISO fuel irradiations in the FRJ2 (DIDO), HFR, and Siloe MTRs.

Comments: The NRC staff will use the requested information to perform selected independent calculations of the spectral burnup isotopics and fission product inventories in TRISO fuel irradiated in the ATR and other MTRs. The independent calculations will help the staff further judge the applicability of MTR fuel irradiation tests and understand potential issues with the nuclear analysis tools and approximations used in simulating the tests as well as the NGNP core. Such analysis issues would involve, among others, (i) the uses of PIE isotopic data in the test analyses, (ii) the modeling of fuel double heterogeneity, (iii) the effects of differences between the historic nuclear yield, decay, and branching data used in the ORIGEN 2.2 computer code and the updated data used in the ORIGEN-S computer code, and (iv) the different treatments of the temperature- and neutron-spectrum-dependence of nuclide-specific nuclear reaction rates in the JMOCUP computer code and modeling methods used by INL and the SCALE computer code methods used by NRC (e.g., the specific nuclides and temperatures for which spectrum-specific cross sections are calculated and used by the respective ORIGEN codes).

Response FQ-5/MST-6:

It is not possible, because detailed data are no longer available, to provide detailed information on the neutronics of past TRISO fuel irradiations in European reactors; however, these data are not germane to the qualification of fuel for the NGNP, which will be based solely on the AGR Program irradiations. As-

run data on AGR Program irradiations will be provided at the conclusion of each irradiation test. However, because of classification restrictions associated with the Naval Reactors Program detailed neutronics from the ATR are not available to be distributed by the AGR Program.

RAI FQ-6/MST-7: For the pebble-bed NGNP design (or similar) described in the white papers, identify the basic design provisions and operating strategies that will be used to monitor and control the long-term inherent instability of pebble flow-rate, power, and temperature profiles. If the evolution and effects of such pebble flow profile instabilities are believed to be known, small, negligible, or somehow non-existent, describe the technical basis for that belief considering the comments that follow.

Comments: In view of related problems encountered in the THTR, it would appear that such provisions would be necessary in a pebble bed NGNP design in order to ensure that operating conditions (e.g., fuel temperatures, burnup levels, and fluence levels) are held within in the qualified and analyzed limits presented in the FQ, MST, and HTM white papers. As noted below, pebble bed designs without provisions for radially zoned fuel loading would have no means to counteract the inherent instability of pebble flow profiles. Such designs would thus have to qualify fuel and materials and compute source terms over an appropriately expanded envelop of operating conditions.

The so-called "slippery inner core" problems in THTR have been attributed to the destabilizing effects on pebble flow caused by the inverse temperature dependence of the high pebble-to-pebble friction coefficients in dry helium. Namely, the peaked local power densities and temperatures in the inner core region resulted in lower local pebble friction, which in turn led to higher local pebble flow rates, lower average local burnups, and increasing local power densities and temperatures. The result was a positive feedback loop on the coupled radial gradients of pebble flow rates, average fuel burnups, power densities, and temperatures, respectively.

The accelerating passage of pebbles through the inner core region thus gave rise to a large inner-core hot spot that tended to intensify over the THTR's short operating lifetime. This occurred despite (i) the ability in THTR to monitor this unanticipated effect using fuel residence-time data inferred from the fuel handling system as well outlet temperature profiles measured by the "NATE" sensors in the bottom reflector, and (ii) the ability in THTR to dampen and control the effect by using the central and peripheral loading tubes to implement adaptive inner- and outer-core loading strategies (i.e., loading absorber pebbles and more highly burned fuel pebbles into the inner core region and fresher fuel pebbles into the outer core region).[18, 19]

Given that thousands of pebbles were unexpectedly fractured by the repeated insertion of THTR's in-core control rods, the instability caused by temperature-dependent pebble friction was likely further boosted by the accumulation of pebble fragments on the core floor. In other words, pebble debris resting on the core's conical floor impeded the passage of pebbles along the floor towards the discharge chute, thereby causing local pebble traffic slowdowns that propagated up the core periphery.

One might expect such pebble flow profile instabilities to be less pronounced in modern cylindrical-core pebble bed designs (e.g., the depicted pebble-bed NGNP design and similar designs such as China's High Temperature Reactor Power Module (HTR-PM), with their relatively slender cores and lack of in-core

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[18] H. Kalinowski, *Core Physics and Pebble Flow - Examples from THTR Operation*, (presentation handout included and summarized by NRC staff in: *Safety Aspects of HTR-technology - NRC visit in Germany – 23-26 July 2001*, GRS, ML092250104).

[19] R. Bäumer, *Selected Subjects on the Operation of the THTR 300*, VGB Kraftwerkstechnik, Feb 1989.

rods, and to have considerably less bearing on the annular-core designs considered until recently. With regard to the potentially exacerbating effects of floor debris on pebble flow profile gradients, it is noted that without in-core rods the AVR reported known pebble fractures at rates ranging from 1 per ~3,000 pebble circulations in the first years (mainly with U.S.-made graphite-shell-type fuel pebbles) to 1 per ~10,000 to 40,000 circulations in the final years of its 21-year operation (almost entirely with German-made A3-pressed fuel pebbles). It would thus appear that pebble fractures cannot be precluded in the pebble-bed NGNP design and will have to be considered in safety analysis and licensing. (See related comments for RAIs DC-1 and DC-8.)

The NRC staff is not aware of any published attempts to analytically predict such pebble flow effects in past or current pebble bed designs. Factors to consider in attempting such analyses would include the following:

- It is not clear how much is known about the temperature dependence, and perhaps the helium-impurity, fluence-damage, and wear dependence, of pebble-to-pebble and pebble-to-wall friction in reactor helium environments.

- It is likewise not clear how much pebble breakage might occur in future designs or how pebble fragments might accumulate on the core floor. It nevertheless seems clear that designers and regulators should consider the possibility that pebble breakage rates might prove high enough to merit attention in terms of potential effects on core pebble flow, etc.

- From recent studies at INL,[20] one can conclude that the ability to simulate multiple cycles of friction-mapped pebble flow in HTR-PM-sized cores with pebble bed mechanics codes like INL's PEBBLES computer code remains elusive, especially with the repetitions needed for nuclear and thermal feedbacks as well as parametric and uncertainty studies. Years of supercomputing clock time would be required unless major breakthroughs are achieved in speeding up the codes.

- Analyzing a much smaller core like that in China's HTR-10 reactor, on the other hand, might prove feasible in the near term with existing pebble bed mechanics codes. It is noted, however, that HTR-10 has operated only occasionally and may not yet have cycled enough pebbles at power to show significant flow profile evolution. Moreover, it appears that HTR-10 would likely require added provisions for measuring radial core and/or core-outlet temperature profiles and tracking pebble residence-time spectra in order to provide data suitable for benchmarking such applications of pebble-bed mechanics codes with nuclear and thermal coupling.

- Until practical tools are developed to rigorously model pebble flow mechanics with locally variable friction and nuclear and thermal coupling, useful studies can and should be done using simplified approximations of pebble flow mechanics. Approximations based on the apparent physical similarities[21] (e.g., similar flow velocity profiles) between pebble flow and the laminar flow of viscous liquids would likely be worth pursuing, noting that both liquid viscosity and pebble friction decrease with increasing temperature. For example:

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[20] J. Cogliati, A. Ougouag, *PEBBLES Mechanics Simulation Speedup*, PHYSOR 2010 Topical Meeting, Pittsburgh, May 2010.

[21] G. Murphy4, *Similitude in Engineering*, Ronald Press Co., 1950.

- A suitable fluid mechanics code could be used to model the 2D or 3D laminar flow of a fictitious viscous liquid representing the pebble bed core. The fluid model could be fed with trial viscosity-versus-temperature functions and coupled with available core thermal-fluid and nuclear codes.
- To treat the non-zero flows along the wall and floor boundaries, one could introduce a thin artificial boundary layer of low-viscosity fluid and increase its viscosity or reduce its local thickness to approximate the effects of floor debris, etc.
- Such models could be initially tuned and tested against the predicted flow profiles (i.e., with low, uniform friction) and inferred actual flow profiles (i.e., with high, temperature-dependent local friction) published for THTR [H. Kalinowski 2001, R. Bäumer 1989].

THTR experience also seems to suggest that it should be possible to adequately monitor and control such pebble flow instability effects through appropriate design choices and operating strategies. Such design choices would obviously have to include those providing for continual or periodic monitoring of the effect by measurements of (a) radial temperature profiles in the core (e.g., melt-wire pebbles) or bottom reflector (e.g., "NATE" sensors) and/or (b) pebble residence-time and/or burnup spectra.

Design choices to provide necessary stabilization of pebble flow profiles would likely have to include basic provisions for adaptively loaded inner and outer core fueling zones, i.e., peripheral loading tubes. Alternatively, modifying the core exit geometry (e.g., multiple discharge chutes instead of one) could help minimize the intrinsic instability of pebble flow profiles, but - given the large unknowns and uncertainties – may prove inadequate to eliminate or control the effect without help from adaptive zonal loading or other active measures.

Only if analysis can show that, without core zoning or other active control measures, the instability effects will evolve so slowly that they can be tolerated over many years (e.g., 10 years) of operation, then periodically resetting to a flattened radial profile of average pebble burnup by completely defueling and refueling the core (i.e., as may be needed anyway for replacing reflector blocks) may prove adequate. In that case, however, the lack of fully stabilized pebble flow profiles would ultimately preclude the attainment of equilibrium core configurations.

Response FQ-6/MST-7:

The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977, "Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011.

RAI FQ-7/MST-8: Describe how the potentially extended core residence times of TRISO fuel in intact pebbles, broken pebbles, and pebble fragments are factored into the fuel qualification plans and treated in MSTs calculations for the pebble bed NGNP design.

Comments: Experience from the AVR as well as the THTR (e.g., see related comments under RAI FQ-6) suggests the likelihood that at least a small population of TRISO fuel in intact pebbles, broken pebbles, and pebble fragments will be subjected to highly extended or indefinite core residence times. For example, it is expected that some broken pebbles and pebble fragments will find stable resting positions on the core floor before reaching the core discharge chute. These stationary fragments could then also slow or permanently block the flow of neighboring intact pebbles to the discharge chute.

The fuel population thus subjected to greatly extended or indefinite core residence times could then experience burnup and fluence levels well beyond those for which the fuel is otherwise to be qualified. Fuel performance and release calculations should account for the resulting potential for elevated failure fractions and reduced retentiveness in the affected populations of CFPs, along with their relatively high irradiation temperatures at the bottom of the core (as well as the relatively limited accident heatup temperatures at that location) and their elevated fission product inventories corresponding to high or extreme burnup. The RAI response should also consider how the size of the affected fuel population will be estimated or assumed and what measurements or inspections will be performed to ensure that the actual affected populations are within expected or assumed limits. Related comments are also provided under RAIs FQ-3, FQ-6, and FQ-11.

Response FQ-7/MST-8:

As stated in INL letter dated May 3, 2011, (CCN 223977 "Contract No. DE-AC07-051 0 14517 – Next Generation Nuclear Plant Project Submittal - NRC Project number 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers"), the material in Section 5.2 of the FQ White Paper should be withheld from review. In addition, the objectives in Section 1.3 and in Section 6 of the FQ White Paper related to qualification of pebble fuel based on the Pebble-Bed Modular Reactor (PBMR), Pty Ltd approach should be withheld from further review at this time. The NGNP Project plans to update both the FQ and Mechanistic Source Terms white papers once the pending NRC requests for additional information (RAIs) are satisfactorily addressed.

RAI MST-9: Provide an analytical study evaluating the proposed use of a Fick's Law bulk diffusion approximation to model the combinations of more complex transport phenomena noted in Appendix Section C-4 (page 71) of the Mechanistic Source Terms White Paper.

Comments: The study should compare releases computed with the proposed diffusion model against releases computed with more rigorous transport models of individual or combined transport phenomena under steady state conditions and well as the cyclic conditions of flux and temperature experienced by fuel pebbles as they are cycled through the core (i.e., including holdup and breakthrough effects). The thus-estimated conservative and non-conservative potential distortions of releases computed with the proposed diffusion approximation should be summarized in a manner that conveys their significance in relation to overall release uncertainties.

Response MST-9:

As discussed in the response to RAI MST-62, the design goal is to predict fission-metal release from the core to within a factor of 10 at 95% confidence. It is anticipated that a transport model based upon Fick's 2nd Law (transient homogeneous diffusion) using effective diffusion coefficients and a convective boundary condition will be adequate to achieve and demonstrate that goal. This expectation is based in part on experience to date in comparing predicted and measured fission metal release in operating HTGRs such as Peach Bottom Unit 1 and Fort St. Vrain. The NGNP/AGR Fuel Technology Development Plan includes tests to develop the needed material property data (principally, the AGR-3/4 test) and to independently validate the design methods for predicting fission metal release (principally, the AGR-8 test).

While it may be conceptually ideal to employ more rigorous mechanistic transport models (e.g., a surface diffusion-trapping model for transport in graphite), such an approach is judged to be unnecessary if, as expected, the design goal can be met with a transport model based upon Fick's 2nd Law. The requisite single-effects data to derive the material properties needed for more complicated transport models (e.g.,

surface diffusivities, trap densities as a function temperature and fast fluence, etc.) are not available from the existing international database. Hence, the analytical study requested in the RAI cannot be conducted with the available data. Once the AGR-3/4 test data become available, alternative transport models would be considered to correlate the data if the current Fick's 2nd Law diffusion-based model is determined to be inadequate. If a more complex model is ultimately adopted, supplemental testing would likely be needed to obtain the supporting material property data.

RAI FQ-8/MST 10: Derive and explain the use of release-to-birth ratio (R/B) in terms of total mass balance (i.e., birth, decay, burnout, release) and the fission product inventory releases (including metallic fission products) resulting from transport through intact, defective, and failed CFPs. Identify the cases and situations in which it is not appropriate to use R/B in place of the release fraction

Response FQ-8/MST 10:

R/B is derived in Section 15.3.5 in Reference [1] and applied to in-pile fission gas release in Section 15.5.2 of the same reference. It is also derived and discussed in Section 3.3.1 of Reference [2]. The use of R/B in place of release fraction is appropriate for short-lived gaseous radioisotopes whose rates of production in and release from the kernel quickly (relative to the duration of irradiation) reach a steady-state. For long-lived radioisotopes whose inventory continues to increase throughout the irradiation, the use of release fraction is more appropriate. R/B is generally applied to the release of short-lived fission gases from exposed kernels (as a result of fuel manufacture or incremental particle failure during normal operation). Release fractions are used for the release of long-lived metallic fission products from intact fuel particles, particles with defective SiC, and failed particles, and for releases during heating tests when no fission product births are taking place.

References:

1. D. R. Olander, *Fundamental Aspects of Nuclear Reactor Fuel Elements*, Technical Information Center, Energy Research and Development Administration, 1976.

2. IAEA-TECDOC-978, *Fuel Performance and Fission Product Behaviour in Gas Cooled Reactors*, International Atomic Energy Agency, November 1997.

RAI FQ-9/MST-11: Describe how the nuclide birth and inventory values used in evaluating past and planned MTR-based TRISO-fuel-irradiation tests are calculated.

Comments: The response should describe the computer codes (e.g., JMOCUP and MCNP for the ATR irradiations, INVENTAR and/or KORIGEN for the FRJ2 and HFR irradiations) and modeling approximations (e.g., test geometry, particle double heterogeneity) used in analyzing the respective irradiations and how the computed isotopic burnup results are checked or calibrated against PIE isotopic measurements.

Response FQ-9/MST-11:

Nuclide birth and inventory values from ATR irradiations are determined from JMOCUP depletion analyses that use coupled MCNP5 and ORIGEN2 computer codes. This depletion analysis is executed with 24 hour, or less, time steps with ATR operation and configuration data averaged over a given 24 hour time interval. ATR and AGR experiment isotopics are updated and used as input for the next time step. AGR test fuel results are provided for each analysis cell, which is one half compact in volume. End of irradiation inventories are provided by the JMOCUP analyses. Details describing the calculation methodology may be found in the reference [1]. Daily averaged birth rates for selected short-lived fission

gases are calculated for each AGR test capsule. These values are determined by the ORIGEN2 computer code with input from the JMOCUP analyses. Documentation of birth rate calculation methodology will be issued after the final results have been reviewed. Applicable results from the analyses will be compared to PIE data when available and will be documented in the PIE results report. However, preliminary PIE determined burnup of an AGR-1 compact is within 1% of the calculated JMOCUP value.

Reference:

1. J. Sterbentz, "JMOCUP As-Run Daily Depletion Calculation for the AGR-1 Experiment in ATR B-10 Position", ECAR-958, May 14, 2010.

MST-12: Clarify the statement on page 3 of the Mechanistic Source Terms White Paper regarding the generation and transport of each radiologically significant species of fission product. Explain whether the stated functional dependencies should also include diffusion through kernel and intact coating layers as functions of operating and accident conditions. Explain whether the list of conditions should also include neutron fluence and plutonium burnup.

Response MST-12:

As elaborated in Section 4.5.1 of the Mechanistic Source Terms (MST) White Paper, which discusses the reference General Atomics fission product transport codes, the dependencies referred to in the RAI are included. Four sources of fission product release from the core are modeled: (1) as-manufactured, heavy-metal contamination, (2) exposed kernels (all coating layers failed), (3) "partially failed" particles (particles with a defective or failed SiC coating but with at least one pyrocarbon coating intact), and (4) intact particles (also referred to as "standard" particles as opposed to defective or failed particles). These models take into account diffusion through the kernel and intact coating layers.

The total inventories of radionuclides in the core are calculated using fission weighted yields of U-235, Pu-239, and Pu-241 at the end of equilibrium core cycle (3-year fuel residence in the core). Hence, plutonium burnup as well as burnup of other fissile isotopes is accounted for properly in determining fission product generation rates, and plutonium burnup does not merit special note relative to burnup of other fissile isotopes on page 3 of the white paper.

The transport and release of gaseous and metallic fission products from the UCO fuel kernels and from failed and intact particles are modeled. The diffusion coefficients of volatile metals (e.g., Ag, Cs, Sr) in the UCO kernels are functions of burnup and temperature. The fission gases, including iodine and tellurium isotopes, released from the fuel kernels are completely retained by PyC and/or SiC coatings. The transport of volatile fission metals through the SiC coating of intact particles is modeled; diffusive release from intact particles is only predicted to be significant for Ag isotopes and tritium. The statement on page 3 of the Mechanistic Source Terms White Paper notes that generation and transport of radionuclides from the fuel kernel through the TRISO particle coatings is calculated.

As discussed in the response to RAI FQ-2/MST-2, there are data that indicate that the rate of diffusion of cesium in SiC is a function of fast-neutron fluence. However, cesium release from intact TRISO-coated fuel particles is expected to be small relative to its release from heavy metal contamination, failed particles, and degraded (or functionally failed) particles. The effects of fast-neutron fluence on cesium diffusion in SiC can be bounded in the fission product transport models by using the diffusion rate at the maximum design fast-neutron fluence in the core.

Based upon the available data, the sorptivity of the matrix materials used in fuel compacts in prismatic cores and in fuel spheres in pebble-bed cores appears to be unaffected by fast-neutron fluence. However,

the sorptivity of fission metals on nuclear graphite has been shown to increase with increasing fast fluence (evidently, neutron damage creates sorption sites), and this effect is modeled for prismatic fuel. Hence, in this regard, the statement on page 3 of the Mechanistic Source Terms White Paper will be expanded to include functional dependence on fast-neutron fluence when the paper is updated following resolution of responses to the RAIs.

RAI FQ-10/MST-13: Provide an analytical study evaluating how fast-neutron fluence > 0.1 MeV correlates to material damage (i.e., the total fluence spectrum folded with the respective material's displacements-per-target-atom (dpa) damage response function) for the specific material fluence spectra encountered in (a) NGNP service and (b) the irradiation tests conducted in MTRs for the fuel materials addressed in the white paper. The study should report how the evaluated correlations of >0.1 MeV fluence to material damage vary between the different spectral conditions encountered in material service versus testing.

Comments: The statement about fast fluence on page 48 of the white paper seems to suggest that only fast neutrons (E >0.1 MeV) can displace atoms, which is of course not true. A graphite-moderated HTGR spectrum generally tends to have more neutrons in the slowing-down range just below 0.1 MeV than does a water-moderated MTR spectrum. This suggests that, with equal fast fluences in the energy range E >0.1 MeV, an HTGR spectrum may displace more atoms than an MTR spectrum. The requested study should use calculations with dpa-based damage functions to evaluate the magnitude of such differences for the respective spectral conditions encountered in service versus testing. If HTGR material service spectra are found to displace significantly more atoms than the respective MTR test spectra, the study should include recommendations on how to account for such spectral differences in the planning and evaluation of MTR-based irradiation tests for HTGR materials.

Response FQ-10/MST-13:

An analytical study that evaluates fast-neutron fluence for material damage is provided in TEV-1278 [1]. The study shows that the different neutron spectra of the ATR and a representative prismatic design have an insignificant effect on material damage (total dpa).

Reference:

1. J. Sterbentz, *Analytical Neutronic Studies Correlating Fast-neutron Fluence to Material Damage in Carbon*, Silicon, and Silicon Carbide, TEV-1278, Idaho National Laboratory, June 2011.

RAI MST-14: Provide an expert review of the conflicting explanations for how carbonaceous dust is produced in pebble-bed and block-core HTGRs. Key reference documents should be provided and translated as necessary to facilitate the review. Summarize the review findings, including any significant outstanding questions and recommendations for resolving them, and revise or supplement the associated information in Table 5-1 and on page 48 of the Mechanistic Source Terms White Paper as warranted.

Comments: [Note: Efforts in response to this RAI should be coordinated with the ongoing INL/NRC interactions on dust issues.] The NRC requester is aware of two conflicting explanations for how the observed amounts of dust were produced in the AVR. On the one hand, an article published in 2008 by Rainer Moormann of Forschungszentrum (Research Center) Jülich (FZJ) attributes the source of the AVR's finer dust mainly to abraded A3-3 pebble matrix binder coke, noting its especially high affinity for

cesium.[22] On the other hand, a German article published in 1990 by Rudolf Nieder, the AVR's resident chemist, concludes that most of dust was chemically produced from catalyzed reactions of the relatively high CO and H_2 impurity levels in the AVR primary circuit.[23] A translated excerpt from the latter article follows:

"Carbon Transport

As 21 years of operating experience have shown, the relatively high impurity concentrations did not permanently impair AVR operations. This also applies to graphite dust, which probably occurred as a result of the relatively high CO concentrations. Dust has radiological significance because solid fission and activation products adsorbed on dust are transported in the circuit. Experience shows, however, that the contamination problem is easy to control.

The graphite corrosion reactions
$C + H_2O = H_2 + CO$ *(1)*
$C + CO_2 = 2CO$ *(2)*

can run in the reverse direction under certain conditions, i.e., carbon is deposited from mixtures of carbon monoxide and hydrogen [2][24]. These conditions are
- suitable catalyst,
- relatively high CO and H_2 *concentrations,*
- reducing gas atmosphere.

Deposition of carbon occurs mainly in the presence of metallic catalysts, e.g., with Fe but not metal oxides. Iron carbides are also effective catalysts. With highly alloyed steels, which have a protective oxide layer of chromium oxide or iron-chromium-spinels, a reduction to elemental iron in an HTR primary circuit atmosphere is not possible. On the other hand, with the carbon and low-alloyed steels used in the AVR steam generator, the Fe3O4 protective layer originally present is reduced to elemental iron at H_2/H_2O *ratios >10. Such relatively high hydrogen concentrations were always present in the AVR as the result of diffusion from the water-steam loop. Applying laboratory results to the AVR primary loop shows carbon deposition to have already started after a relatively short incubation time, whereupon continuing deposition led to the formation of iron carbides with a significantly stronger catalytic effectiveness [3][25].*

Using a simplified mass balance [4][26] one can estimate a dust production rate of 3 to 3.5 kg per year, meaning that after 21 years of reactor operation there should have been about 60 kg of chemically produced dust. The additional production of dust by mechanical means would likely be modest by comparison. Pebble breakage occurred only very occasionally. Pebble wear during operation was

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[22] R. Moormann, *Fission Product Transport and Source Terms in HTRs: Experience from AVR Pebble Bed Reactor*, Science and Technology of Nuclear Installations, Article ID 597491, June 2008.

[23] R. Nieder, *Schlußfolgerungen für die HTR-Chemie aus 21 Jahren Betrieb des AVR-Reaktors (Conclusions about HTGR Chemistry from 21 Years of AVR Operation)*, p. 133-137, Chemie im Kraftwerk (Chemistry in Power Plants) 1990.

[24] [2] E. Stolz, F. L. Werner, *Kohlenstofftransport in den Hochtemperaturreaktoren THTR und AVR (Carbon Transport in the THTR and AVR High Temperature Reactors)*, atw. Februar 1968, S. 99-103.

[25] [3] M. R. Everett, *Some Aspects of Carbon Transport in High Temperature Gas Cooled Reactors*, Dragon Project Report DPR 365, 1965.

[26] [4] U. Wawrzik, *Interner AVR-Bericht (AVR Internal Report)*, 1984.

only trivial; machining marks were still largely recognizable on visually inspected discharged pebbles. Grain-size analysis of dust samples showed particle diameters consistently <1 micrometer, and predominantly <0.5 micrometer [5][27], a finding likewise consistent with chemically produced dust.

Independent of these experimental findings, there are model calculations that can determine the carbon deposition rate for given chemical conditions such as catalysis, etc., in an HTR primary loop [6][28]. Based on a rough approximation, the calculated amount of deposited carbon is found to be of the same order of magnitude as actually observed. The calculations, however, are being continued."

Response MST-14:

Dust effects have been the subject of ongoing interaction within a working group involving experts from the NRC and NGNP Project, including expert representatives from INL, HTGR vendors, universities, and various international organizations. Workshops on dust phenomena in the modular HTGR were conducted in November 2009, and in March 2010. A Dust Issues Assessment meeting was held from March 14-16, 2011. Each of these meetings was attended by staff from the NRC and the NGNP Project under the auspices of the Department of Energy (DOE)-NRC Memorandum of Understanding for the NGNP Project.

A document that describes potential modular HTGR dust safety issues and research and development needs was drafted in advance of the March 2011 assessment and will be revised based upon the discussions of that meeting. The document is scheduled to be completed by June 30, 2011. It will supplement the AGR Fuel Qualification and NGNP Methods Technical Development Plans and will also be used in the development of Calls for Proposals for DOE-funded university research. It is expected that these ongoing interactions will lead to a consensus regarding which dust phenomena are important to each modular HTGR design variant.

The general consensus of this working group appears to be that the primary source of dust in modular HTGRs is mechanical abrasion of graphitic components in the reactor core and fuel handling system rather than CO disproportionation as postulated in the German paper by Dr. Rudolf Nieder, which is referred to in the comments that accompany this RAI. Dust generation in pebble bed reactors is believed to result primarily from friction between pebbles within the core, between pebbles in the core and the side reflector, and between pebbles and the fuel handling system. Smaller quantities of dust may be present in a prismatic reactor for a variety of reasons, including foreign material introduced during construction or refueling, friction or erosion of prismatic fuel and reflector surfaces exposed to helium, foreign material from interfacing systems, corrosion or erosion of metallic surfaces in the coolant system, or decomposition of carbon monoxide. Most of these sources of dust could also contribute in small amounts to additional dust generation in pebble bed reactors. It is noteworthy that JAEA has concluded that the HTTR prismatic core is not a significant source of the dust in the primary circuit.

The dust generation mechanism postulated by Dr. Nieder is possible if CO and H_2 concentrations are high in a reducing gas atmosphere, and in the presence of a suitable catalyst. Such catalysts include iron carbides and metals, but not metal oxides. However, Fe_3O_4 can reduce to metallic iron if H_2/H_2O is >10 (high levels of H_2 were present in AVR). As chromium oxides (as would be expected in the presence of

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[27] [5] U. Wawrzik, et al, *Staubverhalten im AVR Reaktor (Dust Behavior in the AVR Reactor)*, Jahrestagung Kerntechnik (Nuclear Technology Annual Conference), Travemünde,1988.

[28] [6] D. V. Kinsey, *The Estimation of Carbon Deposition in HTRs*, Dragon Project Technical Note DPTN/565, 1974.

high nickel alloys) are not a suitable catalyst, and provided that impurity concentrations are controlled, this mechanism should not be present in future modular HTGRs. Furthermore, this generation mechanism would be active in either a prismatic or pebble bed reactor. The apparent absence of graphite dust in Fort St. Vrain suggests that the mechanism was not a contributor to dust generation in that reactor.

Similarly, only dust from abrasion of graphite piston rings has been observed in the prismatic Japanese HTTR. No contribution from the core itself has been identified [1]. This further suggests that the mechanism postulated by Dr. Nieder is not a contributor to dust generation.

Reference:

1. INL/EXT-11-21097, "HTGR Dust Safety Issues and Needs for Research and Development", June 27, 2011

RAI FQ-11/MST-15: Describe how the material properties of fuel pebbles can affect fuel service conditions and performance, how accurately the material properties are known over anticipated service conditions (i.e., variable temperature, neutron fluence, helium impurity levels, etc.), how the properties have been measured, what additional data, if any, would be needed to adequately characterize the material properties of fuel pebbles, and what plans have been made to address such data needs.

Comments: It is noted that the properties of pebble matrix material A3-3 are nowhere addressed in the FQ, MST, and HTM white papers. It is nevertheless known that a number of pebble material properties affected the safety-related performance of AVR and THTR in various ways that were largely unanticipated. Examples of pebble material properties and their importance to fuel performance and reactor safety follow:

a) The mechanical strength and toughness of fuel pebbles determines how many of them will fracture in service, e.g., one known fracture per ~10,000 pebble circulations as observed in the AVR. Fragments from fuel pebbles that break in the core can reside on the core floor indefinitely and disrupt or block the local flow of intact pebbles.

b) Lessons from THTR operating experience suggest that the high pebble-to-pebble friction coefficients in dry helium and their decrease with increasing temperature will again create an inherent pebble flow profile instability characterized by a gradually growing ratio of inner- to outer-core pebble flow rates and increasing radial temperature gradients in the core and outlet plenum.

c) It has been hypothesized that fuel pebble friction and wear may contribute significantly to dust production.

d) The fact that pebble friction in helium decreases on adding water vapor suggests that partial compaction of the pebble-bed core could result from moisture ingress events and that the core compaction resulting from seismic events could increase in severe seismic scenarios that include moisture ingress.

e) The chemical properties of fuel pebble matrix material are known to affect fission product retention by the pebble, the fission product activity of dust produced by pebble abrasion, and the chemical reactions of pebble material with moisture, air, etc.

Related comments are also provided under RAIs FQ-3, FQ-6, and FQ-7.

Response FQ-11/MST-15:

The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977 "Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011. It is assumed that "material properties" in the RAI description were intended to be "mechanical properties" (e.g., strength, toughness, surface friction factor, abrasion resistance), since the fuel qualification white paper addressed plans for acquiring pebble fuel matrix material property data in Section 5.2.2.4.

RAI-FQ-12/MST-16: Revise the white paper to correctly refer to the A3-3 matrix material in fuel pebbles as "carbon composite," not "graphite."

Comments: This and other NGNP white papers incorrectly refer to the pebble matrix material as "graphite." Although often called graphite in the HTGR literature, the A3-3 matrix material in modern fuel pebbles in fact contains non-graphitized binder coke in addition to the graphite filler particles. The presence of non-graphitized coke in the A3-3 matrix material gives it chemical, mechanical, and tribological properties notably different from those of any grade of true graphite. Referring to pebble matrix material as "graphite" has led to unnecessary confusion and waste in the past and is likely to continue to do so until stopped.

Response FQ-12/MST-16:

Following resolution of the RAIs on the Fuel Qualification and Mechanistic Source Terms White Papers, the white papers will be updated to eliminate use of the term "graphite matrix" and any reference to matrix material as graphite.

RAI FQ-17/MST-22 - Comment: Where conservative estimates, approximations, assumptions, and results are noted in the white paper, describe also, how best-estimate (BE) and best-estimate-and-uncertainty (BEAU) mechanistic source terms would be calculated.

Comments: By definition, BE analysis precludes the use of conservative or pessimistic simplifications. Conservative estimates, approximations, assumptions, and results are noted on pages 39, 42, and 74 of the white paper without indicating how BE (50% one-sided probability) and BEAU (e.g., 95% one-sided probability) MST calculations would be performed.

Response RAI FQ-17/MST-22 - Comment:

Best estimate (BE) and best estimate and uncertainty (BEAU) mechanistic source terms will be calculated as part of the probabilistic risk assessment (PRA). The uncertainty distribution in the source terms and the resulting offsite doses will include 5% lower bound, 95% upper bound, and mean values to show compliance with and margins to the Top Level Regulatory Criteria, as appropriate. Conservative estimates generally will use the 95% upper bound value, for example, for the assessment of consequences of Design Basis Accidents (DBAs) in comparison with 10CFR50.34 requirements.

BE analysis does not preclude the use of conservative assumptions or simplifications when other data are not available to serve as the basis for a less conservative assumption. In such cases, BE represents the most reasonable and defensible interpretation of the available data. The passages from pages 39, 42, and 74 of the Mechanistic Source Terms White Paper cited in the comments that accompany this RAI are good examples of the application of this approach to available data.

RAI FQ-13/MST-18: Identify all chemical elements (and the chemical forms thereof) that can pass through structurally intact SiC (a) during power operation at temperatures up to (i) 1250°C and (ii) 1350°C (i.e., as might occur in undetected hot spots) and (b) during core heatup accidents at temperatures up to (i) 1500°C, (ii) 1600°C, and (iii) 1700°C. Indicate the yields of each element from the thermal-neutron-induced fission of U-235, Pu-239, and Pu-241, respectively, and estimate the total amounts that pass through the SiC layer.

Comments: Include all stable and radioactive nuclides that can pass through the SiC layer, including europium, palladium,[29] and other elements, regardless of whether they have direct radiological significance. This requested information should be used to inform the response to RAI FQ-14.

Response FQ-13/MST-18:

Silver is not fully contained by SiC under normal irradiation at temperatures in excess of about 1100°C. Silver, a noble metal, is expected to be in elemental form. There is preliminary evidence from post irradiations examinations that up to 1% of europium is released from UCO fuel particles to the matrix during normal operation, as well as more substantial releases during post-irradiation heating from both UO_2 and UCO fuel. Europium is expected from thermodynamic considerations to be present in UO_2 fuel as Eu_2O_3 and in UCO fuel as EuC_2. Cesium is well contained by intact SiC during normal irradiation and its release during post-irradiation heating may be a result of enhanced permeability associated with degradation of the microstructure of the SiC under accident conditions. Cesium in the fuel particle is generally considered to be in elemental form. Strontium release from UO_2 TRISO-coated particle fuel is measurable under accident conditions at 1600°C and becomes significant (on the order of 0.1%) at 1800°C. The chemical form of strontium is expected to be SrO in UO_2 fuel and SrC_2 in UCO. Palladium can attack SiC during normal operation, but does not penetrate the layer. It has been found to accumulate at the inner surface of the SiC during post-irradiation heating of UO_2 TRISO fuel. Palladium, also a noble metal, will be in elemental form, but in the SiC it can form Pd_2Si.

In the response to RAI FQ-4/MST-5, it was established that the influence of inventories of silver and palladium on the mobility and/or interaction of these materials with the SiC layer is minimal. In the response to RAI FQ-2/MST-2 it is pointed out that the main factor, besides temperature, influencing the diffusion of cesium in SiC is neutron fluence. Currently, there is no information regarding the influence of the inventory of europium on its mobility though SiC. Given the general lack of importance of the inventories of fission products with their interactions with the SiC layer, the requested information on fission product yields and calculation of total amounts that could pass through the SiC is not included in this response. If evidence is found in the PIE of fuel irradiated in the AGR Program that fission product inventories are important, it will be analyzed and reported.

Silver

The behavior of silver is addressed in the first three paragraphs of the response to RAI FQ-4/MST-5.

In the AGR Program, silver transport through intact SiC under irradiation, including irradiation temperatures in the range 1250-1400°C, will be evaluated by post-irradiation measurements of silver in the compact matrix and elsewhere in the irradiation capsules. The release of silver will be measured during post-irradiation heating to temperatures in the range 1600-1800°C. Additional information on silver release from particles will be available from individual microsphere gamma analysis in both as-

[29] J. H. Neethling, J. H. O'Connell, E. J. Olivier, *Palladium Assisted Silver Transport in Polycrystalline SiC*, HTR-2010 Topical Meeting, Prague.

irradiated and post-heating conditions. Evidence of silver mobility through intact SiC under irradiation has been measured in the on-going PIE of the AGR-1 irradiation.

Europium

The mobility of europium, which is expected to be in the form of EuC_2 during irradiation of UCO fuel [18], through intact SiC under long-term post-irradiation annealing (10,000 hours = 417 days) of UCO fuel particles at temperatures in the range 1200-1500°C has been demonstrated [8]. Evidence of the release of Eu during AGR Program irradiations over a wide range of temperatures up to 1400°C will be sought in PIE by examining components of irradiation capsules, especially the compact matrix, for deposited Eu. The release of Eu will be measured in post-irradiation heating tests at temperatures in the range 1600-1800°C, and holdup in graphite will be measured in the PIE of the AGR-3/4 test, which will contain designed-to-fail (DTF) fuel particles and rings of fuel matrix graphite and fuel element graphite at a variety of temperatures. Evidence of the mobility of europium through intact SiC under irradiation has been measured in the on-going PIE of the AGR-1 irradiation.

Cesium

An evaluation of experimental data has resulted in a model that uses fast-neutron fluence as a parameter in the Arrhenius temperature dependence of cesium diffusion through SiC under irradiation, as shown in Figure A-4 of Reference [19]. The influence of fast fluence in the model becomes insignificant at 1600°C.

In the AGR Program, the diffusion of cesium through SiC under irradiation will be evaluated by measuring Cs-134 and Cs-137 released from fuel particles and retained in the matrix of compacts and other locations within the capsules during PIE. Capsules with no exposed kernels identified by fission gas R/B will be used for these measurements. The PIE (e.g., leach-burn-leach) will determine if there are any fuel particles with defective or failed SiC. Data from the hot capsule (irradiated at 1400°C) in AGR-2 and capsules in the AGR-7 irradiation margin test will be of special interest. If cesium is measured outside fuel particles (in excess of that contributed by heavy metal contamination and defective/failed SiC), an analytic method based on a transient diffusion solution will be used to establish the diffusion coefficient of cesium in SiC under irradiation.

Based on the current PIE of AGR-1 fuel, no cesium above that expected due to contamination has been found, suggesting that enhanced diffusion via vacancies under irradiation may be a very small effect.

During post-irradiation heating tests of UO_2 TRISO-coated particle fuel, cesium release has been observed to increase after about 200 hours at 1600°C [2]. Metallographic examination of the SiC after heating for 500 hours at 1600°C shows evidence of microstructural damage that appears to increase with burnup. It was hypothesized that the damage is caused by fission product enhanced local changes of the SiC structure. The release of cesium is temperature dependent, increasing somewhat at 1700°C and markedly at 1800°C. The exact reason for changes in SiC structure has not been firmly established.

The release of cesium in post-irradiation heating tests in the range 1600-1800°C will be measured in the AGR Program. The microstructure of the SiC following heating will be carefully examined by metallography and scanning electron microscopy to evaluate the nature of any changes in the microstructure of the SiC and look for the presence of fission products, such as palladium.

Strontium

The fractional release of strontium measured from spherical elements of UO_2 TRISO-coated particle fuel is about 4E-6 after 300 hours at 1600°C and 1E-2 after 100 hours at 1800°C [Figures A-14 and A-20, respectively, Fuel Qualification White Paper].

The release of strontium in post-irradiation heating tests in the range 1600-1800°C will be measured in the AGR Program. The transport and holdup of strontium in fuel matrix and fuel element graphite will be measured under irradiation conditions at a variety of temperatures up to approximately 1100°C (graphite) to 1200°C (matrix) in the AGR-3/4 irradiation test. Measurements of strontium release from these materials will be made in post-irradiation heating tests at temperatures from 1600 to 1800°C.

Palladium

Palladium is known to interact with SiC under irradiation forming metallic nodules at sites of attack [15,20]. The attack is localized and highly variable and does not cause thinning of the SiC. The rate of attack was faster in out-of-pile tests in which a temperature gradient of 278°C/cm was applied than in irradiated coated fuel particles and faster in accelerated irradiations than in real-time irradiation tests [20]. While there are reports of the buildup of palladium at the IPyC-SiC interface in German high temperature safety tests [2] there are no reports of SiC corrosion by palladium in German irradiation tests, which were generally conducted at an acceleration factor relative to real-time HTGR irradiation of three or less.

An analysis of the depth of SiC corrosion by palladium in irradiated fuel particles under the assumption that the reaction was controlled by the release of palladium from the fuel kernel concluded that the maximum reaction depth depends on the amount of palladium released from the kernel to the one-third power [20]. In these irradiations burnup varied from 1.4 to 9% FIMA and average temperatures ranged from 1065 to 1510°C.

In another study, evidence indicated no threshold concentration necessary for palladium penetration of SiC and increasing amounts of palladium increased only the number of nodules in the SiC [21]. In these irradiations the amount of palladium per fuel particle varied by an order of magnitude and the average temperature ranged from 1000 to 1425°C. Temperature was the major factor affecting penetration of the SiC and was found to fit an Arrhenius relationship. Penetration rate data plotted in Reference 21 include results from the FTE-13 experiment [22] in which TRISO-coated PuO_2 kernels were irradiated to 70% FIMA. Included in this experiment are fuel particles with palladium concentrations per unit geometric surface area of the kernel (a metric for mass flux across a surface) about four times, or per unit geometric surface area of the inside of the SiC layer (a metric for reaction with the SiC) about 3 times, compared with values typical of LEU TRISO-coated fuel particles considered for use in an HTGR. Even with this significantly larger inventory of palladium associated with the irradiation of pure plutonium fuel, the overall palladium penetration was not significantly different than in the irradiation of LEU fuels.

In TEV-1022 [17], the inventory of palladium in an HTGR is calculated to be 1.49 times the inventory of palladium in the AGR-1 experiment. The cube root dependence of palladium penetration of the SiC from [20] indicates an effect of $(1.49)^{1/3} = 1.14$ on penetration due to the greater inventory of palladium in the NGNP relative to that in the AGR-1 experiment, whereas no effect of palladium inventory would be expected based on Reference [21].

Measurements of the distribution of palladium (including especially the SiC coating) in fuel particles irradiated over a wide range of burnup and temperature will be made in the PIE of the AGR Program to provide insight on the processes controlling Pd/SiC interaction. Similar measurements will be made

following heating tests in the range 1600 to 1800°C. Metallography and scanning electron microscopy are the principal tools that will be used.

References:

1. H. Nabielek, P. E. Brown, and P. Offermann, "Silver Release from Coated Particle Fuel," Nucl. Technol. 35, 483 (1977).
2. W. Schenk, G. Pott, and H. Nabielek, "Fuel Accident Performance Testing for Small HTRs," J. Nucl. Mater., 171, 19 (1990).
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12. I. Szlufarska, "Ag Transport through SiC: Modeling and Experiment," VHTR R&D FY10 Technical Review Meeting, Denver Colorado, April 27-29, 2010.
13. D. A. Petti, J. Buongiorno, J. T. Maki, R. R. Hobbins, and G. K. Miller, "Key Differences in the Fabrication, Irradiation, and High Temperature Accident Testing of US and German TRISO-Coated Particle Fuel, and Their Implications on Fuel Performance," Nucl. Eng. Design, 222, 281 (2003).
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15. R. J. Lauf, T. B. Lindemer, and R. L. Pearson, "Out-of-Reactor Studies of Fission Product-Silicon Carbide Interactions in HTGR Fuel Particles," J. Nucl. Mater. 120, 6 (1984).

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19. IAEA-TECDOC-978, *Fuel Performance and Fission Product Behaviour in Gas Cooled Reactors*, International Atomic Energy Agency, November 1997.

20. K. Minato, T. Ogawa, S. Kashimura, K. Fukuda, M. Shimizu, Y. Tayama, and I. Takahashi, "Fission Product Palladium-Silicon Carbide Interaction in HTGR Fuel Particles," J. Nucl. Mater. 172, 184 (1990).

21. T. N. Tiegs, "Fission Product Pd-SiC Interaction in Irradiated Coated-Particle Fuels," Nucl. Technol. 57, 389 (1982).

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RAI FQ-14/MST-19: Describe the potential cumulative effects that the passage of silver and other elements could have on the retentiveness and strength of intact TRISO coating layers and the data available/needed for evaluating such effects.

Comments: For example, as also noted in part under RAI FQ-1, researchers have hypothesized in the past that the cumulative effects of silver transport and/or palladium transport could alter the SiC grain boundaries, e.g.: "In the part played by silver it is not clear whether the release is determined by an independent diffusion process or whether silver and palladium first widen the SiC grain boundaries and can be regarded as precursors of SiC damage." [Jül-2234(1988)] One could further hypothesize that the cumulative effects of silver, palladium, and other passing elements on SiC grain boundaries could also increase the subsequent grain boundary transport of cesium and other elements.

Response FQ-14/MST-19:

While there has been some speculation that the diffusion of silver through SiC may be a precursor of SiC damage [1], that hypothesis has not been confirmed. Silver in very high concentrations (100 times greater than those expected in high burnup LEU fuel in a modular HTGR) will, at temperatures in the range 1200-1500°C, attack SiC, but at concentrations more typical of a modular HTGR, silver passes through intact SiC under normal irradiation conditions without evidence of attack [2]. Although in one out-of-pile experiment, silver transport via grain boundaries in SiC was found to take place only in the presence of palladium [3], this result has not been reported elsewhere.

There is evidence from metallographic examination of the SiC after heating for 500 hours at 1600°C of microstructural damage to the SiC that appears to increase with burnup [4]. It is hypothesized [4] that the damage is caused by fission product enhanced local changes of the SiC structure, resulting in increased permeability to the release of cesium under accident conditions.

In the AGR Program, the releases of metallic fission products such as silver, cesium, europium, and strontium from TRISO-coated particle fuel will be measured after irradiation (by mass balance of fission products outside of the fuel particles and within the irradiation capsule) and during post-irradiation heating tests. The irradiations will be conducted at a variety of temperatures, including margin tests up to 1400°C and over a wide range of burnup (up to 20% FIMA) and fast-neutron fluence (up to 5E25 n/m^2). If fission product releases are measured post-irradiation from particles with intact SiC, effective diffusivities under irradiation conditions for the released fission products will be established by use of an analytic method based on a transient diffusion solution. The TRISO layers (PyC and SiC) of the fuel particles will be examined by metallography and scanning electron microscopy post-irradiation and post heating to evaluate the microstructure and the presence of fission products. If there are effects of fission product accumulation and/or penetration on the permeability of TRISO layers, especially SiC, that are relatively wide spread across fuel particle populations as opposed to being isolated, the tests and examinations planned in the AGR Program will document them.

References:

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2. R. J. Lauf, T. B. Lindemer, and R. L. Pearson, "Out-of-Reactor Studies of Fission Product-Silicon Carbide Interactions in HTGR Fuel Particles," J. Nucl. Mater. 120, 6 (1984).

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4. W. Schenk, G. Pott, and H. Nabielek, "Fuel Accident Performance Testing for Small HTRs," J. Nucl. Mater., 171, 19 (1990).

RAI FQ-15/MST-20: Provide a discussion of mechanisms and conditions that can increase the diffusion or permeation release of metallic fission products from intact TRISO particles. It is noted that FQ White Paper describes "failure" mechanisms (in Section 3.1.2) but not mechanisms that can increase diffusive releases from that vast majority of CFPs whose SiC layers remain structurally intact.

Response FQ-15/MST-20:

Thermally activated diffusion is the principal mechanism that can increase diffusive releases of metallic fission products through intact SiC layers. The influence of temperature is captured in the Arrhenius relationships of diffusivities of, for example, silver, cesium, and strontium presented in Figure A-3 of Reference [1]. Another mechanism that is known to increase the diffusivity of cesium through intact SiC under irradiation is fast-neutron fluence as demonstrated in Figure A-4 of Reference [1]. The methods to be used in the AGR program for evaluating the release of cesium from particles with intact SiC are discussed in the response to FQ-2/MST-2.

References:

1. IAEA-TECDOC-978, *Fuel Performance and Fission Product Behaviour in Gas Cooled Reactors*, International Atomic Energy Agency, November 1997.

RAI FQ-16/MST-21: Describe in detail how the AGR test program will evaluate cesium diffusivity through intact SiC under (a) NGNP operating conditions as well as (b) accident heatup conditions.

Comments: The response should build on relevant information provided in response to RAIs FQ-1, FQ-2, FQ-13, FQ-14, and FQ-15.

Response FQ-16/MST-21:

In the AGR Program, the diffusivity of cesium through intact SiC under irradiation will be evaluated by measuring Cs-134 and Cs-137 released from fuel particles and retained in the matrix of compacts and other locations within the capsules during post-irradiation examinations. Cesium in the graphite fuel body will be measured by gamma scanning. Cesium on the metallic inner surface of the irradiation capsule will be measured by gamma counting the solution obtained by acid leaching of the surface. Cesium in the fuel matrix of the compacts will be measured by deconsolidating the compacts and gamma counting the deconsolidation solution; followed by two leaches and gamma counts of the residue of the deconsolidation; and finally by burning the residue followed by two leaches and gamma counts. In some cases, the residue from deconsolidation will be sieved to separate matrix material from particles before burning, leaching and gamma counting so that no contribution of cesium from particles with defective SiC (which would dominate) is included. Most of the compacts examined thus far in the AGR-1 PIE have indicated no evidence of defective SiC.

Capsules with no exposed kernels identified by fission gas R/B will be used for these measurements. Data from the hot capsule (irradiated at 1400°C) in AGR-2 and capsules in the AGR-7 irradiation margin test will be of special interest. If cesium is measured outside fuel particles (in excess of that contributed by heavy metal contamination and defective SiC from fuel manufacture), an analytic method based on a transient diffusion solution will be used to establish the diffusion coefficient of cesium in SiC under irradiation.

The release of cesium through intact SiC under accident conditions will be measured during heating tests as a function of time at temperatures in the range 1600 to 1800°C. Cs-134 and Cs-137 will be collected on deposition coupons that are replaced periodically during the test and then gamma counted. The presence of defective SiC will be manifested by a high release fraction of cesium (equivalent to the inventory of one or more fuel particles) early in the test.

RAI FQ-17/MST-22: Note: The review objectives of the white paper can be addressed in the near term without a response to this comment. The comment may nevertheless be suitable for discussion in another forum.) The NRC staff would benefit from enhanced access to the technical literature cited in the white papers and related technical reference documents. The central importance of TRISO fuel and fission product behavior to the NGNP safety case underscores the necessity of effective knowledge transfer and management in this area. Facilitated access to relevant technical information will aid knowledge transfer and help develop the specialized technical expertise needed for evaluating the NGNP safety basis and related technical issues. Further collaboration is needed on developing technical training materials and searchable technical reference collections. A significant number of supporting technical documents are still available only in German and will need to be translated. The NRC's HTGR knowledge management program has already sponsored translations of several documents and will continue doing so on a limited basis. NGNP stakeholders should share translated documents so as to avoid duplication of translation effort, develop a shared list of German documents still to be translated, and coordinate the sponsorship of translations by the respective stakeholders.

Response FQ-17/MST-22:

No response required.

RAI MST-23: Clarify how liquid water (as opposed to water vapor) can be present in the primary circuit as mentioned in Section C-6 of the MST white paper.

Response MST-23:

The temperature, pressure, and enthalpy of the feedwater are such that a steam generator tube leak at the feedwater end will discharge liquid water into the steam generator tube bundle and potentially into the steam generator lower plenum. It is not impossible for liquid water to be present in the primary circuit, but it is unlikely that any water introduced into the primary circuit would remain in liquid form in the longer term. In particular, with the location of the steam generator well below the reactor pressure vessel, after the Heat Transport System is tripped upon detection of a leak and the main loop circulator shutoff valve closes, it would be difficult to get liquid water into the reactor vessel other than by droplet entrainment in flowing helium during forced circulation.

Liquid water residing in the steam generator lower plenum would not be expected to vaporize while the system remains pressurized. As an example, the enthalpy of the feedwater in a recent prismatic modular HTGR design is approximately 840 kJ/kg, while the enthalpy of saturated water at primary system pressure is around 1272 kJ/kg, the enthalpy of saturated steam is around 2773 kJ/kg, and the saturation temperature of liquid and steam is about 287°C. Since the helium exit temperature from the steam generator is also about 287°C, no significant vaporization due to heat transfer from the steam generator lower plenum metal would be expected.

The liquid water from a steam generator tube leak could be in a mist or spray which can remove plateout activity from the steam generator tubes. The mist or spray could also be transported with the helium to other parts of the primary system or through the primary system relief valve, if it should open due to high pressure. Higher temperature surfaces such as the steam generator shroud and the core barrel in the flow path from the steam generator to the reactor inlet plenum would vaporize a significant fraction of the liquid entrained in the helium. Additional vaporization and chemical reactions would take place in the top reflector graphite. Any remaining liquid water that manages to reach the core graphite would be vaporized.

RAI MST-24: Expand on the entrainment discussion in white paper Section C-6.2 to also address the entrainment effects of non-break flow transients, break shock and vibration, and break-induced local flow reversal on dust, friable surfaces, etc.

Response MST-24:

No direct reentrainment measurements have been made for the conditions mentioned in the RAI (no direct reentrainment measurements have been made for the conditions mentioned in the RAI). The ex situ blowdown tests conducted on the Peach Bottom steam generator tube specimens altered the flow configuration. In the reactor, the tubes were in cross flow across the tube bundle. In the blowdown tests, the flow was longitudinal down the length of the tubes. This does not represent a 180 degree flow reversal, but it does represent a significant change in flow direction. When these data are compared with other blowdown data, no effects of this change in flow direction are evident.

In principle, the wall shear stresses induced by such conditions can be calculated with modern computational fluid dynamics (CFD) codes. It is also noteworthy that the primary circuit components,

including the steam generator tube bundle where most of the plateout activity accumulates, are subjected to significant acoustic and flow-induced vibrations during normal plant operation and are designed accordingly. To the extent that these phenomena result in additional reentrainment, the calculated plateout distributions during normal operation, which do not take these phenomena into account, are conservative.

The entrainment discussion the Mechanistic Source Terms White Paper referred to in the RAI supports a liftoff model that is a simple empirical correlation of measured liftoff fractions versus the shear ratio. Shear ratio is the ratio of the wall shear stress during depressurization to that during normal operation.

The helium pressure boundary in a modular HTGR is comprised of three interconnected pressure vessels, which are designed and constructed to ASME Section III Division 1 requirements. The failure of these vessels is not expected to be within the frequency range of the Licensing Basis Events (LBEs). However, various piping (e.g., helium purification system inlet and outlet piping) and small-diameter instrument lines are connected to these vessels and could potentially produce small to moderate-sized helium leaks. Based on the spectrum of leak/break LBEs selected with the PRA for the steam-cycle MHTGR, the largest failure of the helium pressure boundary is that of the primary helium relief valve nozzle, which is located near the top of the steam generator vessel. This failure would result in a 13 in^2 rupture. The coolant flow does not reverse for this event, and the peak shear ratio during such rapid depressurization accidents is typically <1.1.

The most reliable liftoff data currently available are from the COMEDIE BD-1 test during which four in situ depressurization tests were performed at successively higher shear ratios. The largest measured liftoff fraction at a shear ratio of 5.6 was <1%, and the liftoff model was shown to be conservative for all four shear ratios. By comparison, when limits are derived for allowable plateout activity in the primary coolant circuit during normal operation, a liftoff fraction of 5% is assumed for all radionuclides.

The large assumed liftoff fractions relative to the experimental data base, as well as the very high shear ratios (up to 5.6) evaluated in the depressurization tests, are expected to accommodate the uncertainty associated with a lack of data on the effects of non-break flow transients, break shock and vibration, and flow reversal.

RAI FQ-18: Explain the significance and utility of the R/B versus half-life plots shown in Figures 7 and 8 of the white paper. How is the plotted information used?

Response FQ-18:

R/B is often plotted versus half-life of fission gases to distinguish mechanisms of release. For Fick's Law diffusive release, R/B should be proportional to the square root of the half-life [1]. For knockout, R/B would be proportional to the half-life, and for direct recoil, R/B would be independent of half-life [1]. The slopes of the data plotted on a log-log scale in Figures 7 and 8 of the Fuel Qualification White Paper could be used to evaluate the release mechanisms of the fission gases. For example, pure Fick's Law diffusion would correspond to a slope of 0.5, but since other mechanisms contribute to release, the slope deviates from 0.5. Figures 7 and 8 in the White Paper are to illustrate the ability of the NOBLEG code to predict fission gas release from failed particles.

Reference:

1. D. R. Olander, *Fundamental Aspects of Nuclear Reactor Fuel Elements*, Technical Information Center, Energy Research and Development Administration, 1976.

RAI FQ-19/MST-25: Indicate the fission power levels achieved by reirradiation in the AGR program and the earlier German programs in relation to those needed to match the short-lived nuclide inventories expected to be present during an NGNP heatup accident.

Response FQ-19/MST-25:

The intention in the AGR program, similar to earlier German programs, is to perform reirradiation to provide inventories of short-lived radionuclides that will permit measurements of releases of these radionuclides under post-irradiation heating tests. There is no intention to produce inventories expected to be present during an NGNP heatup accident. Consequently, the fission power levels and irradiation times will be much less than those required to produce the inventories of short-lived fission products expected to be present during an NGNP heatup accident. The short-lived radionuclides produced by reirradiation will be generated within the structure of the fuel particles developed during irradiation conditions resulting in burnups at a variety of temperatures that will bound conditions within the NGNP prior to a heatup accident. The masses of fission products will be dominated by stable and long-lived isotopes, so the inventories within the test fuels will be prototypical.

RAI FQ-20/MST-26: Discuss the validation benchmark calculations that either have been or will be conducted for the German test data.

Comments: The fuel qualification testing plans for a pebble bed NGNP design should address how the German test data will be used to validate the code predictions of fission product releases from intact and failed TRISO-coated fuel particles.

IAEA Coordinated Research Project 6 recently evaluated analytical benchmark calculations performed by international participants using national codes to predict the experimentally measured releases of strontium from German TRISO-coated fuel particles during accident condition heat-up tests. The benchmark results showed calculated releases that consistently over-predicted the measured releases by several orders of magnitude. The evaluation of these results concluded that the legacy models for the temperature-dependent diffusion coefficient for Sr in SiC significantly over-predict the diffusion of Sr in SiC for the German fuel particles.

Although the results are conservative in this instance, the findings indicate uncertainties in either the basic separate-effects test data or the analysis of those test data in developing the diffusion coefficients for the German TRISO fuel particle components.

Response FQ-20/MST-26:

This RAI is primarily relevant to the Mechanistic Source Terms (MST) White Paper since it relates to the validation of fission product transport codes. Fuel irradiation and testing associated with fission product transport, specifically AGR-3/4 and AGR-8, are discussed in the Mechanistic Source Terms White Paper. Thus this response is in the context of the Mechanistic Source Terms White Paper.

German fuel irradiation and testing data are important constituents of a broad set of international experimental results on fission product transport in coated particle fuel that has been produced, exchanged, and subjected to international review over several decades. A primary example of data exchange and review is a document produced by the International Atomic Energy Agency (IAEA)[1]. A more recent example is benchmark fuel performance and fission product transport code comparisons conducted under the IAEA Coordinated Research Project on Conservation and Application of HTGR Technology: Advances in HTGR Fuel Technology, designated CRP-6. In CRP-6, comparisons of code to experiment were made for irradiation testing of German, Japanese and U.S. fuel and for heating tests of

German and Japanese fuel. As noted in the NRC staff comments that accompany this RAI, the heating test code results conservatively over predicted release.

Fission product transport models used and planned to be used by the NGNP Project for mechanistic source term calculations have been developed with consideration of this international database, including German data, for both the prismatic and pebble designs. The NGNP fuel development and qualification program incorporates testing to generate data for model development (AGR-3/4) and validation (AGR-8) of fission product transport codes. Near term NGNP fuel development activities have been adjusted to incorporate scope supporting pebble fuel particles. Specifically, LEU UO_2 TRISO fuel particles generally consistent with the German particle design and produced by Babcock and Wilcox, AREVA and PBMR, (Pty) Ltd. are currently under irradiation in compacts in the AGR-2 test train in the ATR at INL. Building on the ATR irradiations that are currently underway, if the pebble bed concept is selected updated information regarding the revised plan for pebble bed fuel qualification will be provided once that plan is established and those additional details are available. It is expected that the scope and objectives of the revised pebble bed fuel plan will build on the existing plan and be adjusted for pebble bed fuel specific design and service conditions. In the course of the NGNP Project, fission product transport models will be reviewed and, if appropriate, revised based on data generated by the project as well as relevant international data, and validated based on data generated by the project. A more detailed response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977 "Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011.

Reference:

1. IAEA-TECDOC-978, "Fuel Performance and Fission Product Behaviour in Gas-Cooled Reactors" November 1997.

RAI FQ-21/MST-27: Since the German UO_2 test data for TRISO-coated particle diffusion rates are largely based on post-irradiation heating tests, discuss how the additional testing for pebble bed reactor fuel will evaluate fission product transport under high temperature irradiation.

Comments: In TEV-1022, INL states: "To accurately model fission product transport in TRISO-coated particle fuel under high temperature irradiation, use of 'effective' diffusion coefficients for the kernel and coatings (as presented in IAEA-TECDOC-978 [1]) obtained from post-irradiation heating tests is not recommended because those coefficients do not consider the irradiation effects, either implicitly or explicitly."

Response FQ-21/MST-27:

Data on TRISO-coated particle diffusion rates based on in-pile testing have been analyzed and used to construct diffusion correlations, as is evidenced for the case of cesium documented in Reference 1. An IAEA Coordinated Research Project is in the early phases of forming. As part of this project, it was recommended to critically review and analyze historical data on both in-pile and out-of-pile fission product diffusion in TRISO-coated particle fuel that will enhance the database.

In the AGR Program fuel will be irradiated at high temperatures to test margins in the ongoing AGR-2 irradiation and in the AGR-7 irradiation. Releases of fission products under irradiation will be measured

in PIE and evaluated with an analytic method based on a transient diffusion solution to establish diffusion coefficients under irradiation. These data will be compared to the historic values from Reference 1.

References:

1. IAEA-TECDOC-978, *Fuel Performance and Fission Product Behaviour in Gas Cooled Reactors* International Atomic Energy Agency, November 1997.

RAI FQ-22: Discuss the safety classification of the NGNP fuel components (e.g., kernel, coatings, matrix, graphite block) as well as the development of extensive and unique special treatment requirements for the fuel components.

Comments: Section 1.4 does not mention the relationship of the Fuel Qualification white paper to the System, Structure, and Component (SSC) Safety Classification white paper.

Response FQ-22:

The HTGR fuel is the central feature of the safety design approach. The fuel particles (including both the fuel kernels and the fuel particle coatings) are the primary barrier for the radionuclide retention function during normal operation and off-normal events. The fuel particles perform the required safety function of radionuclide retention during Design Basis Events (DBEs). For this reason they are classified as safety related. Similarly, the prismatic fuel elements and the pebble bed fuel elements in which the fuel particles are contained are relied on during DBEs for the function of radionuclide retention. The prismatic fuel element design has the fuel particles in cylindrical fuel compacts inserted in blind holes separated from the coolant holes within the hexagonal fuel blocks; the pebble bed fuel element has the fuel particles within the spheres with an outer spherical layer of carbonaceous material that separates them from the coolant. The prismatic fuel compacts and both fuel element types are classified as safety-related.

The HTGR fuel elements also contribute to the required safety function of controlling heat generation by providing an important contribution to the highly negative temperature coefficient. They contribute to the required safety function of core heat removal by contributing to the passive heat transfer to the reactor vessel. They contribute to the required safety function of control of chemical attack by withstanding the postulated ingress of chemical impurities such as steam and air. They contribute to maintaining core geometry for all three of these safety functions.

These safety classifications are consistent with the results that are expected to be obtained when the safety classification process described in the Structures, Systems, and Components Safety Classification White Paper is applied to the NGNP.

To assure its capability and reliability the modular HTGR fuel will be subjected to a spectrum of special treatment requirements commensurate with its safety importance. This will include compliance with Quality Assurance (QA) requirements during fuel design and development, test capsule irradiation and post irradiation heatup testing, and manufacturing. Fuel performance will be monitored via primary coolant activity surveillance, as discussed in the response to RAI FQ-23/MST-28, and via post irradiation examination of fuel removed from the reactor, as discussed in the response to RAI FQ-26/MST-31.

RAI FQ-23/MST-28: (a) Discuss how the regulatory requirements for technical specifications (TS), as noted in the comments below, will be applied to the NGNP fuel design. (b) Discuss whether the NGNP TS will contain requirements for controlling the initial accident source terms to those assumed in the accident analyses by monitoring and limiting gaseous fission product releases (for controlling the fraction

of failed fuel particles in the core during normal operations) and monitoring and limiting metallic fission product releases (for controlling releases from failed and intact fuel particles during normal operations).

Comments: The white paper discusses NUREG-0800, Standard Review Plan, Section 4.2, "Fuel System Design." Additionally, NUREG-0800, Standard Review Plan, Chapter 16 "Technical Specifications" states that at the Operating License stage, compliance with Title 10 Code of Federal Regulations (10CFR) 50.34 requires applicants to propose TS in accordance with 10CFR50.36. 10CFR50.36 requires proposed TS to include the following:

- 10CFR50.36(c)(1)(i)(A) Safety Limits. Safety limits apply to important process variables necessary for an appropriate level of protection for the integrity of certain physical barriers that guard against the uncontrolled release of radioactive material.
- 10CFR50.36(c)(2) Limiting Conditions of Operation (LCOs). A TS LCO of a nuclear reactor must be established for each item meeting one or more of the following 10CFR50.36(c)(2)(ii) criteria:

  (ii) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

  (iii) Criterion 3. An SSC that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

Also, the Section 3.4.16 of the standard technical specifications for pressurized water reactors includes primary coolant activity limits.

Response FQ-23/MST-28:

A comprehensive set of Technical Specifications will be proposed by the license applicant for the single-module NGNP demonstration plant (and for follow-on commercial modular HTGRs) to assure that safety-related systems, structures, and components meet design requirements throughout their service lifetimes.

It is early, with design still underway and the license applicant having not yet been identified, to discuss specific commitments for the content of the Technical Specifications for the NGNP Plant. However, it is instructive to examine the Technical Specifications for FSV for indications of the kind of Technical Specifications that were in force for that facility and the content that generally might be likely to be included in the NGNP Technical Specifications. Although FSV was not a modular HTGR, and its fuel performance and accident response characteristics were significantly different from those of a modular HTGR, the particular FSV Technical Specifications discussed in this RAI response are, with the exception of the reactor core safety limit, relatively generic in nature.

FSV Technical Specifications discussed in this response are the reactor core safety limit, those for primary helium activity levels and secondary coolant (steam system) activity, and the associated surveillance requirements. These FSV Technical Specifications and their bases are summarized in this response. Potential application of similar specifications to the NGNP is also discussed, although specific commitments at this time would be premature.

Similar to many other nuclear power plants that received their operating licenses in the 1960s and 1970s, the FSV Technical Specifications, which were initially issued in 1973, were written in a plant specific,

customized format. When the NRC developed its Standard Technical Specifications (STS) program in the late 1980s, many of the plants operating under custom technical specifications participated in a program to convert to the STS format. FSV began an effort to convert its Technical Specifications to the STS format. However, this effort was terminated in late 1989 when the licensee, Public Service Company of Colorado, decided to end operations at FSV. Only one of the later amendments to the Operating License and the Technical Specifications were issued in the STS format, and the remaining Technical Specifications not affected by that amendment were never converted to the new format. This background explains why the numbering system for the FSV Technical Specifications discussed in this RAI response may be unfamiliar to those who are most familiar with the STS format.

The following FSV Technical Specifications (Safety Limits [SL], Limiting Conditions of Operation [LCO], and Surveillance Requirements [SR]) are summarized below:

- SL 3.1, Reactor Core
- LCO 4.2.8, Primary Coolant Activity
- LCO 4.3.8, Secondary Coolant Activity
- SR 5.2.6, Plateout Probe Surveillance
- SR 5.2.11, Primary Reactor Coolant Radioactivity Surveillance
- SR 5.3.7, Secondary Coolant Activity Surveillance

SL 3.1, Reactor Core

The FSV reactor core safety limit, SL 3.1, was applied to limiting combinations of core thermal power and core helium flow rate, with the objective of maintaining the integrity of fuel particle coatings. Limits were imposed over the lifetime of each fuel reload segment on the reactor power-to-flow (P/F) ratio and on the total integrated operating time at specified ranges of P/F ratio. Corrective actions (consisting of reduction of reactor power within specified time limits) were required should these integrated time limits be exceeded.

The Basis for SL 3.1 was focused on the potential for fuel particle failure due to kernel migration, which was thought in the early 1970s to be the controlling fuel particle coating failure mechanism for the HTGR. SL 3.1 was a complex safety limit, and assessing compliance with the limit was a complex procedure that was neither operator friendly or transparent for the regulatory staff. Based on advances in understanding of coated fuel particle performance and failure modes over the last 35 to 40 years, it is now known that fuel kernel migration is not the controlling fuel particle coating failure mechanism.

Further work needs to be done to determine whether a reactor core safety limit and/or some other safety limit is most appropriate for the modular HTGR. Safety limits need to be based on the current understanding of coated fuel particle performance and modular HTGR plant performance and need to be implemented in a manner that is transparent for both the operating staff and the regulatory staff. Specific commitments at this time would be premature.

LCO 4.2.8, Primary Coolant Activity

LCO 4.2.8 imposed limits on radionuclide activity in the primary coolant helium. Limits were imposed on circulating noble gas beta plus gamma activity, circulating halogen activity (expressed as an iodine-

131 thyroid dose equivalent), plateout halogen inventory (also expressed as an iodine-131 thyroid dose equivalent), and plateout inventory of strontium-90. Determination of circulating noble gas activity was required at least once per month or at each time when primary coolant radioactivity concentration changed (increased) by 25% from the previous measurement at the same reactor power level.

The Basis for LCO 4.2.8 limits on primary coolant noble gas beta plus gamma activity were focused on a postulated accident sequence in which the prestressed concrete reactor vessel (PCRV) was rapidly depressurized and the entire circulating inventory was assumed to be released to the reactor-building and to the atmosphere through the reactor-building louvers. The circulating activity limit was calculated based on an accepted whole body gamma dose at the FSV Exclusion Area Boundary (EAB) under certain assumed weather conditions and short-term atmospheric dilution factors.

The Basis for LCO 4.2.8 limits on iodine-131 equivalent circulating and plateout inventories and strontium-90 plateout inventory were focused on another postulated accident sequence in which the PCRV was rapidly depressurized (at a faster rate than the accident sequence referred to above) and the entire circulating inventory, 6% of the plateout halogens, and 5% of the plateout strontium is carried out of the PCRV and out of the reactor-building louvers. The limits were based on the resulting doses at the EAB under certain assumed dilution factors, weather conditions, and dose equivalent affectivities derived from ICRP publications.

Noble gas circulating inventory was calculated from primary coolant helium grab samples and from readings of the primary circuit gross gamma activity monitor. The inventory of any non-measured noble gas nuclides was calculated by assuming, based on theoretical investigations and experiments, that the steady-state release rate of such radionuclides is proportional to the square root of the radionuclide half life.

The methods for determining iodine-131 circulating and plateout inventory were not directly described in the basis of LCO 4.2.8, but SR 5.2.6, Plateout Probe Surveillance, discusses analysis of the plateout probes for iodine-131.

Strontium-90 inventory was determined by analysis of the plateout probes in the primary circuit. Estimates of strontium-90 inventory between plateout probe analyses were determined based on the most recent plateout probe analysis and calculations that take into account strontium-90 decay and krypton-90 precursor decay.

Although specific design features and Licensing Basis Events (LBEs) for the NGNP have not yet been determined, it is likely that a Technical Specification similar in intent to FSV LCO 4.2.8 will be proposed for the NGNP. Specific commitments at this time would be premature.

LCO 4.3.8, Secondary Coolant Activity

LCO 4.3.8 imposed limits on the radionuclide activity in the secondary coolant of 0.009 μCi/cc for iodine-131 and 6.8 μCi/cc of tritium. Although the FSV main steam pressure was higher than the primary system pressure, the reheat steam system operated at a pressure lower than the primary system pressure, so the limits of this technical specification took into account possible leakage of primary coolant helium into the reheat steam system.

The Basis of LCO 4.3.8 was focused on a postulated accident sequence involving loss of offsite power, main turbine trip, and failure of the diesel generator to start. In this event, about 52,000 gallons of water would be vented to the atmosphere as steam. The limits were based on resulting two hour thyroid and whole body exposure doses at the EAB.

Although specific design features and LBEs for the NGNP have not yet been determined, it is expected that steam pressure will be higher than primary coolant helium pressure such that radionuclide content of the steam system generally will be limited to tritium that enters the system by diffusion through steam generator tubes. Based on this consideration it is possible that a Technical Specification similar in intent to FSV LCO 4.3.8 could be proposed for the NGNP, but it would have significantly different content and bases. Specific commitments at this time would be premature.

SR 5.2.6, Plateout Probe Surveillance

SR 5.2.6 provided requirements for surveillance of the primary circuit plateout probes. FSV had two plateout probes installed in the primary circuit, one in each loop in PCRV penetrations extending into the steam generator shrouds and then into the helium coolant gas stream. Measurements from the probes were used to assess compliance with the strontium-90 and iodine-131 plateout limits of LCO 4.2.8. Removal of a plateout probe for analysis was required at the second, fourth, and sixth refueling and at intervals not to exceed five refueling cycles thereafter. Provisions were made for more frequent probe removal and analysis in the event of significant increases in primary coolant noble gas circulating activity (25% or more over the average activity of the previous three months at the same reactor power level) and measured circulating activity exceeding 25% of the Final Safety Analysis Report (FSAR) "design" level.

Although specific design features for the NGNP have not yet been determined, it is likely that a Technical Specification similar in intent to FSV SR 5.2.6 will be proposed for the NGNP. Specific commitments at this time would be premature.

SR 5.2.11, Primary Reactor Coolant Radioactivity Surveillance

SR 5.2.11 provided requirements for surveillance of primary coolant circulating activity. FSV had the capability to take and analyze grab samples of the primary coolant. In addition, a continuous primary coolant activity monitor was provided. Measurements from these features were used to assess compliance with the circulating noble gas activity limits of LCO 4.2.8.

A grab sample of primary coolant was to be analyzed for its radioactive constituents at a minimum of once per week during reactor operation and used to calibrate the continuous primary coolant activity monitor. If the continuous activity monitor were to become inoperable, or if the primary coolant activity were to reach 25% of the limits of LCO 4.2.8, or if the primary coolant activity level were to increase by 25% over the previous equilibrium value at the same reactor power level, the frequency of sampling and analysis was to be increased to a minimum of once per day until the activity level decreased or reached a new equilibrium value, at which time weekly sampling could be resumed.

Although specific design features for the NGNP have not yet been determined, it is likely that a Technical Specification similar in intent to FSV SR 5.2.11 will be proposed for the NGNP. Specific commitments at this time would be premature.

SR 5.3.7, Secondary Coolant Activity Surveillance

SR 5.3.7 provided requirements for surveillance of secondary coolant activity. The secondary coolant was to be analyzed for iodine-131 and tritium, as well as gross beta plus gamma content, once per week during reactor operation to assess compliance with the limits of LCO 4.3.8. If the secondary coolant activity level were to reach 10% of the LCO 4.3.8 limits, the frequency of sampling and analysis was to be increased to a minimum of once each day until the activity level decreased to less than 10% of the limit, at which time weekly sampling could be resumed.

Although specific design features for the NGNP have not yet been determined, it is likely that a Technical Specification similar in intent to FSV SR 5.3.7 will be proposed for the NGNP. Specific commitments at this time would be premature.

It is anticipated, but subject to final design decisions, that the following instrumentation would be included in the plant design to demonstrate compliance with the technical specifications related to radionuclide control:

- Ion chambers for continuous measurement of total gamma + beta activity in the primary coolant.
- Sampling and analysis system for measuring release rate-to-birth rate ratios (R/Bs) for individual noble gases.
- Plateout probes for measuring core release rates of condensable radionuclides (I-131, Cs-137, Sr-90, etc.).
- Sampling stations and instrumentation (e.g., beta spectrometers) for determining an overall tritium plant mass balance.
- In situ gamma scanning equipment for measuring the plateout activity on accessible primary circuit components
- Standard radiochemistry instrumentation and laboratory for environmental monitoring.

The core-average fuel failure fraction can be estimated from the measured fission product release rates from the core (see Appendix C of the Mechanistic Source Terms White Paper). While there is currently uncertainty in such calculations, the design methods for predicting fuel failure and fission product transport will have been verified and validated by the AGR Fuel Development and Qualification Program to have the required predictive accuracies before the NGNP begins power operation. See the response to RAI MST-62 for accuracy goals.

In addition to the measurements of primary and secondary coolant activity levels, it is likely that selected spent fuel elements discharged from the reactor will undergo post-irradiation examination to confirm that the fuel has performed as required. Fuel surveillance is addressed in the response to RAI FQ-26/MST-31.

In summary, the NGNP Technical Specifications are likely to include one or more appropriate safety limits and provisions for monitoring the fission product release rates from the core, from which the fuel performance can be inferred, thereby helping to ensure that the source terms for LBEs will be less than assumed in the Safety Analysis Report. Technical Specifications will be developed as the design and safety analysis progress for the modular HTGR. Insights from FSV as described in this RAI response and from other HTGRs will be taken into account. Since the safety design approach for the modular HTGR is centered on radionuclide retention within the fuel, and since normal operation fuel performance is a strong indicator of the fuel's radionuclide retention during off-normal events, the technical specifications that are closely related to fuel performance and radionuclide control will receive particular focus.

RAI FQ-24/MST-29: Discuss how the "functional failure" mechanism indicated in the comments below will be addressed in the calculation of MSTs.

Comments: Significantly higher than predicted diffusion/release of metallic fission products from the coating layers of intact fuel particles could be considered a "functional failure" of TRISO-coated fuel particles. Elevated and potentially unacceptably high fission product diffusion could result, for example,

from significantly higher than expected fuel particle operating temperatures. See also related RAIs MST-1, MST-2, MST-18, MST-19, MST-20, and MST-21.

Response FQ-24/MST-29:

"Functional failure" of coated fuel particles, as referred to in Appendix C of the Mechanistic Source Terms White Paper (Section C-2, page 67), refers to those circumstances under which fuel particle coating structural degradation occurs to the point that the radionuclide retention capabilities of the particle are reduced relative to a defect-free TRISO fuel particle. In such cases, the TRISO coating may, for some degradation mechanisms, appear to be structurally intact, but the fission product retention capability is degraded, hence the use of the term "functional failure". As used in the white paper, "functional failure" is not intended to include the circumstances of diffusion/release of metallic fission products from the coating layers of intact fuel particles (either at predicted or higher than predicted operating temperatures) as presented in the comments that accompany this RAI.

For example, diffusive release of silver from an intact TRISO particle at high temperature is not an indication of "functional failure" because these radionuclides can diffuse (or "permeate") through SiC coatings that meet all of the as-manufactured SiC coating specifications. The most important mechanisms that can lead to "functional failure" of the SiC coating are fission product induced SiC corrosion and SiC thermal decomposition. The former dominates during normal operation, and the latter contributes during core heatup accidents. Models based upon experimental data are used to predict such SiC degradation as a function of time and temperature, and criteria are established for when the coating is "functionally failed" (e.g., corrosion of 50% of the SiC coating thickness). At this point, no further credit is taken in the models for the SiC coating as a fission product retention barrier. However, for this example, the pyrocarbon coating layers would remain intact for some period of time following functional failure of the SiC coating, so the particle would continue to retain gaseous fission products and would not be considered to be totally failed.

Regardless of this distinction regarding use of the term "functional failure", the analysis methods used to calculate mechanistic source terms for the HTGR take into account fission product release from intact, degraded (or functionally failed), and totally failed fuel particles under both normal operating and accident conditions.

RAI FQ-25/MST-30: Discuss how analytical predictions of the NGNP fuel design (in connection with the NGNP fuel qualification program or the NGNP safety analysis) will be conducted to ensure that the fuel design bases are met. If analytical predictions are to be provided, discuss whether the analytical models and methods for these analytical predictions will be developed and evaluated in accordance with RG 1.203, "Transient and Accident Analysis Methods."

Comments: It is noted that Section 5 of the white paper makes no mention of "analytical predictions."

Response FQ-25/MST-30:

As described in Section 4 and in Appendices D and E of the Mechanistic Source Term White Paper, full-core analysis codes are used to predict fuel performance and fission product release from both prismatic and pebble-bed cores during normal operation and postulated accidents. These codes contain component models for predicting the in-core performance of the reference TRISO fuel particles and radionuclide transport in core materials. These design methods will be verified and validated using the existing international data base [1] and the results of the NGNP/AGR fuel development program [2]. In particular, the AGR-7 and AGR-8 tests are planned to provide design-specific integral test data for code validation.

The protocols and procedures for these code validation activities will be informed by and consistent with NRC Regulatory Guide 1.203.

As discussed in response to RAI MST-72 a broad spectrum of approaches has been used to develop design methods for predicting HTGR source terms. While the approaches have been diverse, the models and material property correlations used to predict TRISO fuel performance and radionuclide transport in support of reactor design and safety analysis are, in general, based upon experimental data that have been correlated with phenomenological models based on first principles, rather than simply n^{th}-order polynomial fits of the data. Often, correction factors are added to the first-principles model to account for irradiation effects. The technical bases for the reference models and material property correlations for predicting fuel performance and radionuclide transport that have historically been used in the U.S. HTGR program are summarized in [3].

References:

1. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," TECDOC-978, International Atomic Energy Agency, November 1997.
2. "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," PLN-3636, Rev. 0, Idaho National Laboratory, September 2010.
3. Martin, R.C., "Compilation of Fuel Performance and Fission Product Transport Models and Database for MHTGR Design," ORNL/NPR-91/6, Oak Ridge National Laboratory, October 1993.

RAI FQ-26/MST-31: For the first-of-a-kind NGNP prototype plant, loaded with fuel from a first-of-a-kind prototype fuel production facility, discuss whether the initial fuel core loading will involve any special surveillance procedures and/or testing programs beyond those that would be planned for a follow-on commercial version of the NGNP and fuel production facilities. This might involve special fuel operational performance fission product release monitoring and/or selected fuel pebble or compact post-irradiation accident simulation testing.

Response FQ-26/MST-31:

It is early, with design still underway and the license applicant having not yet been identified, to discuss specific commitments for the content of the various testing programs for the NGNP demonstration plant.

This response provides brief conceptual summaries of the test programs currently envisioned for the modular HTGR demonstration plant. Much of this information is also provided in the response to RAI FQ-3/MST-3. However, specific commitments at this time would be premature. These will be addressed at the appropriate time by the license applicant.

As discussed in the response to RAI FQ-3/MST-3, it is anticipated that multiple test programs, including a comprehensive operational surveillance program, will be conducted to confirm that the demonstration plant meets design requirements with strong emphasis on safety-related SSCs, especially the fuel which is critical to the safety case. As discussed in the response to RAI FQ-23/MST-28, the surveillance program will include multiple elements to confirm that radionuclide control requirements are met at the required confidence level. This program would include provisions for monitoring the fission product release rates from the core from which the fuel performance can be inferred, thereby assuring that the initial primary

circuit radionuclide inventories for any postulated accident would be less than assumed in the Safety Analysis Report.

The NGNP/AGR base technology fuel development program will develop and qualify the reference TRISO fuel for the modular HTGR demonstration plant. In addition, if significant changes were made to the fuel production equipment or processes thus deviating from those used for the AGR qualification fuel, it is expected that an irradiation proof test of the mass-produced fuel for the initial core would be conducted by the NGNP Project and/or the fuel vendor. This proof test would include PIE and post-irradiation heating tests expected to be largely confirmatory of AGR-5/6 and AGR-7/8.

It is anticipated that the following instrumentation would be included in the plant design to monitor fuel performance during initial plant operation and throughout its lifetime:

- Ion chambers for continuous measurement of total gamma + beta activity in the primary coolant.
- Sampling and analysis system for measuring release rate-to-birth rate ratios (R/Bs) for individual noble gases.
- Plateout probes for measuring core release rates of condensable radionuclides (I-131, Sr, Cs-137, Sr-90, etc.).
- Sampling stations and instrumentation (e.g., beta spectrometers) for determining an overall tritium plant mass balance.
- In situ gamma scanning equipment for measuring the plateout activity on accessible primary circuit components

In addition to the on-site measurements, it is further anticipated that selected spent fuel elements discharged from the reactor will undergo post-irradiation examination to confirm that the fuel has performed as required to support the safety case. Although it is premature to make specific commitments for fuel surveillance in the first of a kind NGNP demonstration plant, it is instructive to examine the fuel surveillance program that was followed for the first of a kind FSV plant. It is reasonable to expect that a fuel surveillance program with some elements that are consistent with the intent of the FSV program would be implemented for either a prismatic or pebble bed NGNP, with appropriate adjustments for the different fuel element form of the pebble bed reactor.

A few of the FSV initial core fuel elements (up to 32) and replaceable reflector elements (up to 19) were extensively characterized prior to loading into the reactor core, including a detailed characterization of the fuel rods, burnable poison rods, and the graphite blocks. In addition, these fuel elements included small temperature and fluence monitors in selected fuel compact stacks. The initial fission gas release characteristics were measured on some of the fuel compacts. These fuel and reflector elements were placed in preselected core locations that had specific temperature and fluence conditions of interest.

The original FSV Fuel Surveillance Program (prior to the introduction of changes in fuel element graphite beginning with the third refueling) consisted of the following surveillance items planned for each specific refueling as shown in Table 1. (This information is a summary of the fuel surveillance information in the FSV Final Safety Analysis Report.) Note that FSV fuel operated on a six year fuel cycle, with one sixth of the core removed and replaced during each refueling.

- Obtain a photographic record of all six vertical faces of at least 90% of the spent fuel elements permanently removed from the core during refueling using the Fuel Handling Machine 35mm camera or the Cask Video Monitor.
- Evaluate in a timely manner all photographic records for indications of significant abnormalities which could have an effect on the structural integrity of the elements. Significant abnormalities are any unanticipated characteristics, the origin of which is not readily explainable, which differ in nature from those observed in fuel element inspections conducted previously. Elements exhibiting stains, scratches, abrasions, minor cracks and gouges typical of prior segment inspection results would not be considered abnormal in nature.
- Inform the NRC within 72 hours of any significant abnormalities identified which could have an effect on the structural integrity of a fuel element.
- At the time of refueling, withdraw five elements from the reactor and examine. This examination will include:
    - visual examination
    - measurements to determine graphite dimensional changes.

  Data evaluation and documentation of the PIE results will be provided to the NRC as it becomes available. A report of the examination results is to be submitted to the NRC within 12 months after withdrawal of the elements from the reactor.

If the design basis primary coolant activity limits as established by Technical Specification LCO 4.2.8 had been exceeded, a sufficient number of PIEs (commencing with the next refueling) were required to be performed to determine the cause of the excess activity levels. (FSV Technical Specification LCO 4.2.8 is discussed in the response to RAI FQ-3MST-28.)

A total of three refuelings were conducted at FSV prior to the decision by the owner/operator to end facility operation, so none of these required surveillances were actually conducted. However, in addition to the required fuel surveillances discussed here, destructive PIE was performed on fuel elements removed from the core during the second and third refuelings. Two irradiated fuel compacts removed from the reactor (one during the second and one during the third refueling) were sectioned in a hot cell and examined using metallographic techniques to confirm predicted fuel performance. In addition, 54 fuel and replaceable reflector elements removed during the second refueling and 62 fuel and replaceable reflector elements removed during the third refueling were nondestructively examined for evidence of graphite oxidation or other damage. Additional fuel elements (approximately 175) removed during the third refueling underwent photographic examination.

At the time that the FSV Fuel Surveillance Program was developed, post-irradiation safety heatup testing methods had not yet been developed.

Again, this response provides brief conceptual summaries of the test programs currently envisioned for a modular HTGR demonstration plant. However, specific commitments at this time would be premature. These will be addressed at the appropriate time by the license applicant.

Table 1. Fort St. Vrain Fuel Surveillance Program

Refueling No.

Item No.	4	5	6	7	8	9	10 and after
1	✓	✓	✓	✓	✓	✓	✓
2	✓	✓	✓	✓	✓	✓	✓
3	✓	✓	✓	✓	✓	✓	✓
4	✓	✓	✓			✓	

RAI FQ-27/MST-32: For pebble fuel, discuss whether/how particle defect/failure rate statistics (i.e., 50% confidence, 95% confidence) are related to the calculation of the MSTs for anticipated operational occurrences (AOOs), DBEs (and DBAs) and beyond design basis event (BDBEs) (and emergency preparedness). Discuss the statistic that is used to develop these confidence values.

Response FQ-27/MST-32:

The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977, "Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011.

RAI FQ-28/MST-33: For pebble fuel, will the 95% confidence or the design failure fraction values be used for all MST predictions for all the event categories? If so, discuss the basis for developing the NGNP design failure fraction values from the NGNP 95% confidence failure fraction values shown in Table 13 and the NGNP design failure fraction curve shown in Figure 22 of the white paper.

Comments: Section 3.2 of the white paper on Mechanistic Source Terms states the following (italics text for emphasis):

"Source terms for compliance should be 95% confidence level values based on best-estimate calculations.

Source terms for emergency preparedness should be mean values based on best-estimate calculations."

Section 1.4 of the white paper on LBE Selection states the following (italics text for emphasis):

"Acceptable limits on the event sequence consequences and the analysis basis for the LBE categories are as follows:

- AOOs - 10CFRPart 20: 100 mrem total effective dose equivalent (TEDE) mechanistically modeled and realistically calculated at the exclusion area boundary (EAB). For the NGNP facility, the EAB is expected to be the same area as the controlled area boundary.
- DBEs - 10CFR§50.34: 25 rem TEDE mechanistically modeled and conservatively calculated at the EAB.

- BDBEs - NRC Safety Goal quantitative health objectives (QHOs) mechanistically and realistically calculated at 1 mile (1.6 km) and 10 miles (16 km) from the plant."

Response FQ-28/MST-33:

The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977, "Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011.

RAI FQ-29/MST-34: For prismatic fuel, discuss whether/how the particle defect/failure rate statistics (i.e., 50% probability, 95% probability) are related to the MST calculation bases and LBE consequence limits for AOOs, DBEs (and DBAs) and BDBEs (and emergency preparedness) cited in Section 3.2 of the MST white paper and Section 1.4 of the LBE white paper? Discuss the statistic used to develop these probabilities. (Section 4.3.3) {SR/F9; MST-34} [2-ii], (Section 3) {SR/F9; FQ-29} [2, 3]

Response FQ-29/MST-34:

As discussed in the response to RAI MST-88, Monte Carlo methods are used to determine the overall effect of uncertainties on the source terms and off-site consequences. Typically the range of each uncertainty and the sensitivity of each parameter of interest (including particle defect/failure rates) to the uncertainty are provided to a statistical evaluation package. The analysis results are taken directly at the desired confidence level for each parameter of interest. The response to RAI MST-88 includes information on the uncertainty distributions for various parameters of interest.

The resulting consequence distributions provide a basis for judging acceptability and safety margins for a range of requirements. These requirements include those for which an upper bound evaluation is appropriate to demonstrate safety margin, such as for Design Basis Accidents (DBAs) in Chapter 15 of a safety analysis report relative to the regulations of 10CFR50.34. The requirements also include those for which expected, best estimate safety evaluations are appropriate, such as for assessing normal operation and Anticipated Operational Occurrences (AOOs) to determine off-site dose, and assessing Design Basis Events (DBEs) and Beyond Design Basis Events (BDBEs) relative to the emergency planning Protective Action Guides.

This approach to the uncertainty assessment allows calculation of source terms and offsite dose to be conducted at any desired confidence level (e.g., mean or upper 95% confidence) for each Licensing Basis Event (LBE) – AOOs, DBEs, or BDBEs.

The particle defect/failure rate statistics are related to the mechanistic source terms calculation and LBE consequence limits in two ways, one for normal operation and the other for off-normal events. The fuel performance during off-normal events is strongly influenced by its performance during normal operation. The response to RAI FQ-30/MST-35 discusses this further for off normal events.

The statistics for fuel performance during normal operation influence the "expected" (mean) and 95% upper bound "design" radionuclide releases from the fuel, which in turn affect the circulating and plateout activity around the primary circuit. In the deterministic consequence analyses for off-normal events the upper bound "design" releases are the assumed initial inventories outside the fuel. The "design" activities are used whenever conservative deterministic analyses are required. The "expected" activities are used whenever best estimates are appropriate. "Expected" and "design" radionuclide design criteria are also discussed in the response to RAI MST-87.

RAI FQ-30/MST-35: For prismatic fuel, discuss how the 95% probability failure fraction values will be used for all MST predictions for the event categories

Response FQ-30/MST-35:

For the fuel performance during off-normal events, 50% and 95% probability failure fractions are used to construct log-normal probability distributions that are used in the uncertainty analysis of mechanistic source terms and off-site consequences for the event categories.

Prior to the uncertainty analyses, a detailed full core analysis of the fuel performance during off-normal events is performed. The off-normal fuel performance depends on its initial condition as a result of the manufacturing process and of the reactor normal operation. Thus, the fuel particles are grouped into those that have intact coatings, a defective SiC coating, and an exposed kernel (defective outer PyC, SiC, and inner PyC coatings).

During off-normal transients, such as pressurized or depressurized conduction cooldown when the core heats up due to loss of forced cooling, the fuel performance is impacted in two primary ways due to the time at higher than normal operation temperatures. The particles with failed or degraded coatings release volatile radionuclides at an increased rate and, depending on the transient temperatures, the fraction of failed or degraded particles may increase, which also leads to volatile radionuclide release. The detailed radionuclide release analyses are performed with time-at-temperature input as a function of location.

As discussed in the response to RAI MST-88, Monte Carlo methods are used to determine the overall effect of uncertainties on the source terms and off-site consequences. Typically the range of each uncertainty and the sensitivity of each parameter, such as the fuel failure fractions and radionuclide releases, to the uncertainty are provided to a statistical evaluation package. The analysis results are taken directly at the desired confidence level for each parameter of interest.

The resulting consequence distributions provide a basis for judging acceptability and safety margins for a range of requirements. These requirements include those for which an upper bound evaluation is appropriate to demonstrate safety margin, such as for Design Basis Accidents (DBAs) in Chapter 15 of a safety analysis report relative to the regulations of 10CFR50.34. The requirements also include those for which expected, best estimate safety evaluations are appropriate, such as for assessing normal operation and Anticipated Operational Occurrences (AOOs) to determine off-site dose, and assessing Design Basis Events (DBEs) and Beyond Design Basis Events (BDBEs) relative to the emergency planning Protective Action Guides.

This approach to the uncertainty assessment allows calculation of source terms and offsite dose to be conducted at any desired confidence level (e.g., mean or upper 95% confidence) for each Licensing Basis Event (LBE) – AOOs, DBEs, or BDBEs.

RAI FQ-31/MST-36: In FQ Section 4.3, Prismatic Fuel, no figure similar to Figure 22 (for pebble fuel) is provided. For the prismatic approach, describe the source of the data to be used for developing the codes for conservatively predicting particle failure fractions versus fuel operating conditions (e.g., temperature, burn-up, or time at temperature) to be used in the NGNP safety analysis. Describe the source of the data to be used for validating the codes. Discuss the statistical probability levels (e.g., 50%, 95%) applicable to the failure fractions predicted by the codes used in the prismatic analytical approach.

Response FQ-31/MST-36:

The international data base for the performance of TRISO-coated fuel particles under core heatup conditions demonstrates that the functional failure of the coating system is controlled by time-at-temperature phenomena [e.g., TECDOC 1997]. Consequently, a time-independent failure model such as that given in Figure 22 of the Fuel Qualification White Paper for pebble fuel is very conservative for predicting the full-core behavior of modular HTGRs designed to limit peak fuel temperatures to <1600°C during depressurized core conduction cooldown accidents. As a more realistic, yet still conservative, alternative, the current accident condition model used for prismatic core safety analysis (by General Atomics) is the so-called "1989 Goodin-Nabielek" model [1, 2]. These references describe the experimental data bases upon which the model is based, including an uncertainty analysis of the data bases. This uncertainty analysis supports the objectives of calculating the radionuclide source terms to within a factor of 4 for gaseous radionuclides and a factor of ten for metallic radionuclides, which are discussed in the response to RAI MST-62. This model is based primarily upon German LEU UO_2 TRISO particle data. As post irradiation heating data for the reference LEU UCO TRISO fuel become available from the on-going NGNP/AGR fuel development and qualification program, the 1989 Goodin-Nabielek model will be updated as appropriate.

The gradual degradation of the capability of TRISO particles to retain fission gases and volatile fission metals over time at high temperatures ("functional failure" of the coating system, as discussed in the response to RAI FQ-24/MST-29) is clearly evident from the German heating data for LEU UO_2 TRISO fuel that are summarized in Section A-1.3 of the Fuel Qualification White Paper. Note that irradiated TRISO particles can be heated for >300 hours at 1600°C with no functional failure. (The diffusive release of Ag-110m from intact TRISO particles is not classified as functional failure because silver is released at high temperatures from intact TRISO particles meeting all as-manufactured product specifications.) Limited functional failure is observed at 1700°C, and the failure rate increases as the temperatures further increase, but the failure rate is not rapid even at 2000°C. Modular HTGRs are typically designed to limit peak fuel temperatures during depressurized core conduction cooldown accidents to approximately 1600°C to avoid conditions that could produce rapid or excessive coating failure.

The 1989 Goodin-Nabielek model is based primarily upon post irradiation heating data for high-quality German LEU UO_2 fuel particles. As discussed in Fuel Qualification White Paper Section 3.3.3, post irradiation heating tests have also been performed with LEU UCO TRISO particles [2], but the UCO test fuel generally had as-manufactured defects and/or had experienced failure rates during accelerated irradiation testing prior to heating that exceeded the fuel quality and performance requirements for modern modular HTGRs. Consequently, these early UCO data were judged to be of limited value for deriving failure models for high quality HTGR fuel. Nevertheless, these UCO data are still valuable in that they indicate that the dominant, high-temperature failure mechanisms of fission product/SiC corrosion and SiC thermal decomposition are common to both UO_2 and UCO TRISO particles. Moreover, UCO particles have the advantage that CO formation is suppressed, thereby limiting the potential for pressure-vessel or CO corrosion of the SiC coating, as discussed in Section 3.1.2.2 of the Fuel Qualification White Paper.

As complete sets of post irradiation heating data for the reference UCO TRISO fuel become available from the NGNP/AGR program, the accident condition model will be updated as appropriate. The updated model will be based primarily upon data from the AGR-1, AGR-2 and AGR-5/6 tests and will be independently validated with data from the AGR-7 test.

References:

1. Martin, R.C., "Compilation of Fuel Performance and Fission Product Transport Models and Database for MHTGR Design," ORNL/NPR-91/6, Oak Ridge National Laboratory, October 1993.

2. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," TECDOC-978, International Atomic Energy Agency, November 1997.

RAI FQ-32/MST-37: Compare the conservatism of the pebble-bed empirical approach to the conservatism of prismatic-block analytical approach with respect to the predicted design failure fractions. Compare the statistical statements associated with the predicted particle failure fractions for each fuel type.

Response FQ-32/MST-37:

As discussed in the response to RAI FQ-31/MST-36, the international database for the performance of TRISO-coated fuel particles under core heatup conditions demonstrates that the functional failure of the coating system is controlled by time-at-temperature phenomena. Consequently, a time-independent failure model such as that proposed for pebble-bed fuel is very conservative for predicting the full-core fuel behavior of modular HTGRs. As a more realistic, yet still conservative, alternative, the current accident condition model used for prismatic core safety analysis (by General Atomics) is the so-called "1989 Goodin-Nabielek" model.

The relative degree of conservatism cannot be fully quantified without a full-core analysis of a common core design for a common accident scenario with both the pebble bed failure model (Figure 22 in the Fuel Qualification White Paper) and the 1989 Goodin-Nabielek model used for prismatic modular HTGRs. However, the degree of conservatism with the pebble bed time-independent failure model is anticipated to be significantly larger. For example, the German heating data for LEU UO_2 TRISO fuel that are summarized in Section A-1.3 of the Fuel Qualification White Paper demonstrate that irradiated TRISO particles can be heated for longer than 300 hours at 1600°C with no functional failure. In contrast to the test data, the pebble bed failure model predicts a failure fraction of 2×10^{-4} at 1600°C that is obtained relatively quickly (the "PBMR Expected" failure curve in Figure 22).

With regard to the "statistical statements" associated with the predicted particle failure fractions for each fuel type, the model used for prismatic modular HTGRs, because it addresses more coated particle fuel failure modes, lends itself to a more detailed evaluation of the confidence level associated with calculated fuel particle performance that is based on the level of uncertainty associated with each potential fuel failure mode. However, in general both approaches produce quantitative differences between best estimate and 95% confidence fuel failure fractions that are of about the same order of magnitude. As noted in the response to RAI FQ-31/MST-36, as complete sets of post irradiation heating data for the reference UCO TRISO fuel become available from the NGNP/AGR program, the Goodin-Nabielek accident condition model will be updated as appropriate to provide further support for the model and its associated statistical confidence levels.

RAI FQ-33: Discuss the worst case DBA and worst case BDBA reactivity insertion events expected for the pebble-bed and prismatic-block NGNP plant designs and the associated maximum local kernel energy depositions.

Comments: IAEA-TECDOC-978 Section 4.3 describes the Japanese and Russian reactivity initiated accident testing and associated failure fraction results. Test conditions are described in terms of kernel

energy deposition (J/g UO_2) rather than kernel or particle fuel temperature. The results indicate that the failure fraction can become significant (i.e., $>10^{-5}$) when kernel energy deposition reaches about 600 J/g UO_2. At about 1000 J/g UO_2 the failure fraction can reach 0.1.

Response FQ-33:

In summary, transient kernel energy deposition is not a useful metric for assessing HTGR reactivity insertion events for reasons discussed below. As noted in the NRC comments for the RAI, power pulse tests, which typically are conducted in millisecond time frames, and slower adiabatic heating tests, with both types of testing approaching and exceeding fuel melting temperatures,[30] have produced particle failures. However, HTGR reactivity insertion events typically take place in a time frame of minutes, and, as discussed below, coated fuel particle thermal time constants are a small fraction of a second. Thus, the Japanese and Russian tests are not representative of HTGR reactivity insertion events. Fuel temperature history (time at temperature) is the most direct indicator of challenges to fuel performance. Reactivity events do not produce fuel temperature histories that approach the severity of the depressurized loss of forced convection events with regard to presenting a challenge to fuel performance.

Energy Deposition as a Metric for Assessing Reactivity Insertion Accidents

The use of integrated energy deposition in the heavy metal as a metric for evaluating fuel performance is only appropriate if the time scale of the energy deposition is short compared to the time constants for heat transfer in the core. The time constant for heat transfer from a fuel particle to the surrounding matrix, defined as the product of the kernel mass, the kernel heat capacity, and the thermal resistance of the kernel and coating layers, is particularly relevant. The German UO_2 particle, with a 500 μm kernel diameter, has a time constant of approximately 40 milliseconds, which is slightly longer than the time constant of a UCO particle with a smaller 425 μm kernel. The effect of the thermal time constant and time frame for energy deposition is discussed below.

- If the energy deposition time is short compared to the thermal time constant of the fuel particle, the deposited energy is absorbed primarily by increasing the fuel kernel temperature, and a relatively small fraction of the energy is transferred from the kernel to its surroundings. In the limit, the fuel kernel heatup is adiabatic, with all the energy going into raising the kernel temperature. In this case, 600 J/g UO_2 would cause an increase in the kernel temperature of approximately 1900°C, and 1000 J/g UO_2 would cause an increase of approximately 3100°C. The UO_2 melting point is approximately 2840°C with a volumetric expansion of approximately 22% on melting. Given the shock loading on a particle when sufficient energy to melt the kernel is deposited in a few milliseconds, particle failure is to be expected when the kernel temperature rapidly approaches the melting point.

- If the energy deposition time is long compared to the thermal time constant of the fuel particle, most of the energy is transferred out of the fuel particles, and fuel temperature is much less affected. In the limit of steady-state operation, the fuel kernel temperature remains constant as all of the energy deposited is transferred out of the kernel through the coatings, matrix and graphite to the coolant. For illustration, in normal operation of the MHTGR design with a core power of 350 MWt and a UCO+ThO_2 loading of around 3800 kg, the core average heavy metal energy deposition rate is about 90 J/second/g UCO+ThO_2. If integrated energy deposition were the sole metric, and the failure threshold were 600 J/g UCO+ThO_2, the onset of core wide fuel particle failures could be expected in less than 7 seconds of normal operation. Energy deposition has no relation to particle failure in this

[30] Under steady state or slow transient conditions coated particle fuel does not "melt." The UO_2 or UCO kernels are carbothermically reduced to carbides, which go into solid solution in a sea of carbon.

case, so it is essential to consider the design specific transient characteristics of core heat transfer, excess power, and fuel temperatures associated with reactivity insertion events.

To illustrate the importance of energy deposition time, a simple thermal model of a coated fuel particle suspended in an isothermal medium was constructed and subjected to energy deposition at constant power over varying intervals, with power level adjusted as necessary to maintain a constant total energy deposition for each interval. The results are shown in the Figure 1 for total energy depositions of 600, 1,000 and 1,400 J/g fuel. As shown, if the deposition duration is less than the coated particle thermal time constant, as was the case for the Japanese and Russian pulse testing, the increase in fuel temperature, becomes very large. On the other hand, if the deposition duration is on the order of one second or more, the increase in fuel temperature is very small.

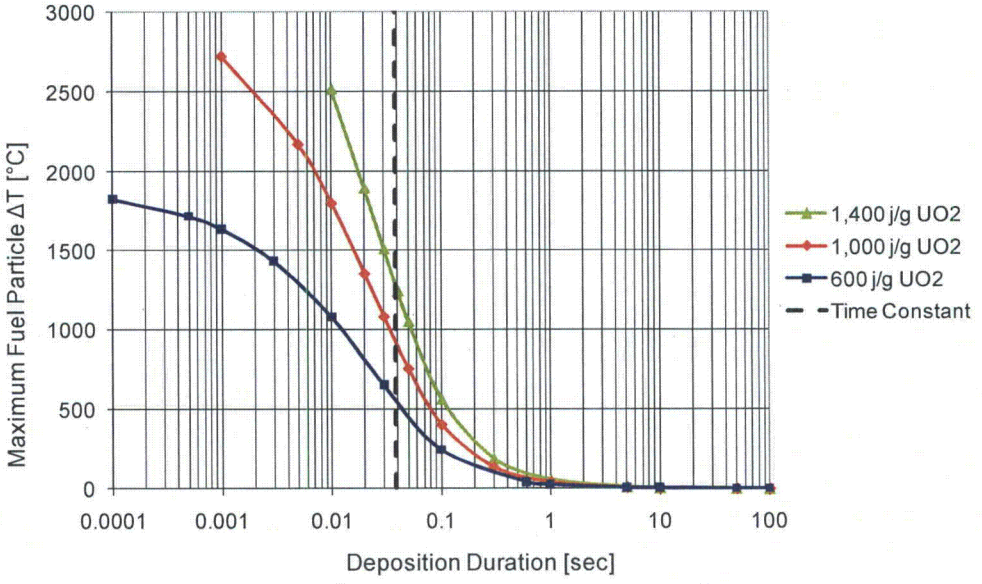

Figure 1. Maximum Fuel Particle ΔT vs.
Energy Deposition Duration Constant Power Over Deposition Interval

Deposited energy can be and historically has been an effective figure of merit for assessing limiting reactivity insertion events for LWRs, in particular the limiting control rod ejection events. However, it is not suitable for HTGRs. The following discussion addresses in general terms the relevant differences between LWRs and modular HTGRs that in combination explain why deposited energy is not a useful metric for HTGRs and why reactivity insertion events do not present a limiting challenge to HTGR fuel integrity.

- Fuel Thermal Time Constant. As discussed above, the transient energy deposition period must be short with respect to the fuel thermal time constant so that the energy is directed primarily toward increasing the fuel temperature. The thermal time constant is defined as the heat capacity of the fuel divided by the heat removal characteristics for a unit fuel segment. In an LWR the unit fuel segment is a long cylindrical stack of UO_2 pellets encased in a metal cladding, approximately 10 mm in diameter, with a time constant on the order of 7 seconds. In an HTGR the unit fuel segment is a

spherical UO_2 or UCO kernel surrounded by multiple layers of pyrocarbon and silicon carbide approximately 1 mm in diameter, with a time constant on the order of 40 milliseconds.

- Reactivity Insertion Time. In an LWR the control and safety rods are rigid and the rod drive housing on the top (or bottom in the case of a BWR) is typically not captured by the refueling floor within a distance that is small compared to the core height. In the case in the MHTGR and expected NGNP design, the control rods are comprised of a flexible string of canisters on a cable with less capability to absorb the high pressure driving force of a housing failure, and the upper shield plug will capture an ejected control string in a short distance relative to the height of the core. Also, the MHTGR core is taller (26 feet/7.93 m for prismatic design) than an LWR core (14 feet/4.27 m). Thus the high pressure driving force if the housing fails can rapidly remove the rod from the core of an LWR, but not the NGNP. In a fast ramp reactivity insertion the peak power is approximately proportional to ramp rate.

- Overpower Capability. The higher power density and larger size of an LWR fuel pin in comparison to a coated particle translates to a much higher maximum local temperature rise from the coolant to the fuel under steady-state full power conditions, about 2000°C for an LWR compared with about 50°C for an HTGR, and the maximum fuel temperature is nominally more than 1000°C higher for an LWR. As a result, assuming no change in temperature at the fuel/coolant interface, an LWR would approach centerline fuel melting at about 120% of rated power compared with about 2000% of rated power for an HTGR (in fact the coolant system could not support this power level, but the ratio is applicable in the short term regarding approach to fuel melting under transient overpower conditions).

- Neutron Migration Length. The neutron migration length determines the area over which the power increase is distributed and the corresponding degree of power peaking associated with a one dimensional localized reactivity insertion such as a control rod removal. A typical migration length for an LWR is about 8 cm, compared with about 60 cm for an HTGR. Thus, the relative area of the peak is about 50 times larger for an HTGR, and the specific power in the peak is correspondingly substantially lower.

Coated Particle Fuel Power Transient Testing

Pulse test reactors are designed to deposit large amounts of energy in test specimens in a very short time (e.g., a few milliseconds). The coated particle transient testing noted in the RAI comments and reported in IAEA-TECDOC-978 are summarized as follows:

- Pulse Testing in Japan - The tests conducted in the NSRR in Japan were of extremely short duration, in which energy deposition ranging from 200 to 2,300 J/g UO_2 occurred with a full width at half maximum power ranging from 10 to 30 milliseconds. Such an excursion in a reactor would require a super-prompt-critical net reactivity, with reactivity insertion amounts and time frames sufficient to override reactivity feedbacks, conditions that are not credible for either NGNP design.

- Power Transient Testing in Russia. The particle power pulse tests conducted in the HYDRA reactor were of even shorter duration, with energy deposition ranging from 100 to 1,700 J/g UO_2, and pulse durations of 1 to 2 milliseconds. Longer duration (0.7 to 30 second) high power transients were conducted on fuel spheres with no provision for heat removal. In these tests the calculated fuel kernel temperature continued to increase with time, and in the longer duration tests the nominal fuel sphere center temperature approached or exceeded the melting point of UO_2 (approximately 3100°K). These conditions are also not credible for either the prismatic or pebble bed NGNP design.

Particle failures were observed in both Russia and Japan. However, the conditions are sufficiently far removed from conditions that could occur in either the prismatic or pebble bed NGNP that these cannot be considered as simulated reactivity insertion accident tests. It would be more appropriate to consider them as an academic exercise to explore the transient temperature response of coated particle fuel without consideration of what is possible within modular HTGR core. In general, particle failures were observed as the fuel kernel temperatures approached and exceeded the melting point, as would be expected.

As discussed in the response to RAI FQ-41/MST-45, the HTGR coated particle fuel designs can be expected to be tolerant of power and temperature transients. This has been generally confirmed by experience with rapid cooldowns associated with irradiations in materials test reactors and the transients experienced during initial loading and recycling of fuel spheres in the AVR reactor, and in post-irradiation heating tests. The coolant flow was upward in the AVR, so the fuel spheres, initially at approximately room temperature, were dropped onto the core at hot core exit conditions with local coolant temperatures as high as 1200°C. The spheres also experienced a sudden increase in power, going from zero power to the power level associated with the thermal flux at the top of the reactor. No detrimental effects associated with these repeated severe power and temperature transients have been observed in large numbers of spheres irradiated in the reactor and subjected to detailed post-irradiation examination and/or accident simulation testing.

Reactivity Insertion Events for NGNP Design

Since the NGNP designs have not progressed to the point of having detailed design information and associated safety analyses, illustrative results are based on the MHTGR. Given the unsuitability of energy deposition as a metric in assessing reactivity insertion events for HTGRs, the summary of limiting reactivity insertion events is directed primarily to fuel temperatures, which are directly related to fuel performance. The following information is taken from the MHTGR Preliminary Safety Information Document (PSID)[1] and PRA [2]:

- Limiting DBA The worst case DBA reactivity insertion event is a control rod group withdrawal without the main heat transport system and shutdown cooling system, DBE-4 as reported in Section 15.5 of the PSID. The conservatively calculated peak power and peak fuel temperature are reported as 515 MWt (147%) and 1394°C (a 170°C increase from the initial conditions), both reached about 100 seconds after the start of the event.

- Limiting BDBE – in the MHTGR design, the control rod drives are located immediately underneath the concrete refueling floor, and the top plug arrangement precludes a rod ejection event. Since the rod drives are captured under the refueling floor, there is no credible mechanism to produce a rod ejection event even at the low frequencies considered for BDBEs, so a rod ejection should not be considered as a BDBE. Nonetheless, in the MHTGR PRA a hypothetical rod ejection event was analyzed as reported in Section G.1.3 to demonstrate the reactivity characteristics for a rapid positive reactivity addition. The best estimate calculated peak power and peak fuel temperature are reported as 2210 MWt (630%) and 1275°C, both reached about 10 seconds after the start of the event, with an integrated energy release of 11 MW-h out to 50 seconds (integrated energy for 50 seconds at rated power is about 5 MW-h). The core average energy deposition corresponding to 11 MW-h would be about 10,000 J/g heavy metal ($UCO+ThO_2$), but for reasons discussed earlier, this is not relevant to fuel integrity.

References:

1. DOE-HTGR-86024, Preliminary Safety Information Document for the Standard MHTGR.
2. DOE-HTGR-86011, Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor.

RAI FQ-34/MST-38: Describe the air ingress testing plans for pebble fuel and block fuel.

Comments: As described and shown in IAEA-TECDOC-978 Section 5.4, air ingress has the potential to significantly increase the particle failure fraction above that associated with a LOFC or DLOFC accident due to the effects of oxidation of the particle coating layers. Air ingress testing plans have not been provided for pebble or block fuel.

Response FQ-34/MST-38:

The approach to addressing air ingress in the AGR Program is documented in Reference [1] and summarized below.

Ingress of helium mixed with air (expected to be only a few percent air, see response to RAI MST-74) into the core of a HTGR can possibly occur following a depressurization accident. The amount of air ingress depends on break size, break location, break orientation, and design of the reactor cavity and the reactor building, all of which influence the ability of air to enter the core via natural circulation, stratified flow, or molecular diffusion. In all cases, the reactor has been shutdown, so all fission product releases associated with oxidation will occur under shutdown conditions, as opposed to operating conditions. In both a prismatic and a pebble bed HTGR, a graphite thickness of about 5 mm must be permeated before the helium-air mixture or reaction products-helium-air mixture can have access to a fuel compact (prismatic design) or fuelled region of a spherical fuel element (pebble bed design). Some specific tasks are discussed, below and others may be identified, subject to specific design issues and the approach taken by reactor designers to mitigate the accident.

To date, computational fluid dynamic (CFD) simulations of natural circulation through a reactor core of prismatic design, a stratified flow experiment for CFD validation, and an investigation of concepts to mitigate air ingress have been conducted [2].

It is planned to conduct safety tests (fission product release at elevated temperatures) on compacts irradiated in graphite sleeves (to simulate the approximately 5-mm thick web) or irradiated spherical fuel elements at various partial pressures of oxygen over a temperature range to be determined. It is also planned to study the air/SiC interaction by experimentally mapping the transition from the formation of protective SiO_2 to the formation of volatile SiO as a function of temperature and oxygen partial pressure to confirm thermodynamic analyses.

References:

1. "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," PLN-3636, Idaho National Laboratory, September 30, 2010.
2. E. S. Kim and C. Oh, "Air Ingress Experiment and Mitigation Method Development," VHTR R&D FY-11 Technical Review Meeting, April 26-28, 2011, Albuquerque, NM.

RAI FQ-35/MST-39: Discuss whether the matrix-graphite abrasion test, or any other test, will be performed to verify the assumed in-reactor fuel pebble matrix dust generation rate (e.g., g/core pass) over the fuel lifetime.

Response FQ-35/MST-39:

The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977 "Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011.

RAI FQ-36/MST-40: Discuss whether and how the irradiation, PIE, and safety testing will provide any separate effects data needed to demonstrate and confirm that the rates of fission product transport through the individual fuel particle coatings and through the matrix material are consistent with those of the reference German fuel particles and matrix material. If confirmatory data will not be collected, discuss the technical basis for assuming that the fission product transport (e.g., diffusion) rates associated with the NGNP fuel will be the same as the German fuel.

Comments: The data to be collected from irradiation, PIE, and safety testing appear to be focused on coated particle failures and integral effects suitable for the validation of analytical models and methods to be used for predicting integral core-wide fission product releases from fuel pebbles.

Response FQ-36/MST-40:

An objective of the AGR Program is to measure fission product transport through the SiC layer, the primary barrier to fission product release from the fuel particle, as well as through the matrix material surrounding the fuel particles. It will be of interest to compare the results of these measurements with the historical database, which includes results from the German program. Details of the measurement of fission product transport through the SiC layer are found in responses to RAIs FQ-2/MST-2, FQ-13/MST-18, FQ-14/MST-19, and FQ-16/MST-21.

The AGR-3/4 experiment, and its associated PIE and safety testing, is designed to measure fission product transport through fuel matrix and fuel element graphite. In this experiment, a known source is provided by designed-to-fail particles in the fuel compact. Fission product transport through concentric rings of matrix and graphite surrounding the fuel compact will be evaluated by measuring concentration gradients of fission products through the rings in PIE. Fission product transport through graphite under heatup accident conditions will be evaluated in safety tests of fuel bodies (specially designed as part of the AGR-3/4 experiment, in which the fuel compact and concentric rings of matrix and graphite are totally surrounded by graphite) by measuring fission product releases as a function of time at temperature.

Separate effects testing using SiC or PyC samples with simulated fission products are not planned because of the non-prototypicality of such tests (e.g., microstructure of samples not being the same as in TRISO fuel, chemical activity of fission products not typical of TRISO fuel).

RAI FQ-37/MST-41: Since the German UO_2 test data for TRISO-coated particle diffusion rates are largely based on post-irradiation heating tests, discuss how the additional testing for pebble bed reactor fuel will evaluate fission product transport under high temperature irradiation.

Comments: Note that in TEV-1022 INL states: "To accurately model fission product transport in TRISO-coated particle fuel under high temperature irradiation, use of 'effective' diffusion coefficients for

the kernel and coatings (as presented in the IAEA-TECDOC-978) obtained from post-irradiation heating tests is not recommended because those coefficients do not consider the irradiation effects, either implicitly or explicitly."

Response FQ-37/MST-41:

RAI FQ-37/MST-41 is identical to RAI FQ-21/MST-27. Please see response to RAI FQ-21/MST-27.

RAI FQ-38/MST-42: For pebble fuel, discuss whether irradiation tests will involve fuel temperature transient conditions which simulate the cyclical fuel temperature changes (shown in Figure 16) that will occur over time during power operation due to the re-circulation passage of fuel pebbles through the core.

Response FQ-38/MST-42:

The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977 "Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011.

RAI FQ-39/MST-43: Discuss the technical basis that was used for developing the "design failure fraction" values in Table 13 and Figure 22 for the HTR-Modul. Discuss the technical basis that will be used for developing the design failure fraction values in Table 13 and Figure 22 for the NGNP PBMR fuel. Given the significant reliance of the predicted pebble-bed NGNP fuel failure probabilities on the historical German data, discuss the basis for projecting that the pebble-bed NGNP design failure fractions will be less than the HTR-Modul fuel design failure fractions as shown in the table and figure

Comments: NGNP pebble-bed fuel performance, including particle failure fraction prediction, will be based to a significant degree on historical German LEU UO_2 TRISO fuel irradiation (and accident heat-up test) failure data. Some of the fuel in these tests may not be exactly of the same design and manufacture as the pebble-bed NGNP fuel. Additionally, some of the irradiation test conditions may not be as bounding as the conditions projected for the NGNP PBMR fuel. These data will be supplemented and confirmed by additional test data to be developed from fuel irradiations (and accident heat-up tests) of fuel manufactured for a pebble-bed NGNP.

Response FQ-39/MST-43:

The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977 "Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011.

RAI FQ-40/MST-44: The German LEU UO_2 TRISO fuel irradiation data was developed using quality assurance procedures that may not be consistent with the NRC regulations in 10CFR50 Appendix B. Discuss the compensatory measures, if any, will applied in using the German data to offset any material deficiencies in the conduct of the German tests and the analysis of the German test data.

Response FQ-40/MST-44:

The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in Idaho National Laboratory (INL) letter CCN 223977

"Contract No. DE-AC07-051 0 14517 - Next Generation Nuclear Plant Project Submittal - NRC Project # 0748 - Supplemental Information to Next Generation Nuclear Plant Project Fuel Qualification (FQ) and Mechanistic Source Terms White Papers," May 3, 2011.

RAI FQ-41/MST-45: In the discussion concerning particle effects, the effect of temperature change on the particles is not listed. Discuss the effects of temperature changes from fabrication, room temperature, normal operation, and abnormal operation.

Response FQ-41/MST-45:

The TRISO-coated fuel particles that are used in both prismatic and pebble-bed HTGR cores are ceramic materials that are capable of experiencing a broad range of temperatures without compromising their primary function of retaining the fission products at their source during normal plant operation and postulated accidents to the extent required to meet all top-level radionuclide control requirements adopted by the NGNP Project. Of these requirements and goals, the goal of meeting the EPA PAGs at an approximately400-m EAB is the most constraining on the fuel performance.

High temperature chemical processes are fundamental to the fabrication of the TRISO-coated fuel particles and to the subsequent fabrication of fuel compacts for prismatic cores and fuel spheres for pebble-bed cores that incorporate the TRISO particles in a carbonaceous matrix. As discussed in Section 3 of the Fuel Qualification (FQ) White Paper and in [1], the PyC and SiC coatings of the TRISO particles are applied by chemical vapor deposition in a fluidized bed operating at high temperature (1200 – 1500°C). The coated particles are then mixed with matrix material and formed into fuel compacts or spheres in a mold. The "green" compacts or spheres are finished by heating them to 1800 – 1950°C under vacuum for about 30 minutes to remove impurities and to partially graphitize the matrix. Quality Control measurements are made during each step of the fabrication process, including on the final product, to assure compliance with the fuel product specifications.

In fact, the compacts or spheres containing the TRISO fuel particles experience higher temperatures during this final fabrication heat treatment than during postulated core heatup accidents. This circumstance serves to illustrate that the failure of TRISO-coated fuel particles is a time-at-temperature phenomenon. Unlike with some other nuclear fuels, there is no rapid melting or deleterious phase changes of any fuel (or core) component under bounding modular HTGR accident conditions.

The thermal transients experienced by a modular HTGR core during normal operation (including startup, shutdown, and load-following), AOOs, and postulated core heatup accidents are slow and benign because of the large heat capacity of the massive graphite core. The FSV HTGR experienced a number of scrams and "rapid" shutdowns for a number of operational reasons, including water ingress from the circulator water bearings [2]. Upon return to power, there was never any indication from the measured circulating noble gas activity that these rapid shutdowns had caused any detectable failure of the TRISO fuel particles used in the FSV core. Moreover, the FSV TRISO particles were arguably less robust than modern TRISO particles in that they had a nominal 25-μm SiC coating compared to a thicker 35-μm SiC coating that will be used in future modular HTGRs.

The thermal transients during pressurized and depressurized core conduction cooldown accidents are also slow. As illustrated in Figure 19 of the FQ White Paper, the time to reach the maximum fuel temperature in a 250 MW(t) pebble-bed core is about 24 hours. For a 350 MW(t) prismatic core, the time to reach the maximum fuel temperature is somewhat longer (approximately 40 hours) because of the higher heat capacity of a prismatic core [3]. In all cases, modular HTGRs are designed such that the maximum fuel temperature is confined to a small portion of the core. The German heating data for LEU UO_2 TRISO

fuel that are summarized in Section A-1.3 of the FQ White Paper demonstrate that irradiated TRISO particles can be heated for >300 hours at 1600°C with no functional failure.

In principle, the most rapid thermal transients in a modular HTGR core would occur during postulated reactivity transients. This topic is addressed in the response to RAI FQ-33. In summary, the TRISO-coated particle fuel used in modular HTGRs can be expected to be tolerant of power and temperature transients associated with credible reactivity transients.

A rough estimate of the stress in the layers of a TRISO particle due to thermal expansion coefficient mismatch can be made using the following simple model. Assume the particle cools from the heat treatment temperature of the compact (1800°C) to room temperature (~25°C). The SiC will be in compression because of the cooling. However, because the SiC is much more rigid than the PyC layer, the contraction of the PyC upon cooling will result in a positive tangential stress in the PyC layers. The magnitude of that stress is given by

$$\sigma_t = \frac{E\alpha\Delta T}{1-\nu}$$

where E is the elastic modulus of PyC, α is the thermal expansion coefficient difference between PyC and SiC, □T is the temperature change and □ is poison's ratio for PyC. For properties from [4], for typical values of anisotropy (approximately 1.03), the thermal expansion coefficient for PyC is approximately 5.9E-06 and for SiC is 4.9E-06. The elastic modulus is 30 GPa and Poisson's ratio is 0.24. This yields a tangential stress in the PyC of approximately 70 MPa. This value is much less than the stresses in the PyC layer induced by irradiation dimensional change, which are on the order of 200-250 MPa. Thus, the thermal expansion mismatch is a modest effect at best. Fuel operation under normal and anticipated off-normal conditions result in much smaller changes in temperature and thus much lower thermal stresses.

Based on the above considerations, thermal transients are not expected to be a significant variable in determining TRISO fuel particle performance during normal plant operation and postulated accidents. Failure of particles with as-manufactured defects is expected to be dominant during normal operation, and time-at-temperature phenomena are expected to be dominant during core conduction cooldown accidents. These expectations have generally been confirmed by experience with rapid cooldowns associated with fuel irradiation tests in materials test reactors, which are much more rapid than modular HTGR shutdown transients. In addition, the transients experienced during initial loading and recycling of fuel spheres in the AVR reactor are noteworthy. The coolant flow was upward in the AVR, so the fuel spheres, initially at approximately room temperature, were dropped onto the core at hot core exit conditions with local coolant temperatures as high as 1200°C. The spheres also experienced a sudden increase in power, going from zero power to the power level associated with the thermal flux at the top of the reactor. No detrimental effects associated with these repeated severe power and temperature transients have been observed in large numbers of spheres irradiated in the reactor and subjected to detailed post-irradiation examination and/or accident simulation testing. Likewise, as discussed above, the numerous scrams and rapid shutdowns experienced by the FSV core had no apparent deleterious effects on the TRISO fuel.

References:

1. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," TECDOC-978, International Atomic Energy Agency, November 1997.

2. Copinger, D. A. and D. L. Moses, "Fort Saint Vrain Gas Cooled Reactor Operational Experience," NUREG/CR-6839 (ORNL/TM-2003/223), Oak Ridge National Laboratory, January 2004.

3. "Preliminary Safety Information Document for the Standard MHTGR," HTGR-86024, Rev. 13, Stone & Webster Engineering Corp., September 1992.

4. G. K. Miller, D. A. Petti, J. T. Maki, and D. L. Knudson, PARFUME Theory and Model Basis Report, INL/EXT-08-14497, September 2009.

RAI FQ-42/MST-46: Neither swelling nor creep (either from high temperatures or from irradiation) is specifically mentioned on page 13. Discuss how these phenomena will be considered.

Response FQ-42/MST-46:

The next version of the Fuel Qualification white paper will include mention of these phenomena. Considering the magnitude of these effects, as documented in the CEGA report [1], fuel performance calculations do consider dimensional change (shrinkage and swelling) and irradiation induced creep of the PyC layer due to their significant effect on stresses in the TRISO coatings. However, these calculations do not include SiC layer creep or swelling because they are small enough that they are of secondary importance in the thermomechanical analysis of a TRISO particle. Thus, most calculations assume that the SiC remains rigid under normal and postulated accident conditions.

Reference:

1. CEGA Corporation, "Material Models of Pyrocarbon and Pyrolytic Silicon Carbide", CEGA-002820 Rev. 1, July 1993.

RAI FQ-43/MST-47: Given "the greater emphasis on fuel retention of radionuclides rather than reactor-building retention following an event," and given that meeting the specifications needed to meet required performance capability "requires precise process control," provide a detailed discussion of the process control and characterization procedures that will be applied to fuel fabrication.

Comments: In its white paper, INL states the following: "Particle fabrication specifications established to meet required performance capability include dimensions (mean and variation), densities, pyrocarbon anisotropy, defect levels and selected process conditions. Meeting the specifications requires precise process control and extensive statistically based characterization procedures."

Response FQ-43/MST-47:

The NGNP/AGR Fuel Development and Qualification Program has been reestablishing U.S. fabrication and characterization capability for high quality TRISO-coated particle fuel. The program is currently in the process of scaling up and optimizing process conditions for fuel particle coating and compacting. Final detailed fuel fabrication process control and product characterization procedures will be established once scale-up and process optimization work is complete and prior to the fabrication of the NGNP first core.

The type of information requested in this RAI is more appropriate for a licensing topical report prepared by the fuel supplier. Although the fuel supplier would likely use the types of quality control (QC) tests that have been historically used and which are substantially the basis for those currently being used to characterize fuel for the NGNP/AGR fuel program, these test methods will have to be scaled-up for high-throughput production. The actual process control and statistical product testing/acceptance procedures

will be specific to the fuel manufacturer. However, it is important to note that the use of statistical QC requires the process to produce fuel at a much higher quality level than the specification; otherwise, the fuel manufacturing economics will not be tenable.

Detailed information regarding the development of fabrication and characterization methods for coated particle fuel is provided in Module 7b of the HTGR Technology Course for the Nuclear Regulatory Commission, May 24-27, 2010 [1].

Reference:

1. https://inlportal.inl.gov/portal/server.pt/community/ngnp_public_documents/, NRC Training Presentations

RAI FQ-44/MST-48: Indicate the ranges of particle design parameters (e.g., buffer layer thickness) and service phenomena (e.g., kernel swelling) that would lead to significant coupling between the kernel and dense coating layers.

Comments: This request relates to the following quote from page 14: "The large body of experience noted in this section includes particle designs with a wide variety of kernel properties. However, the kernel of the coated particle is substantially decoupled from the dense pyrocarbon and SiC layers by the low-density-carbon buffer layer. Thus, the experience generally applies to both LEU UO_2 and LEU UCO fuel from the standpoint of dense pyrocarbon and SiC-layer design and performance."

Response FQ-44/MST-48:

The range of design parameters and service conditions that could result in hard contact (coupling) between the kernel and dense coating layers may be illustrated by simple calculation.

As discussed in the response to RAI FQ-45/MST-49, the maximum observed kernel volumetric swelling is ~30%. For nominal operating conditions, a particle with a 425 μm diameter kernel would make hard contact with the coating layers after 30 % FIMA only if the buffer had an initial thickness of 36 μm or less (and the buffer fully densified). Such an out of specification particle, if ever made, would be rejected during the fabrication inspection process. Thus, no coupling between kernel and coating layers would occur under these conditions.

RAI FQ-45/MST-49: A disadvantage of UC is increased radiation-induced swelling. What is the swelling rate of UCO, and what service conditions could further increase the gas evolution of UCO?

Comments: This request relates to the following quote from page 14: "The large body of experience noted in this section includes particle designs with a wide variety of kernel properties. However, the kernel of the coated particle is substantially decoupled from the dense pyrocarbon and SiC layers by the low-density-carbon buffer layer. Thus, the experience generally applies to both LEU UO_2 and LEU UCO fuel from the standpoint of dense pyrocarbon and SiC-layer design and performance."

Response FQ-45/MST-49:

UC_2, UO_2 and UCO coated particle fuels were irradiated under similar conditions, and post-irradiation particle dimensional change measurements were taken in the HRB-15A[1] and HRB-16[2] irradiation experiments conducted in the High Flux Isotope Reactor at Oak Ridge National Laboratory (ORNL). The results of kernel swelling measurements are summarized below:

HRB-15A. Bonded monolayer wafers, each containing 54 particles of a given fuel type were fabricated and irradiated to allow precise measurements of individual particles before and after irradiation. The results are summarized in the table below:

Table 1. Irradiation-induced Dimensional Changes of HRB-15A

Particle Components in Bonded Monolayers

Irradiation Conditions in HRB-15A at ~15A at ~1100°C					Dimensional Change Percentages			
TRISO Particle Type	Particle Batch Number	Monolayer Position	Neutron Fluence (10^{25} n/n^2, E>29fJ)	Burnup (% FIMA)	Kernel	Diameter SiC Shell	Particle	Thickness OPyC
UC_2	6151-23-020	10	6.4	29.0	10.6	0.5	0.2	-1.8
UO_2	6152-04-010	8	6.4	28.8	10.6	0.8	0.3	-4.7
$UC_{0.5}O_{1.5}$	6157-11-010	5	5.8	26.7	10.5	0.6	0.3	-2.4

As shown in the results above, the kernel dimensional change was essentially the same for all three particle types. Metallography results showed a densified layer of buffer material adjacent to the kernel, with a predominance of radial gaps forming between the buffer and IPyC, indicating that kernel swelling was accommodated within the buffer.

HRB-16. Dimensional changes were determined by post-irradiation radiology compared with nominal pre-irradiation kernel diametric, with the following results reported: "In particles with kernels of UCO UC_2 or UO_2, the kernel swelled about 30% by volume (approximately10% diametric swelling) as a result of the irradiation at burnups greater than 20% FIMA. In these cases, the swelling was independent of burnup. In the case of UC_2 particles, for which data at burnups below 20% FIMA exist, the extent of swelling decreased with decreasing burnup, but only apparently rapidly below 10% FIMA; at about 8% FIMA the swelling became negligible."

In summary, UC_2 data indicate that kernel swelling saturates at approximately 30% volumetric (approximately 10% diametric) swelling at a burnup less than 10% FIMA. No significant differences in kernel swelling associated with differences in kernel type or burnup was observed at burnups in the range of 20-29% FIMA and temperatures of approximately 1100°C.

References:

1. GA-A16758, UC-77, "Capsule HRB-15A Postirradiation Examination Report", July 1984.
2. HTGR-85-053, "Capsule HRB-16 Postirradiation Examination Report", September 1985.

RAI FQ-46/MST-50: Expand the discussion in Section 3.1.2 to address other conceivable failure mechanisms beyond those listed on page 19. For instance, can differential thermal expansions and large temperature swings that create stresses lead to particle failure?

Response FQ-46/MST-50:

As stated in the response to RAI FQ-47/MST-51, the list of failure mechanisms on page 19 of the Fuel Qualification White Paper is a compilation of mechanisms that have been observed over the past 50 years of TRISO-coated particle fuel irradiation and accident heating testing around the world. In modern fuels, these potential failure mechanisms are controlled via fuel fabrication specifications and limits on service conditions in the reactor, so in reality all conceivable failure mechanisms are included in the list on page 19. Differential thermal expansions coupled with large temperature swings do not qualify as a conceivable failure mechanism as discussed in the response to RAI FQ-41/MST-45.

RAI FQ-47/MST-51: Please elaborate on how the list of failure mechanisms is constructed and refined. Are these what the authors consider are the "credible" failure mechanisms, or are these failure mechanisms that have been that have been observed in the past? Describe the basis and reasoning to conclude that this list is comprehensive.

Response FQ-47/MST-51:

The list of failure mechanisms described in the white paper is a compilation of mechanisms that have been observed over the past 50 years of TRISO-coated particle fuel irradiation and accident heating testing around the world [1, 2]. Today, these potential failure mechanisms are controlled via fuel fabrication specification limits (e.g., limits on missing or thin buffer, limits on anisotropy of PyC to prevent cracking) or limits on service conditions in the reactor (e.g., time average peak fuel temperatures) so that overall fuel failure probabilities are low enough to meet incremental failure requirements under normal and off normal conditions. Analytical models for most of these mechanisms have been developed and implemented as part of reactor design tools so that a given reactor core can be assessed against these failure mechanisms.

References:

1. D. A. Petti, J. Buongiorno, J. T. Maki, R. R. Hobbins, and G. K. Miller, "Key Differences in the Fabrication, Irradiation and High Temperature Accident Testing of U.S. and German TRISO-Coated Particle Fuel, and Their Implications on Fuel Performance," Nucl. Eng. Design, 222, 2003, pp. 281- 297.

2. IAEA-TECDOC-CD-1645, *High Temperature Gas Cooled Reactor Fuels and Materials*, International Atomic Energy Agency, March 2010.

RAI FQ-48/MST-52: The two lists of failure mechanisms on pages 19 and 23 do not appear to completely match. (For instance, IPyC cracking, OPyC cracking, and IPyC debonding seem to be categorized differently between the lists. Also issues of "Diffusive release through intact layers," "SiC permeability/SiC degradation," and "SiC failure via heavy metals in buffer/IPyC" are found on one list and not the other.) Discuss the differences between the two lists.

Response FQ-48/MST-52:

It is true that the failure mechanisms listed on page 19 and those in Table 1 on page 23 of the *Fuel Qualification White Paper* do not completely match. For example, failure of the SiC coating caused by

heavy metal dispersion in the buffer and IPyC coating layers does not have a counterpart in Table 1. Likewise, IPyC partial debonding, diffusive release through intact layers, and SiC permeability/SiC degradation in Table 1 do not appear in the list on page 19. Irradiation-induced failure of the outer pyrolytic carbon (OPyC) coating and irradiation-induced failure of the IPyC coating and potential SiC cracking listed on page 19 are covered by irradiation-induced PyC failure leading to SiC cracking.

These discrepancies will be eliminated in the revision of the white paper following resolution of the FQ RAIs. It is noted that some of the so-called failure mechanisms in Table 1 are not really failure mechanisms. Diffusive release through intact layers is not a failure mechanism. SiC permeability/SiC degradation is more properly the result of the potential failure mechanisms of fission product attack of SiC, corrosion of SiC by CO, and SiC thermal decomposition. IPyC partial debonding by itself is not a failure mechanism but can contribute to irradiation-induced PyC failure leading to SiC cracking.

RAI FQ-49/MST-53: Discuss "SiC permeability/SiC degradation" in more detail, and how it is distinguished from "diffusive release through intact layers" and "SiC thermal decomposition."

Response FQ-49/MST-53:

SiC permeability/SiC degradation is incorrectly identified in Table 1 (of the Fuel Mechanism White Paper) as a failure mechanism. It is a condition that can result from failure mechanisms such as fission product attack of SiC, corrosion of SiC by CO, and SiC thermal decomposition. Diffusive release through intact layers is by definition not a failure mechanism, since the layers are intact. SiC thermal decomposition is a failure mechanism at high temperatures in which SiC decomposes into its constituent elements. The confusion in Section 3.1.2 on failure mechanisms and the lack of consistency between the list on page 19 and Table 1 on page 23 will be resolved in the next revision of the white paper.

RAI FQ-50/MST-54: The white paper identifies two main types of chemical attack on SiC: fission products and carbon monoxide (CO). High oxygen inventory may lead to excessive amounts of CO, whereas low oxygen inventory may lead to less fission product oxidation and ultimately more fission product diffusion and chemical attack. However, the impact of low oxygen inventory is not discussed in the white paper. Address the impact of low oxygen inventory.

Response FQ-50/MST-54:

The inventory of oxygen in the fuel particle is determined by the composition of the fuel kernel. UCO is a mixture of uranium dioxide and uranium carbide (for conceptual ease UC_2 is used here although some UC is also usually present), and the oxygen potential is fixed by the equilibrium

$$UC_2 + O_2 = UO_2 + 2C$$

as a function of temperature as long as both uranium dioxide and uranium carbide are present. This thermodynamic relationship is shown in Figure 1 (reproduced from Figure 3 of Reference [1]). Also shown in Figure 1 are equilibria for the oxides and carbides of many fission products. Equilibria at lower oxygen potentials than the UC_2/UO_2 equilibrium, such as those of the rare earths Ce, Y, La, Nd, Pr and Sm, dictate that these elements will be in the form of stable oxides, whereas fission products such as Zr, Sr, Eu, and Ba have equilibria at oxygen potentials higher than the UC_2/UO_2 equilibrium and will be in the form of carbides. The equilibrium for CO production

$$2C + O_2 = 2\,CO$$

has been added to Figure 1 to illustrate that CO is prohibited from forming as long as the oxygen potential is governed by the UC_2/UO_2 equilibrium. Thus, the chemistry of UCO fuel ensures that fission product attack of SiC by rare earths is prevented because those elements are in a stable oxide form, and CO production is also prevented as long as UC_2 and UO_2 are both present in the fuel kernel.

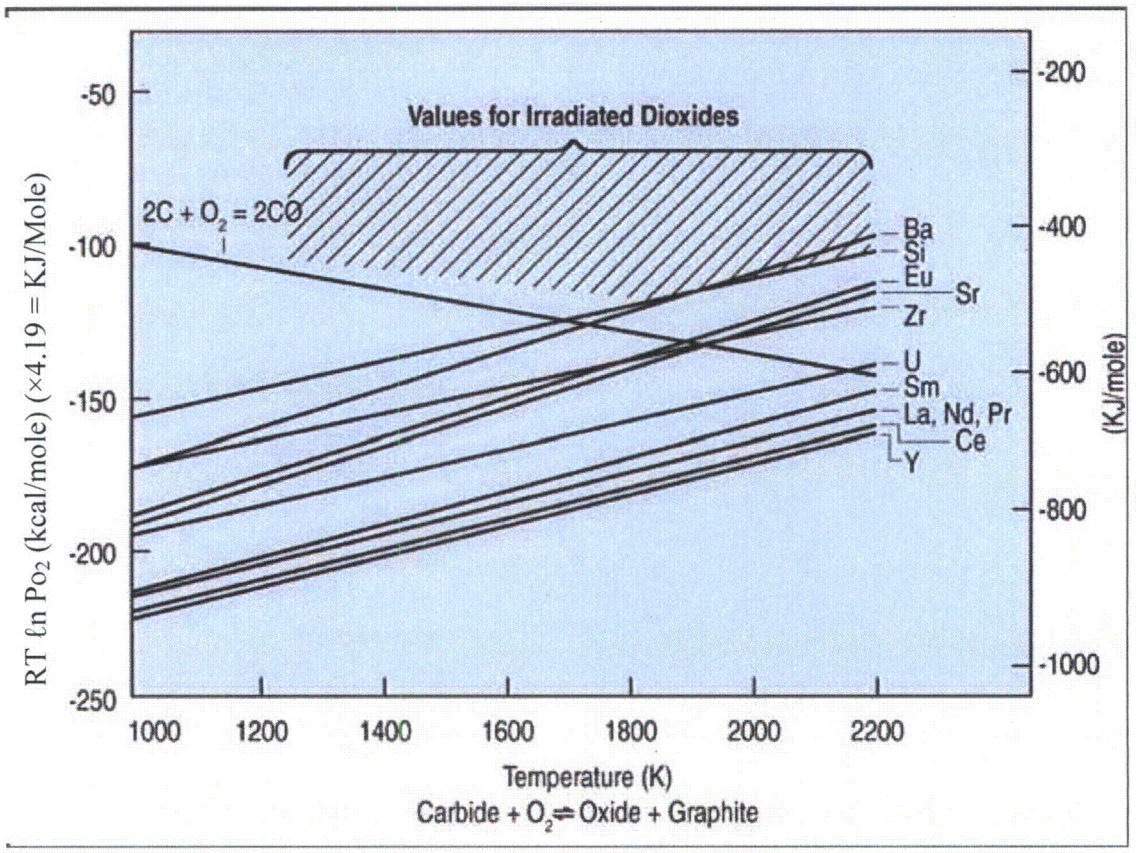

Figure 1. Oxygen potential as a function of temperature (reproduced from Figure 3 of Reference [1])

Interpolation of the data in Figure 2 (reproduced from Figure 4 of Reference [1]) indicates that a starting composition of 15 mole % UC_2 is sufficient to maintain the UC_2/UO_2 thermodynamic equilibrium up to a burnup of approximately 25% FIMA, which exceeds the maximum burnup envisioned for a prismatic design of the NGNP. The fuel specification for irradiations in the AGR Program requires a minimum of 15 mole % UC_2, so control of the oxygen potential in the fuel kernel by the UC_2/UO_2 equilibrium is maintained throughout AGR irradiations to a burnup of about 25% FIMA.

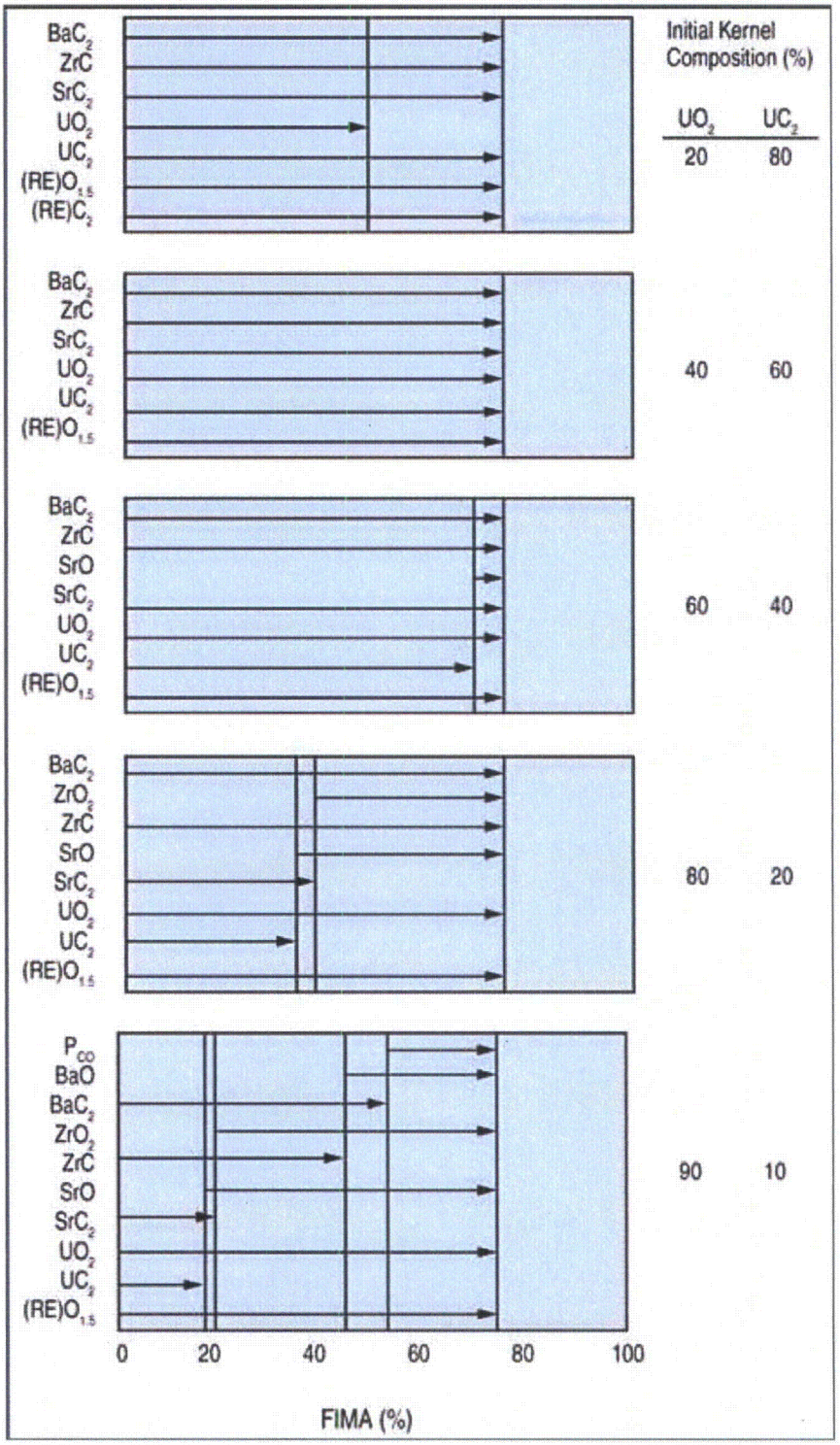

Figure 2. Phases in UCO kernels as a function of burnup and starting fuel composition (reproduced from Figure 4 of Reference [1])

Reference:

1. F. J. Homan, T. B. Lindemer, E. L. Long, Jr., T. N. Tiegs, and R. L. Beatty, "Stoichiometric Effects on Performance of High-Temperature Gas-Cooled Reactor Fuels from the U-C-O System," Nucl. Technol., 35, 428 (1977).

RAI FQ-51/MST-55 (Comment): (Note: The review objectives of the white paper can be addressed in the near term without a response to this comment. The comment may nevertheless be suitable for discussion in another forum.) Provide a discussion that summarizes the advantages and disadvantages of UCO versus UO2 kernels for the NGNP project.

Response FQ-51/MST-55 (Comment):

The design specifications for the NGNP are not yet completely defined, but the maximum burnup envisioned for a prismatic HTGR is in the range 150–200 GWd/MTHM, or 16.4–21.8% fissions per initial metal atom (FIMA) [1] and for a pebble bed HTGR is ~10% FIMA. The Germans have demonstrated excellent performance of SiC TRISO-coated UO_2 particle fuel up to about 10% FIMA, at low power densities (approximately 3 W/cc) and 1150°C at low packing fraction (approximately 10%). UO_2 fuel is completely acceptable in a pebble bed given the less challenging service conditions. However, for a prismatic reactor, UO_2 is known to have limitations because of CO formation (due to chemical reaction of free oxygen released in fission with carbon in the buffer) and kernel migration at the high burnups, power densities, temperatures, and temperature gradients typical of a prismatic HTGR. Given the tight specification for incremental fuel failure under normal operation, UO_2 is viewed as less attractive for a prismatic HTGR. With UCO fuel, the kernel composition (a mixture of uranium carbide and uranium dioxide) is engineered to prevent CO formation and kernel migration, which are potential threats to fuel integrity at higher burnups, temperatures, and temperature gradients. More detail is found in the HTGR training material given to NRC in May 2010.

The higher burnup achievable with UCO would also offer performance advantages for pebble bed concepts in the form of improved fuel utilization. As noted above, this is not required for successful BPR deployment. Although a prismatic reactor can operate with UO_2 fuel, burnup and power densities would probably have to be limited to mitigate carbon monoxide production and amoeba effects of particles in the hottest parts of the core.

Reference:

1. "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," PLN-3636, Idaho National Laboratory, September 30, 2010.

RAI FQ-52/MST-56: Describe any recent or current efforts by DOE or potential vendors to develop new sensor systems for providing on-line measurements of (i) temperatures and (ii) fluxes or power densities in the cores of pebble-bed and prismatic-block HTGRs. If additional technology development is deemed necessary in this area, how is this reflected in the NGNP research and development plans?

Comments: FQ Section 1.3 asks the NRC to "identify any additional information or testing needed to demonstrate adequate NGNP fuel performance." One potential regulatory and licensing concern regarding safety significant parameters is on-line in-core temperature measurement. On-line measurements of in-core temperatures are inherently difficult in HTGRs and have never been performed in any HTGR operated to date. Additionally, historical operational experience has shown that core temperatures have often been higher than predicted/intended, and consequentially, fuel particle integrity

was jeopardized. This issue may be a regulatory and licensing concern and could adversely affect the NGNP development timeline if not resolved promptly during the pre-application period.

Response FQ-52/MST-56:

The historical experience of higher than predicted fuel temperatures mentioned in the comments that accompany this RAI refers to the AVR pebble bed HTGR. The melt wire experiments, with melt wires inserted into selected fuel pebbles, indicated that actual fuel temperatures were considerably higher than predicted by some of the codes and models of the day. These models assumed no core bypass flow and a fuel burnup profile based upon a fairly uniform pebble velocity profile. Recent analyses by PBMR (Pty) Ltd. were able to re-create these higher melt wire temperatures after a detailed investigation of AVR core and reflector geometry effects on bypass flow and with better information about the flow of pebbles near the reflector walls. The pebble and bypass flow profiles of AVR were strongly influenced by the graphite instrument 'noses'. The noses greatly inhibited the adjacent pebble flow and also provided an alternate pathway for coolant flow, thus robbing the core of coolant and elevating the temperature above what was predicted by the simplified model used to predict in-core coolant temperatures.

Unexpected pebble flow patterns were also experienced in the THTR. Although the THTR had no graphite instrument noses like those in AVR, pebble flow along the reflector wall was inhibited by a layer of pebble fragments in the discharge conus. The pebble fragments were caused by the impact of control rods with the fuel.

Modern pebble bed reactor designs feature neither 'noses' nor in-core control rods, and thus these pebble flow inhibiting mechanisms are not present.

In addition to these AVR-specific and THTR-specific issues, other phenomena associated with the flow of pebble fuel elements through the core are being evaluated to improve the fidelity of pebble bed reactor thermal-hydraulic analyses. The elevated temperatures in the central part of the core would be expected to lower the friction between pebbles in that region to a modest extent. This in turn would lead to relatively higher inner core pebble flow and a lower burnup relative to the outer core. This so-called 'slippery inner core effect' would lead to higher fuel temperatures and may have been observed in THTR. The effect is limited, however, and can be captured with reasonable assumptions about pebble flow. An analytical study of the effect is underway at INL.

In summary, with reasonable assumptions for coolant bypass and pebble flow, combined with today's high fidelity physics and thermal-hydraulic analysis capabilities, core flux and temperature profiles, obtained with out-of-core instruments such as those used at FSV, can be computed accurately and reconciled against low-power physics testing and other operational data.

With regard to in core instrumentation, because of their design and composition, HTGRs have large margins between normal operating and fuel failure conditions, and currently there are no requirements for in core temperature, power, or flux measurements. In previous HTGRs, power measurements, including those in support of safety-related control systems, have been made with out-of-core detectors. A few of the FSV initial core fuel elements (up to 32) and replaceable reflector elements (up to 19) included small temperature and fluence monitors in selected fuel compact stacks, but these were for use in post irradiation examination of used fuel. Also at FSV, the 37 region outlet helium temperatures were determined by region outlet thermocouples. In each region, there were four permanently installed thermocouples in the lower permanent reflector plus the capability to accept additional traversing calibration thermocouples. They were ¼ inch I.D. Inconel sheathed, MgO insulated Geminol thermocouples that were used to determine the need for adjustments of the region flow control orifice

valves and thus the helium primary coolant flow through each core refueling region. While these region outlet thermocouples were needed for flow control in the FSV reactor, the modular HTGR does not use flow control valves for various regions or columns of the core, so the need for multiple outlet temperature thermocouples has not been established at present.

While there are no specific instrumentation research and development tasks in the NGNP Technology Development planning documents, various programs to develop instruments with high temperature in-core capability are underway. If it were determined that there is a technical basis for requiring in-core measurements, existing sensors are available that could be used on a non-permanent (movable) basis to measure temperature and neutron flux. Also, fiber optic (ceramic) sensors are under development for in-core use that can withstand temperatures above 1000°C and very high neutron fluences and gamma flux levels. However, the NGNP Project currently has made no commitments to use these in-core sensors.

Advanced (refractory) high temperature thermocouples developed at the INL have undergone testing in the AGR-1 fuel capsule. Early performance was somewhat better than traditional type N thermocouples, but materials issues have yet to be resolved.

Considering other currently available sensors, movable thermocouples could possibly be used in prismatic cores to measure core temperatures. These thermocouples could be inserted through holes formed by drilling out the fuel handling hole, and they would be stored out of the core when not in use to minimize radiation damage. For pebble bed cores these thermocouples could be moved through holes drilled in the reflector blocks. In both cases, the effects of this design alternative on fuel element or reflector performance would need to be assessed before a decision could be made to pursue this approach. Alternatively, melt wires could be used in prismatic fuel elements or in fuel pebbles. As discussed in the response to RAI FQ-26/MST-31, melt wires or other sensors may be used in support of fuel surveillance for the NGNP demonstration plant.

Commercially available self-powered (rhodium) detectors could be used for neutron flux measurements in the prismatic cores. Prismatic reactor designers have studied the use of self-powered (rhodium) neutron detectors for in-core flux measurements in HTGRs. These detectors produce a signal from thermal neutron absorption followed by prompt decay of the rhodium isotope produced. The signal is strong enough to provide a good signal-to-noise ratio and is directly correlated with the neutron flux at the detector. However, the detectors cannot be left in core because of temperature and burnout concerns, so they would need to be used to traverse the core on a regular schedule and then be stored out of core when not in use. Given their size, the best option would be to use a reserve shutdown channel for the detector core traverse. However, before a decision could be made to pursue this option, it would be necessary to assess the effects of such a system on the reserve shutdown system function.

Recent prismatic HTGR designs have used thermal neutron flux detectors (fission chamber), plus thermocouples in a single unit to measure in-vessel thermal neutron flux and temperatures in the core. The unit is movable and is located in the center of the inner reflector; it is withdrawn into the central top head penetration when not in use. The idea is to measure the axial thermal flux and temperature distribution in the central reflector several times a month during operation and compare this to detailed 3-D calculations. The location is chosen to meet the temperature limit on fission chambers, and there are no other in-core detectors.

Radiation hardened and high temperature capable sensors are now under development and are expected to be commercially available within five years. Luna Technologies is developing fiber optic sensors with all ceramic construction. These instruments can measure changes in interference patterns between an optical fiber and a diaphragm, or even from Raleigh backscatter from distributed scattering sites in the fiber, to

infer temperatures. The use of multiple optical property measurements in the fibers to measure neutron flux and gamma dose levels is also under development. These sensors have been tested in the Ohio State University Research Reactor at temperatures over 1000°C, pressures up to 1000 psi, and at fast-neutron fluences of $5x10^{18}$ n/cm^2 (E>1MeV). A current project at Texas A&M University (TAMU) is investigating use of these fiber optic arrays for in-pile flux and temperature measurements. The system is being tested in the TAMU research reactor, and TAMU is in discussions with INL regarding further testing in the ATR. The NGNP Project is planning to have discussions with Japan Atomic Energy Agency (JAEA) about additional testing in the HTTR.

RAI MST-57: For a prismatic-block NGNP design, discuss whether the radial gap between the outside diameter of the fuel compacts and the inside diameter of the fuel hole in the graphite fuel element is credited and modeled as a holdup mechanism or barrier in calculating the mechanistic source terms.

Response MST-57:

The radial gap between the outside diameter of the fuel compacts and the inside diameter of the fuel hole in the prismatic graphite fuel element is relatively small, nominally 0.13 mm (0.005 in) for fresh fuel. The gap annulus is sealed at the bottom because the fuel holes in the graphite fuel element are drilled as blind holes that do not penetrate the entire length of the graphite block. It is sealed at the top by the graphite fuel hole plug, which is glued in place at the top of the fuel element during fuel fabrication after the fuel compacts are loaded into the fuel hole. As a result, the gap is a small, essentially stagnant volume that in and of itself, neither provides an inherent resistance to nor a driving force for fission product transport.

As noted in Section C-5 of the Mechanistic Source Terms white paper, the fuel matrix material is relatively porous and provides little holdup of the fission gases (including halogens) that are released from the fuel particles. The effect is generally neglected in calculations of fission gas transport in the fuel matrix, as well as in the fuel block graphite.

It is therefore assumed that the fuel compact/graphite gap has no effect on the transport of these gaseous fission products.

As also noted in Section C-5, the fuel compact matrix is a composite material that has a high content of amorphous carbon, and this constituent of the matrix is highly sorptive of metallic fission products, especially strontium and actinides. While the matrix is highly sorptive of metals, it provides little diffusional resistance to the release of fission metals because of its high interconnected porosity. The graphite block of a prismatic fuel element, which is denser and has a more ordered structure than the fuel matrix material, is somewhat less sorptive of the fission metals than the matrix, but it is more effective as a diffusion barrier.

For transport of fission metals during normal operation the analysis model assumes that sorption equilibrium prevails in the gap between the fuel compact and the fuel hole surface of the fuel element. At equilibrium, the metallic fission product vapor pressure in the gap and the solid phase concentration on the fuel compact surface are uniquely related to one another by a fuel matrix sorption isotherm that is determined experimentally. Likewise, the metallic fission product solid phase concentration on the graphite fuel hole surface and the fission product vapor pressure in the gap are also uniquely related to one another by a graphite sorption isotherm. Hence, the sorption characteristics of the fuel compact matrix material determine the amount of metallic fission products that are released to the gap, and the sorption characteristics of the fuel element graphite determine the amount of metallic fission products that

are sorbed by the fuel element graphite and are thus available to diffuse through the graphite web to the coolant hole surface.

In this representation, the fuel compact/graphite gap itself is not modeled as a holdup mechanism or a barrier to metallic fission product release. It is the relative sorptivity of the fuel compact matrix and the fuel element graphite, at their respective calculated temperatures, that determine the amount of metallic fission product that is transferred from the fuel compact into the gap and then onto the graphite for subsequent transport to the coolant hole surfaces.

For transport of fission metals during off-normal events, the analysis model conservatively neglects the effect of the relative sorptivity of the fuel compact matrix and the fuel element graphite on the transfer of metallic fission products across the gap.

RAI MST-58: It is understood by the NRC Staff that the fuel irradiation and accident condition testing programs will focus on obtaining data for specific species of specific gaseous and metallic fission products. Discuss how these limited data will be used (i.e., extended) to model the transport of all of the radiologically significant species of fission products.

Response MST-58:

As noted in Section 4.4 of the Mechanistic Source Terms White Paper, particularly in Table 4-2 and its accompanying text, the radionuclides of interest in modular HTGR design and safety analysis can be broken into classifications for analysis purposes. Classifications are, in most cases, based simply on the periodic table of the elements. The radionuclides within a classification exhibit similar in-core and ex-core behavior for the purpose of radionuclide transport modeling. They have similar transport behavior in core materials and similar sorption and liftoff behavior for materials in the core and in the primary circuit. Source term calculations take into account differences in half-life and fission product yield of each radionuclide.

Analyses of fission product transport in the modular HTGR include 250* radionuclides, including all those that are radiologically significant. The properties of the radionuclide classes presented in Table 4-2 of the Mechanistic Source Terms White Paper are such that it is not necessary to collect data on all radionuclide species that are analyzed in the calculation of mechanistic source terms. Testing programs are designed to collect transport data on all radionuclides in each classification that are resolvable by beta and gamma spectroscopy. Fuel safety testing may be preceded by reactivation to obtain data on radiologically important short–lived radionuclides such as I-131. The data obtained are conservatively extrapolated to other radionuclides in the group. For example, although significant release of barium radionuclides has not been measured in fuel safety tests, it is conservatively assumed for analysis purposes that barium behavior is the same as that of strontium. As another example, iodine release from fuel kernels is assumed to be the same as xenon release, which is conservative based on measurements of iodine and xenon release from UO_2 and UC_2. This conservatism will be confirmed by testing of UCO fuel in the AGR Fuel Development and Qualification Program.

This approach to testing and data extrapolation ensures that no radiologically important radionuclides are omitted.

*This value pertains to analysis methods used by General Atomics. Methods used by other HTGR reactor designers are similar, but the specific number of radionuclides analyzed may vary.

RAI MST-59: Discuss the extent to which the helium purification system which is provided to remove gaseous and solid impurities (including, radionuclides) circulating in the primary system during normal

operation is credited and modeled as part of the transport and release of radionuclides in the mechanistic source term calculations.

Response MST-59:

The Helium Purification System (HPS) is credited and modeled as part of calculation of the transport and release of radionuclides during normal operation and for those Licensing Basis Events (LBEs) in which the HPS continues to operate. Since the HPS is not safety-related, it is not credited and modeled in the analysis of Design Basis Accidents (DBAs). The extent to which the HPS affects the circulating activity of various radionuclides depends on the radionuclide half life and on whether the radionuclide species is gaseous or condensable.

While the primary purpose of the HPS is to control chemical impurities, it also serves to remove radionuclides from the circulating helium coolant, including noble gases and tritium. The HPS also removes condensable radionuclides, including iodine isotopes and volatile fission metals (Ag, Cs, Sr, etc.), but these condensable radionuclides typically deposit ("plate out") on the surfaces in the primary coolant circuit much more rapidly than they are removed by the HPS.

In the simplest case, the primary coolant activity of member i of a decay chain is given by [1]:

$$\frac{dA_{i,circ}}{dt} = (R/B)_i B_i - (\lambda_i + r_{HPS} + r_{po}) A_{i,circ} \quad (1)$$

and

$$r_{HPS} = \frac{\dot{m}_{HPS}}{M} \quad (2)$$

and

$$r_{po} = -\frac{\dot{m}_{He}}{M} \ln(1-\tau) \quad (3)$$

where: $(R/B)_i$ = release rate-to-birth rate ratio for nuclide i (numerically equal to the steady-state fractional release),

$A_{i,circ}$ = circulating activity of nuclide i (Ci),

B_i = birth rate for nuclide i (Ci/sec),

λ_i = decay constant (sec^{-1}),

r_{HPS} = purification constant (sec^{-1}),

r_{po} = plateout removal constant (sec^{-1}),

$\dot{m}_{HPS}$ = side stream HPS mass flow rate (kg/sec),

M = total mass of circulating He in primary circuit (kg),

$\dot{m}_{He}$ = primary He mass flow rate (kg/sec),

τ = "plateout per pass," the fraction of a condensable radionuclide removed during one transit through the primary circuit.

With the initial condition of $A_{i,circ} = 0$ at $t = 0$, Equation (1) is integrated to give:

$$A_{i,circ} = \frac{(R/B)_i B}{\lambda_i + r_{HPS} + r_{po}} \left(1 - e^{-(\lambda_i + r_{HPS} + r_{po})t}\right) \tag{4}$$

Typically, the HPS in a modular HTGR has an effective half-life of approximately 4.5 hr (i.e., the concentration of a chemical impurity in the primary coolant, without a continuous source, is reduced by half in 4.5 hr). By manipulation of Equation (4), one can show that the effect of HPS on the circulating noble gas activity is given by the following relation:

$$\frac{A_{w/HPS}}{A_{wo/HPS}} = \frac{4.5}{4.5 + t_{1/2}} \tag{5}$$

where: $A_{W/HPS}$ = circulating activity with HPS (Ci)

$A_{WO/HPS}$ = circulating activity without HPS (Ci)

$t_{1/2}$ = radioactive half life (hr)

Consequently, if radionuclide half live $t_{1/2} << 4.5$ hr (e.g., 33-sec Kr-90), the HPS has no effect on the circulating activity of the radionuclide. However, if radionuclide half life $t_{1/2} >> 4.5$ hr (e.g., 5.3-day Xe-133), the HPS has a large effect on the circulating activity of the radionuclide.

The circulating activity of condensable radionuclides in the primary circuit is calculated using an appropriate plateout per pass (τ). To account for the possibility of "saturation" and circulating dust borne activity, a conservative factor of 40 is applied to the plateout per pass (i.e., 40% plateout per pass for the "Maximum Expected" and 1% per pass for the "Design" criteria discussed in the response to RAI MST-87) when calculating the circulating activities of all condensable species except iodine.

Measurements in Fort St. Vrain and in-pile loops suggest an iodine plateout per pass of the order of 1% to 10%. Consequently, for iodine 1% per pass is assumed for both the "Maximum Expected" and "Design" calculations.

Even with a plateout per pass as small as 1%, more than 99% of the condensable activity is plated out, so the contribution of the HPS to removal of condensable radionuclides from the primary circuit is very small.

Reference:

1. Haire, M.J., and D.W. McEachern, "Gaseous Radioactivity Levels in the Primary Coolant of an HTGR," GA-A12946 (GA-LTR-14), General Atomic, October 1974.

RAI MST-60 (Comment): Discuss the extent to which filtration systems located in the low pressure reactor-building are credited and modeled as part of the transport and release of radionuclides in the mechanistic source term calculations for normal operations, transients and accidents.

Response MST-60 (Comment):

Various reactor-building options have been studied for both prismatic and pebble bed modular HTGR designs. The response to RAI MST-82 provides more details on the reactor-building options considered in prior studies and their advantages and disadvantages. As the design and technology development proceeds for the NGNP Project, the details of the reactor building, including the extent to which filtration systems are credited and modeled in mechanistic source term calculations, will be determined.

RAI MST-61: Discuss the inherent or passive safety characteristics of HTGRs that are intended to reduce or limit the extent of chemical attack (e.g., oxidation) of the fuel with its associated increase of fuel particle failures, as well as the safety characteristics of HTGRs that are intended to reduce or limit the rate of transport of radionuclides (i.e., via natural circulation of air though the core). Discuss the basis for the basis for the statement that "...steps have been taken to prevent ingress of contaminants..." given that NGNP has not yet progressed beyond the conceptual design stage.

Comments: Excerpt from MST Section 2.3.4: "...The severity of these events may also be affected by air or water ingress, which can increase heat generation, produce steam and other gases (e.g., H_2, $C0_2$ and/or CO), affect the fuel and core graphite structure, and increase the rate of release of some fission products from the fuel. Steps have been taken to prevent ingress of contaminants, and consequences are expected to be acceptable if they occur.

Response MST-61:

Steps to address water and air ingress have been taken from the origination of the HTGR concept. The helium coolant is a fundamental choice that took into account the experience of early gas-cooled reactors that utilized first air and then CO_2 as coolants. The seven licensed HTGRs to date have had excellent experience with pressurized helium that is chemically inert and neutronically transparent. Contaminants are managed at the parts per million levels by the Helium Purification System that continuously operates by purifying a side stream from the primary system.

Chemical attack on fuel particles and on the graphite core structure can result from off-normal events that lead to water or air ingress into the primary system. The following discussion is derived from the Preliminary Safety Information Document (PSID) for the Modular HTGR [1]. Although specific design details may vary and will be determined as the NGNP design effort progresses, it is expected in general to be applicable also to the NGNP.

Water Ingress Events

The likelihood of water entering the primary system is limited by the absence of high pressure and high energy sources of water in proximity to the primary system, with the exception of the higher pressure steam/water in the steam generator (SG) in steam-cycle HTGR designs. Under normal operating conditions, all other water coolers and heat exchangers operate at lower pressures than the pressure of the helium with which they exchange heat. In the event of a tube leak in the higher pressure SG, the leak would be detected by the moisture monitors of the Investment Protection System (IPS) or, if that system fails, on detection of high primary coolant pressure by the Reactor Protection System (RPS). Both the IPS and RPS isolate the SG to limit the amount of water ingress to the reactor by actuating feedwater and main steam isolation valves that are powered by an Uninterruptable Power Supply (UPS). In addition, the IPS dumps the SG to water tanks to limit the moisture cleanup time. The reactor would also be tripped by insertion of control rods by either the IPS or RPS.

Reactor core cooling is designed to be provided by the Shutdown Cooling System (SCS) to limit the potential for chemical attack of the graphite reactor internals by water from SG leaks. If this forced core cooling system is unavailable, the potential of water transport to the core would be limited by the lack of circulation from the SG vessel to the hotter reactor vessel. Moisture that does get transported to the reactor by leakage through the Heat Transport System (HTS) main circulator shutoff valve or by diffusion/convection back through the hot duct would be expected to react with graphite above ~370°C (at reactor locations during normal operation other than the cold plenum).

However, the reaction of steam and graphite is slow and endothermic and therefore is not self-sustaining. The reaction of one molecule of steam with graphite leads to two molecules (H_2 and CO). If moisture reaches the active core, to reach the fuel particles it must diffuse 1) without first reacting with any graphite along its path, 2) against any pressure build up from those steam molecules that have reacted, and 3) against the temperature gradient from the fuel outward through the prismatic fuel elements and the fuel compacts or the fuel spheres. Exposure to moisture does not affect fuel particles except for the very small fraction (on the order of 10^{-5} to 10^{-4}) with exposed fuel kernels. Gaseous radionuclide release from such particles would increase as a result of fuel kernel hydrolysis, and the overall release from the fuel to the HPB during the pressurized conduction cooldown would be incrementally increased. . It is also possible that H2O or the reaction product CO would cause fission metals, especially Cs, sorbed on the matrix and graphite to be released. The latter has not been quantified but will be evaluated in the AGR program.

The primary adverse consequence of water ingress depends on whether the added amount of steam and reaction products causes the primary system pressure to increase to the level where the vessel system relief valve lifts one or more times. If it does lift, it may reseat as expected, or it may fail open relieving the mixture of mostly helium gas, along with reaction products and volatile radionuclides, to the SG cavity of the reactor-building and from there out the reactor-building vent to the environment. These design specific event scenarios were evaluated for the MHTGR in the PRA and PSID. The water ingress sequences led to the highest consequences for that design, but these still met the Top Level Regulatory Criteria.

Air Ingress Events

The likelihood of a breach of the helium pressure boundary (HPB) such that air ingress becomes a concern is limited by the high integrity associated with the pressure vessels and the limited size of penetrations. In the event of a leak or break, primary helium coolant would leak out until the HPB and reactor-building pressures equilibrated (from minutes to days for breaks and leaks, respectively). Most importantly, the higher pressure helium would act to expel the reactor-building air out from the below-grade, multi-compartment reactor-building up to the above-grade vent.

After trip of the reactor by either the IPS or RPS, forced core cooling of the reactor can be provided by either the HTS or the SCS to limit the potential for chemical attack of the air with graphite reactor internals. If neither of these two independent forced convection core cooling systems is available, the resulting conduction cooldown to the RCCS leads to a depressurized heatup of the reactor core and primary system that causes the helium to expand for one to two days. For smaller leaks this expansion will be masked by the helium depressurization; for breaks the expansion follows the depressurization. After the depressurization/expansion phases, the helium-air gas mixture will be drawn into the HPB while the reactor graphite temperatures are relatively high.

The ingress rate of the helium-air gas mixture into the HPB due to contraction would be small, as the gas mixture tries to enter the breach primarily by natural circulation and diffusion at the same time as helium

coolant, which it is displacing, tries to exit through the same hole. Large gas mixture ingress rates would require an implausible scenario of two concurrent breaches of an ASME Code Section III vessel in order to set up an effective circulation path. However, even in that circumstance, the gas mixture flow would be restricted by the flow resistance characteristics in the core.

In contrast to coal and charcoal, graphite is a highly structured, highly dense, and very pure form of carbon that will not allow self-sustained combustion. Nuclear grade graphites have lower levels of impurities than conventional graphites. As a result, they are even more difficult to oxidize. Demonstration tests have shown negligible mass loss for nuclear-grade graphite under conditions for which charcoal and coal were completely consumed and reduced to ash. The combustion characteristics of graphite are more like that of diamond, another highly structured and highly pure form of carbon.

Nevertheless, the O_2 in the helium-gas mixture will react to some degree with the nuclear grade graphite in one of two exothermic reactions that produce either CO or CO_2. As with water ingress, any air in the gas mixture that enters the primary coolant must react with graphite elements and prismatic/pebble fuel elements before it can reach and chemically react with the embedded refractory-coated fuel particles.

Finally, the amount of air in the predominately helium mixture is limited by the size of the low leakage below-grade reactor building. As a result, the overall heat of reaction of air with graphite remains small relative to the core decay heat. Prior analyses of the MHTGR did not find the air ingress phenomena to have a significant impact on the safety margins to the offsite consequences.

In summary, control of chemical attack has been a focus of the safety design approach from the inception of the HTGR. A blend of active and passive design features have been utilized with the inherent characteristics of the inert high pressure helium coolant, nuclear grade graphite moderator, and coated particle fuel to assure adequate safety margins.

Reference:

1. "Preliminary Safety Information Document for the Standard MHTGR," HTGR-86024, Rev. 13, September 1992

RAI MST-62: Discuss the technical basis as well as the safety basis for applying a factor of four as the design margin for the release of fission gases from the core while applying a factor of 10 as the design margin for the release of fission metals from the core to account for uncertainties in the design methods.

Response MST-62:

As described in Section 2.4 of the Mechanistic Source Terms (MST) White Paper and in the response to RAI MST-87, standard design practice in the U.S. HTGR program has been to define a two-tier set of radionuclide design criteria, - referred to as Maximum Expected and "Design" criteria - for allowable core releases for normal operation and AOOs. The "Design" criteria are derived from externally imposed requirements, such as site-boundary dose limits, occupational exposure limits, etc. Once the "Design" criteria have been derived from the radionuclide control requirements, the corresponding "Maximum Expected," criteria are derived by dividing the "Design" criteria by an uncertainty factor, or design margin, to account for uncertainties in the design methods. This uncertainty factor is typically a factor of four for the release of fission gases from the core and a factor of 10 for the release of fission metals from the core.

These design margins of four for gas release and 10 for metal release from the core are based largely on engineering judgment. The radionuclide design criteria that incorporate these design margins, or

uncertainty factors, provide a figure of merit for judging the adequacy of candidate core and fuel designs. From these allowable core releases, corresponding in-service fuel performance requirements (e.g., fuel failure fractions, etc.) and, in turn, as-manufactured fuel quality requirements (e.g., heavy-metal contamination fraction, SiC defect fraction, etc.) can be derived. These design margins also provide valuable guidance to the supporting technology development programs regarding required predictive accuracies of the design methods used to predict fuel performance and fission product release from the core.

A smaller design margin is specified for fission gas release (factor of four) than for fission metal release (factor of ten) because predicting gas release is less complex, and gas release is less sensitive to service conditions in the core. As described in Appendix C of the Mechanistic Source Terms White Paper, the dominant sources of fission gas release, which includes iodine and tellurium isotopes as well as noble gas isotopes, from a modular HTGR core are (1) as-manufactured heavy-metal contamination and (2) exposed fuel kernels. The heavy metal contamination fraction is controlled by the fuel product specification, and the exposed kernel fraction is predicted with fuel performance models. Subsequently, the fractional releases of fission gases from heavy metal contamination and exposed kernels are predicted with experimentally determined release correlations.

Fission metal release from the core is more complex. In addition to metal release from heavy metal contamination and exposed kernels, volatile metals (Ag, Cs, Sr) can also be released from "partially failed" particles with defective or failed SiC coatings but with at least one pyrocarbon coating intact. Moreover, Ag isotopes can diffuse through intact TRISO coatings if the temperature is sufficiently high. Once these volatile metals have been released from fuel particles, they must migrate through the fuel compact matrix, across the gap between the fuel compact and the graphite block, through the graphite web, and finally into the flowing helium coolant. In other words, there are many additional transport steps in the release of fission metals from the core as compared to fission gases. Moreover, the activation energies for metal transport in core materials are much larger than those for gas transport so that the former is more sensitive to uncertainties in predicted core temperature distributions.

In principle, the design margins could be adjusted as the design evolves through the conceptual, preliminary, and final phases. For example, if the NGNP/AGR fuel development and qualification program is successful in developing and validating fuel performance and fission product transport models with experimentally derived uncertainties that are much smaller than four times for gas release or ten times for metal release, then the design margins could be reduced.

While the current design margins are based largely upon engineering judgment and subject to refinement as the design and technology programs progress, they appear to be reasonable and attainable goals based upon the current knowledge base. As summarized in Section 4.6.1 of the Mechanistic Source Terms White Paper and elaborated in [1], a number of comparisons have been made between code predictions and integral test data from operating HTGRs and in-pile tests to assess the predictive accuracy of the design methods. The comparisons made for the FSV HTGR are perhaps the most relevant to date. Both fission gas release and fission metal release at FSV were predicted to well within factor of four and factor of ten accuracy goals, respectively. For a modular HTGR, an important consideration will be the accuracy to which the fuel performance and fission product release can be predicted for the AGR-7 and AGR-8 validation tests.

As noted in Section 2.4 of the Mechanistic Source Terms White Paper, the use of a factor of four for the release of fission gases from the core and a factor of ten for the release of fission metals as the design margin between the Maximum Expected criteria and "Design" criteria in the U.S. HTGR program is similar to the practice that was used in the South African PBMR program. The PBMR program used a

different approach to determining design margins for some fission product elements - sensitivity analyses of uncertainties in factors that affect fission product transport. The similarity of their results provides further confidence in the appropriateness of the design margins used in the U.S. HTGR program.

References:

1. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," IAEA-TECDOC-978, International Atomic Energy Agency, November 1997.

RAI MST-63: Given the health effects importance of the chemical forms of radioactive iodine gas release and the uncertainties in the design methods, discuss why a factor of four would be considered appropriate rather than a factor of ten.

Response MST-63:

Section 2.4 of the Mechanistic Source Terms (MST) White Paper and the responses to RAIs MST-62 and MST-87 provide the bases for and implementation method of the design margins of four times for fission gas release and ten times for fission metal release from the core. These margins are intended to compensate for the projected uncertainties in the predicted release rates from the core. The magnitude of the assigned margins is commensurate with estimated uncertainties in the predictive methods rather than the radiological importance of a particular radionuclide.

With regard to the chemical forms of radioactive iodine released from the HTGR, the available experimental data indicate that iodine isotopes are released from a modular HTGR core as elemental iodine [1,2]. The partial pressures of iodine isotopes in a modular HTGR during normal operation and during depressurized core conduction accidents are too low for the formation of stable compounds in the core, primary circuit, or reactor building. This iodine behavior is contrary to that indicated for LWRs during core-melt accidents, for which there is experimental evidence that iodine is present as cesium iodide and, for certain types of control rods, as silver iodide. The fractional releases of iodine and cesium from HTGR cores and in fuel irradiation tests are very different, and the plateout distributions of cesium, silver and iodine isotopes in operating HTGRs and in in-pile loop tests are very different as are liftoff fractions during depressurization tests. Accordingly, iodine released from the HTGR is unlikely to be combined with the other elements as it is in the radionuclides released from an LWR.

The release rates of iodine isotopes are assumed to be equal to that of xenon isotopes of the same half life. During normal plant operation, this assumption is evidently conservative for iodine isotopes by approximately two times, based upon measured fission gas release rates for UC_2 and UO_2 TRISO fuel particles [2]. For the reference UCO TRISO fuel particles, the fission gas release rates will be measured directly in the AGR-3/4 test and independently validated in the AGR-8 test. Additional iodine design margin is added when the circulating iodine inventory is calculated. To account for the possibility of "saturation" and circulating dust borne activity, an additional factor of 40 is applied to the "plateout per pass" (i.e., 40% plateout per pass for the "Maximum Expected" and 1% per pass for the "Design" release determination) when calculating the circulating activities of all condensable species except iodine. Measurements in Fort St. Vrain and in-pile loops suggest an iodine plateout per pass of the order of 1% to 10%. Consequently, for iodine only 1% plateout per pass is assumed for both the "Maximum Expected" and "Design" circulating inventory calculations. Moreover, a factor of 10 margin is added to the predicted iodine fractional liftoff during rapid depressurization accidents. Hence, in the case of iodine, there is already significant additional conservatism in the assumed release rates in addition to the applied design margin of four times.

The inventory of iodine isotopes in the primary circuit can be monitored during normal plant operation. The fractional releases (R/Bs) of xenon isotopes from the core will be measured continuously, and as described above the fractional releases of iodine isotopes can be conservatively inferred from the measured xenon R/Bs. The plateout inventories of I-133 and I-135 can be determined by measuring the concentrations of their Xe daughters in the primary coolant during plant shutdowns. Finally, it is anticipated that there will be plateout probes in the primary circuit for measuring the core release rates of condensable radionuclides, including I-131 (as was done in Peach Bottom 1 and Fort St. Vrain).

References:

1. Martin, R.C., "Compilation of Fuel Performance and Fission Product Transport Models and Database for MHTGR Design," ORNL/NPR-91/6, Oak Ridge National Laboratory, October 1993.

2. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," IAEA-TECDOC-978, International Atomic Energy Agency, November 1997.

RAI MST-64: Discuss whether radionuclide-specific design margins may be proposed for the NGNP that differ from a factor of four for fission gases and a factor of 10 for fission metals.

Response MST-64:

As discussed in response to RAI MST-62, the radionuclide design margins could, in principle, be adjusted as the design evolves through the conceptual, preliminary and final phases and the NGNP/AGR fuel technology program progresses. For example, if the NGNP/AGR fuel development and qualification program is successful in developing and validating fuel performance and fission product transport models with experimentally derived uncertainties that are much smaller than four times for gas release or ten times for metal release, then the design margins could be reduced. Any changes or updates would be provided by the license applicant.

RAI MST-65: Provide a few illustrative examples of the kinds of NGNP licensing basis events that would be viewed as a "major accident" to be considered under 10 CFR 52.17. Discuss whether such events would be expected to be found in the DBE or BDBE frequency range.

Response MST-65:

Three illustrative examples of the kinds of NGNP licensing basis events that might be viewed as appropriate HTGR events for the site safety assessment under 10 CFR 52.17 are as follows:

1. Moderate primary coolant leak (pressure relief line) with loss of main loop and auxiliary cooling systems
2. Small primary coolant leak (instrument line) with loss of main loop and auxiliary cooling systems
3. Steam generator tube break without moisture monitor detection or steam generator dump and without auxiliary cooling system

In the MHTGR PSID [1], these events are identified as SRDC-10, SRDC-11, and SRDC-6, respectively. All three of these events rely only on safety-related structures, systems, and components (SSCs), so they would be Design Basis Accidents in addition to being in the BDBE frequency range.

It is important to note that, due to the safety characteristics of the HTGR, these potential accidents for consideration under 10 CFR 52.17 do not have consequences like those of LWR "major accidents" considered under this regulation. In particular, it is expected that the off-site dose consequences of these events would be such that the plume exposure EPA Protective Action Guides are not exceeded at the Exclusion Area Boundary.

Reference:

1. "Preliminary Safety Information Document for the Standard MHTGR," HTGR-86024, Rev. 13, September 1992

RAI MST-66: Discuss why BDBEs would not also be evaluated for purposes of containment design.

Comments: Excerpt from Table 3-1: "Chapter 15 accident analyses will include a deterministic analysis of the release from the reactor-building for all DBAs to determine the calculated source term for each event. The reactor-building leak rate and fission product cleanup systems (if any) will be taken into account in determining radionuclide release from the reactor building."

Excerpt from page 18: "Events considered in the analyses to develop the set of source terms for each design are selected to bound severe accidents and design-dependent uncertainties."

Response MST-66:

The excerpt from Table 3-1 quoted in the RAI is included in several rows of the table. The excerpt from page 18 is taken from the third bullet on that page of the Mechanistic Source Terms White Paper. The bullets are presented as a summary of the NRC staff recommendations in SECY-93-092 regarding conditions under which use of mechanistic source terms should be allowed.

In the last row of Table 3-1, the Mechanistic Source Terms White Paper states, "A mechanistic source term that projects the radionuclide release to the environment from the reactor-building will be calculated for each LBE." Since, as explained in the LBE Selection White Paper, BDBEs are included in the scope of LBEs, BDBEs will be evaluated in source term calculations. Source term calculations are an evaluation of the design and performance of the functional containment of the HTGR. Therefore, BDBEs will be evaluated for purposes of determining the performance of the functional containment, which includes the five barriers to release of radionuclides: fuel particle kernel, fuel particle coatings, fuel matrix and fuel element graphite, the helium pressure boundary (primary circuit), and the reactor building.

RAI MST-67: Discuss how the statement from the July 30, 1993, Staff Requirements Memorandum (SRM)," that allows the use of scenario specific source terms, provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties". will be interpreted and applied in the NGNP deterministic safety analyses (i.e., dose consequence analyses) to mean that a bounding approach - applying deterministic engineering judgment - (rather than a best estimate approach) will be used to calculate the scenario specific MST for AOOs and BDBEs, as well as DBAs.?

Response MST-67:

The modular HTGR safety characteristics allow use of a systematic Monte Carlo uncertainty assessment that is based upon the "understanding and assurance of plant and fuel performance" referred to in this RAI. Deterministic engineering judgment is a key ingredient in selecting the uncertainty distributions for

each of the independent variables. The resulting output distributions provide a basis for judging acceptability and safety margins for a range of requirements. These requirements include those for which an upper bound evaluation is appropriate to demonstrate safety margin, such as for Design Basis Accidents (DBAs) in Chapter 15 of a safety analysis report relative to the regulations of 10CFR50.34. The requirements also include those for which expected, best estimate safety evaluations are appropriate, such as for assessing normal operation and Anticipated Operational Occurrences (AOOs) to determine off-site dose, and assessing Design Basis Events (DBEs) and Beyond Design Basis Events (BDBEs) relative to the emergency planning Protective Action Guides.

This approach to the uncertainty assessment allows calculation of source terms and offsite dose to be conducted at any desired confidence level (e.g., mean or upper 95% confidence) for each Licensing Basis Event (LBE) – AOOs, DBEs, or BDBEs.

RAI MST-68: The white paper does not discuss the proposed licensing basis for calculating the scenario-specific mechanistic source terms for NGNP LBEs in the AOO frequency range, the DBE frequency range, or the BDBE frequency range. Nor does it discuss the proposed basis for calculating scenario-specific source terms for DBAs or emergency planning licensing basis events. For each of these event categories, discuss whether a best-estimate approach or a conservative approach will be used for the calculating the licensing basis mechanistic source terms. Where a best estimate approach is to be used, discuss the technical approach for the calculation. Where a conservative approach is to be used, discuss the technical approach for the calculation. The requested discussions should address whether a best-estimate or conservative approach will be used for predicting the long-term accumulation in the primary circuit of fission products released from the fuel during normal operation as needed for predicting the initial mechanistic source terms (i.e., radionuclide releases for each of the above event categories).

Comment: Page 20 of the white paper notes that the framework approach described in SECY-05-006 included the following features: Source terms for compliance should be 95% confidence level values based on best-estimate calculations. Source terms for emergency preparedness should be mean values based on best-estimate calculations.

Response MST-68:

The LBEs (AOOs, DBEs, and BDBEs) will be evaluated with the entire plant responding as anticipated and with the mechanistic source terms predicted on a best estimate basis with an uncertainty distribution. The initial radionuclide inventories within the helium pressure boundary at the onset of all LBEs are the levels allowable by the technical specifications.

The results for each LBE category will be reported showing the mean, lower 5% bound, and upper 95% bound offsite dose consequences to show compliance with and margins to the Top Level Regulatory Criteria. The 10CFR20 offsite dose acceptance criteria for the AOOs are compared with the mean. The 10CFR50.34/52.79 offsite dose criteria for the DBEs are compared with the upper bound. The EPA PAG offsite dose criteria are compared with the mean. The entire offsite dose consequence distribution for all LBEs is included in the assessment of cumulative individual risk relative to the Quantitative Health Objectives.

The DBAs are derived from the DBEs by deterministically assuming that only safety-related SSCs are available, and their upper bound consequences will be shown to meet the 10CFR50.34/52.79 offsite dose criteria. This process is described in more detail in the Licensing Basis Event Selection White Paper [1].

Reference:

1. INL/EXT-10-19521 Next Generation Nuclear Plant Licensing Basis Event Selection White Paper, September 16, 2010

RAI MST-69-(Comment): (Note: The review objectives of the white paper can be addressed in the near term without a response to this comment. The comment may nevertheless be suitable for discussion in another forum.) Page 16 of the white paper states that: "The assessment will also identify areas where new regulations are needed to address HTGR technology, as applicable." However, the NGNP Licensing Strategy proposes that the NGNP license be based on (applying and/or adapting) existing regulations, requirements and guidance. It does not propose that NGNP licensing be based on establishing new regulations. Moreover, the white paper does not appear to identify any areas where new regulations are needed to address HTGR technology.

Response MST-69:

The assessment referred to on page 16 of the Mechanistic Source Terms White Paper is the Regulatory Gap Analysis. This effort is currently underway and is scheduled to be completed by the end of FY11. The results will be shared with the NRC at that time.

RAI MST-70: Describe any plans for peer review of the NGNP mechanistic source terms by appropriately qualified subject matter experts.

Comments: Development of the LWR Alternate Source Term included peer review by appropriately qualified subject matter experts.

Response MST-70:

At present, there are no plans for peer review of the NGNP mechanistic source terms that is strictly analogous to that conducted for the LWR Alternate Source Term. Requirements for independent review of technology development activities that support source term calculations and of source term calculations themselves, such as the independent review requirements of 10CFR50 Appendix B, will be met using qualified subject matter experts. Following resolution of the RAIs, the Mechanistic Source Terms White Paper will be revised to clarify that there are no plans for peer review of mechanistic source terms that is analogous to that conducted for the LWR Alternate Source Term.

RAI MST-71: For the NGNP, discuss whether the major accident selected for the NGNP siting assessment will involve an examination of events from the BDBE region as well as events in the DBE region (i.e., DBAs).

Response MST-71:

The NGNP siting assessment will include examination of events from the BDBE region as well as events in the DBE region. However, it is anticipated that, based on previous experience in modular HTGR safety analyses, there will be no equivalent of a maximum hypothetical accident such as the large break LOCA of an LWR with significant core melt and radionuclide release.

RAI MST-72: What process is/was used to identify the underlying phenomena that are considered important to fission product transport and release (e.g., diffusion through intact particle coatings) in the mechanistic source term calculation for the NGNP plant design.

Response MST-72:

To identify the important underlying phenomena for fission product transport and release in the mechanistic source term calculation, the NGNP Project is drawing upon more than fifty years of HTGR technology development.

As summarized in Appendix C of the Mechanistic Source Terms White Paper, international data bases for characterizing TRISO fuel performance and radionuclide transport in HTGRs have been developed over the past five decades. These databases have been summarized in several documents that are publicly available via the internet (e.g., from IAEA and EPRI). Although it is now more than a decade old, IAEA TECDOC-978, with its extensive bibliography, is still the best single document [1]. More recent radionuclide transport data have been published in numerous papers presented at the biennial international HTR conferences [2] and in peer-reviewed technical journals [3].

Many approaches have been used to characterize radionuclide transport in HTGRs that range from laboratory measurements with surrogate materials, to reactor surveillance programs at the seven HTGRs that have been built and operated, to atomistic modeling on supercomputers in recent years. While the approaches have been diverse, the transport models and material property correlations used to predict radionuclide transport in support of reactor design and safety analysis are, in general, based upon experimental data that have been correlated with phenomenological models based on first principles, rather than simply n^{th}-order polynomial fits of the data. Often, correction factors are added to the first principles model to account for irradiation effects. The technical bases for the models and material property correlations for predicting radionuclide transport in prismatic HTGRs in the 1990s are summarized in [4].

While the international data bases for characterizing radionuclide transport phenomena in HTGRs are extensive and often robust, there are nevertheless gaps that will be addressed by the NGNP/AGR fuel technology program. For example, there are considerable data on fission product release from UC_2 and UO_2 kernels, but there are relatively few data for the reference LEU UCO kernels. Likewise, there are considerable data on the transport of volatile fission metals (Ag, Cs, Sr) in German sphere matrix materials and in earlier grades of nuclear graphite, such as H-451, but there are presently no data for the new graphites being developed by the NGNP Project (e.g., PCEA). The planned AGR-3/4 test, which is scheduled to begin irradiation in the Fall of 2011, is designed to address these data needs, and the refined radionuclide transport models and correlations derived from these AGR-3/4 data will be independently validated by the planned AGR-8 test.

References:

1. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," TECDOC-978, International Atomic Energy Agency, November 1997.

2. van der Merwe, J.J., "A Method to Evaluate Fission Gas Release During Irradiation Testing of Spherical Fuel," Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology, September 28–October 1, 2008, Washington, DC, HTR2008-58184.

3. van der Merwe, J.J., "Evaluation of Silver Transport through SiC During the German HTR Fuel Program.," Journal of Nuclear Materials, Volume 395 (2009), pp 99-111, December 2009.

4. Martin, R.C., "Compilation of Fuel Performance and Fission Product Transport Models and Database for MHTGR Design," ORNL/NPR-91/6, Oak Ridge National Laboratory, October 1993.

RAI MST-73: The values shown in Table 4-3 of the Mechanistic Source Terms White Paper for heavy metal contamination are very similar for pebble fuel and block fuel. However, the SiC defect fraction for block fuel is shown to be about an order of magnitude higher than heavy metal contamination. This would suggest that the PBR fuel particles are manufactured to a higher quality standard than the block fuel. If true, discuss why this is acceptable. If not true, discuss why the pebble fuel does not require a separate SiC defect fraction specification, especially with regard to accurately predicting alkali metal and alkaline earth releases from the fuel vs. fission gas release from the fuel.

Response MST-73:

Table 4-3 of the Mechanistic Source Terms White Paper is reproduced here as Table 1 to facilitate the response:

Table 1. Typical fuel quality and performance values

Parameter	Allowable Core Average Fraction (≥95% Confidence) Prismatic	Allowable Core Average Fraction (≥95% Confidence) Pebble Bed
Fuel Quality*		
Heavy Metal Contamination	$\sim 2\times10^{-5}$	$\sim 6\times10^{-5}$
SiC Defect Fraction	$\sim 1\times10^{-4}$	—
In-service Fuel Failure during Normal Operation	$\sim 2\times10^{-4}$	$\sim 5\times10^{-5}$
In-service Fuel Failure during Accident Operation	~6×10-4	$\sim 5\times10^{-4}$
* Fuel quality for a pebble bed HTGR is defined as the sum of heavy metal contamination and SiC defects. In the prismatic HTGR these specifications are determined separately.		

As stated in the note below the table, the fuel quality specification for the pebble bed, 6×10^{-5}, is the sum of the heavy metal contamination and SiC defect fractions, and the equivalent sum for the prismatic fuel would be 1.2×10^{-4}, a factor of 2 higher than the value for pebble fuel. These values evolved in different technology and design programs in Germany and the U.S. based on different reactor and fuel designs. The specified values are based on a combination of fuel fabrication experience, in terms of what can be practically achieved, and safety analysis assumptions, in terms of what is needed by the reactor design. While there were substantial technology exchange activities between the two programs, it is reasonable to expect that fuel specifications and associated characterization procedures would not be identical.

The fuel quality parameters in the U.S. are determined by two separate characterization procedures. Historically, the heavy metal contamination value, which includes exposed heavy metal from both tramp material in the matrix and bare kernels or particles with connected cracks through all the layers, was determined by measuring noble gas release/birth rate from compacts irradiated in a TRIGA reactor. In the NGNP program, heavy metal contamination is determined by deconsolidating and leaching sample fuel compacts prior to oxidation. The deconsolidated particles are then subjected to sustained oxidation at elevated temperature to remove all exposed pyrocarbon. The particles are then leached to determine the

SiC defect fraction. The heavy metal contamination fraction and the SiC defact fraction are among the parameters used to predict fission product release from the fuel.

In the single characterization procedure used by the German program, sample fuel spheres are oxidized to the SiC layer and then leached. This procedure results in a value which includes both heavy metal that was initially leachable and heavy metal exposed by the oxidation. The resulting value, termed a "free uranium fraction", is among the parameters used to predict fission product release from the fuel. The heavy metal contamination fraction and the SiC defect fraction are among the parameters used to predict fission product release from the fuel. This is a conservative assumption, as it would over-predict fission product releases resulting from manufacturing defects.

Both the German and U.S. TRISO particle fuel development programs have been directed toward producing the highest quality fuel practically achievable, and specifying allowable fuel quality meeting safety analysis requirements while providing margin to expected fuel quality. Results to date indicate that the NGNP fuel development and qualification program will establish a fuel quality level (heavy metal contamination fraction and SiC defect fraction) as good as or better than that of the German program.

RAI MST-74: Discuss whether a natural circulation driving force could develop during the core heat-up phase if air infiltrates the primary system due to a break in the HPB.

Comments: Page 32 of the white paper states: "Once the blowdown is complete, there is relatively little driving force from the primary circuit to the reactor-building to affect significant releases during the later core heat-up phase of the event."

Response MST-74:

There are three phases during a depressurized conduction cooldown, the length and other characteristics of which depend on the leak size and location: blowdown, expansion, and contraction.

During the blowdown phase of a depressurized conduction cooldown event, natural circulation within the core and between the upper and lower plenums can occur to augment the conduction and radiation heat transfer within the core. As the pressure between the primary system and reactor building equilibrates, the lower density helium may, depending on the leak size and location, exchange with the higher density helium-air mixture in the reactor building. However, analyses have shown that the helium-air mixture is only a few percent air for a spectrum of leak sizes and locations. These analyses were conducted for the PBMR reactor building for the South African Demonstration Plant, but the conclusion is expected to be the same for other vented reactor building designs in which a large portion of the air gets displaced through the reactor building vent during primary system depressurization. This expectation will be confirmed as the design of the NGNP progresses.

As the core and helium within the primary system heat up, thermal expansion will cause additional displacement from the primary system to the reactor building. As the core and helium within the primary system cool down after peak core temperatures have been reached, thermal contraction of the gases within the primary system will cause the helium-air mixture within the reactor building to be drawn into the primary system.

Natural circulation during the core heatup and cooldown is established within the core between the upper and lower plenums. Local natural circulation patterns will also be established between hot and cold surfaces (e.g., between the core barrel and reactor vessel). This natural circulation exists regardless of the ingress of any air in the helium-air mixture into the primary system. The amount of natural circulation that occurs is minimal when the primary system is depressurized, again regardless of the ingress of any air

in the helium-air mixture. It is important to understand that the most likely depressurization locations are not where natural circulation patterns are likely to occur. The most likely depressurization locations are penetrations such as instrument lines, helium purification system lines, and pressure relief valve lines. These locations do not directly communicate with the natural circulation flow occurring through the reactor core, where radionuclide release would occur. Therefore, natural circulation is not expected to be a driving force for significant transport of radionuclides between the reactor building and primary system during either the thermal expansion or contraction phases, regardless of the degree of air ingress. Detailed analysis that supports this expectation for a wide variety of leak locations, sizes, reactor building designs, and helium-air mixtures will be performed as the design of the NGNP progresses.

RAI MST-75: For events involving breaks in the helium pressure boundary where core cooling is maintained by active cooling systems rather than passive cooling, discuss why significant circulation through the core is not expected during core heat-up.

Comments: Page 32 of the white paper states: "Once the blowdown is complete, there is relatively little driving force from the primary circuit to the reactor-building to affect significant releases during the later core heat-up phase of the event."

Response MST-75:

For events involving breaks in the helium pressure boundary where core cooling is maintained by active cooling systems, there *will* be significant circulation through the core but *there will be no core heat-up.* Maximum and average core temperatures will decrease immediately after reactor trip and will never rise above the values experienced during normal operation. As a result, radionuclide release will be dominated by the release during the initial blowdown, and the source term will be lower than that associated with a depressurized loss of forced cooling. As the core and helium within the primary system cool down due to forced convection cooling, thermal contraction will cause the helium-air mixture within the reactor building to be drawn into the primary system. Pressure will equilibrate at the leak location between the primary system and the reactor building. In addition, the reactor building atmosphere will be predominately helium after the depressurization thus minimizing the density differences between the primary system and reactor building gases.

RAI MST-76: Discuss the extent to which fission product transport delay and holdup mechanisms in the matrix and fuel element graphitic materials are modeled for accident heat-up conditions.

Comments: Page 32 of the white paper states that: "The graphitic materials (matrix and fuel element graphite) do not retain fission gases and iodine, but experience indicates that fission metals like strontium and europium are strongly retained in these materials unless temperatures approach 1800°C."

Figure 4-8 of IAEA TECDOC -978, indicates that significant fractional release of Cs-137 can occur in 200 hours at 1600°C. The release is due to Cs that had been picked in the matrix from cross contamination during irradiation and from failed particles in the element. The data suggests that matrix material does not retain fission metals as well as the PyC layers.

Response MST-76:

As noted in Appendix C of the Mechanistic Source Terms White Paper, the fuel compact and pebble matrix material provide negligible holdup of fission gases. The amorphous carbon content of the matrix material is highly sorptive of metallic fission products, especially strontium and actinides. While the matrix is highly sorptive of metals, it provides little diffusional resistance to the release of fission metals.

The graphite block of a prismatic fuel element, which is denser and has a more ordered structure than matrix material, is somewhat less sorptive of fission metals than the matrix, but it is more effective as a diffusion barrier. The database for nuclear graphite is large in recognition of its effectiveness as a release barrier. Only cesium and silver nuclides effectively migrate through fuel element graphite at normal operating temperatures. The other fission metals, including strontium and the actinides, are completely retained by the graphite during normal operation. A fraction of the strontium is released under core heatup peak temperature conditions, but the actinides are still completely retained.

Fission-metal transport in the matrix materials can be modeled as a transient Fick's Law diffusion process with an effective diffusion coefficient. For pebble bed fuel, the transient diffusion equation for spherical geometry is solved with an evaporative boundary condition.

For prismatic core designs, the accident analysis models historically used in the U.S. HTGR program make the conservative assumption under accident heat-up conditions that fission metal concentrations across the fuel matrix and graphite are uniform, thereby neglecting any diffusional gradients. This approximate treatment is conservative because it neglects any resistance of the graphite to the radionuclide transport and results in a maximum assumed concentration of the radionuclides on the coolant hole surfaces.

At the coolant hole surface, the mass flux from the surface into the flowing coolant is given by the product of a convective mass transfer coefficient and a concentration driving force, which is the difference between the desorption pressure (expressed as a volumetric concentration) and the "free stream" or mixed mean concentration in the coolant. The coolant flow can either be entirely due to natural circulation during the accident heat-up or it can include a blowdown flow during the depressurization phase of an accident.

The equilibrium desorption pressure in the boundary layer is calculated with a sorption isotherm (as described in the discussion of fission product transport across the fuel rod/graphite gap in the response to RAI MST-57) and converted to a volumetric concentration using the ideal gas law. The mixed mean coolant concentration is often conservatively set to zero; alternatively, a two-dimensional model can be used that integrates the total flux into the coolant as it passes through the fuel element or core, thereby providing the coolant concentration at each local point.

For prismatic fuel elements, the mass transfer coefficient is calculated from an empirical correlation for the Sherwood number. In general, the Sherwood number is given as functions of the Reynolds, Schmidt, and Grashof numbers. For spherical pebble bed fuel elements the approach to calculating fission product mass flux from the pebble surface into the flowing coolant is essentially the same, with appropriate adjustments for fuel geometry and for different fuel matrix material with different sorption characteristics.

RAI MST-77: Discuss the relevance of "pressure vessel" particle coating failure probabilities to the NGNP UCO fuel design. Discuss the source of data for developing the models for the different NGNP UCO fuel particle failure mechanism predictions.

Comments: The WP states that: "PISA was developed to calculate fuel coating pressure vessel failure probabilities," and "SURVEY and TRAFIC-FD perform the same calculations for entire HTGR cores under normal operating conditions." The INL staff who are involved in the AGR fuel development and qualification program have stated that the SiC layer of UCO fuel particles are designed and predicted to remain in compression over the design burn-up lifetime of the fuel particles.

Response MST-77:

A traditional failure mechanism addressed in particle fuel performance codes is pressure vessel failure of a one-dimensional spherical particle. The particle is one-dimensional because of perfect symmetry in the tangential and azimuthal directions. Early during irradiation, shrinkage of pyrocarbon layers puts the SiC layer in compression. As irradiation progresses, irradiation-induced creep of the pyrocarbon layers tends to relieve some of this compressive stress. Additionally, buildup of fission gas pressure tends to put the coating layers in tension. If the gas pressure increases enough, the tangential stress in the SiC layer could eventually become tensile. A traditional pressure vessel failure is expected to occur only if the tangential stress reaches a value that exceeds the strength of the SiC for that particle.

NGNP fuel design ensures that traditional pressure vessel failure probabilities remain extremely low. More than sufficient void volume exists within the buffer to accommodate fission gases during the lifetime of the fuel. Additionally, UCO fuel produces essentially no CO gas during irradiation as compared to UO_2 fuel, which tends to dramatically reduce internal gas pressures. These internal gas pressures are such that stresses within the SiC layer remain in compression for most, if not all, of its in-reactor lifetime and remain well below strength levels during postulated accident conditions. Since stresses do not exceed the strength of the SiC layer, failure probabilities remain vanishingly small. PIE results from decades of U.S. and world particle fuel irradiation experiments have indicated only a few particles have ever failed by this mechanism. Results from the recent IAEA CRP-6 normal operation fuel performance code benchmark [1] predict that the SiC remains in compression for both UO_2 and UCO TRISO particle irradiations. This confirms that traditional pressure vessel failure rarely occurs.

Traditional pressure vessel failure would occur only if a particle were to have a very thin buffer so that there would be insufficient void volume to accommodate gas release under normal and off normal conditions. Fuel specifications limit the minimum thickness of the buffer layer so that the probability of pressure vessel failure remains very low.

INL TRISO-coated fuel particle failure predictions are based upon models derived from first principles and are described in the PARFUME theory and model basis report [2]. The failure mechanisms that are modeled are based on the irradiation database obtained over the past 50 years for TRISO fuel. These models are heavily dependent upon material properties used for the fuel. Thermo-mechanical material property correlations used in PARFUME are based upon Reference [3]. Data for physio-chemical failure mechanisms like kernel migration and Pd attack of the SiC are based on the assessment of historical international database an example of which is shown in Reference 4.

References:

1. "Advances in HTGR Fuel Technology," IAEA TECDOC, March 2011.
2. G.K. Miller, D.A. Petti, J.T. Maki and D.L. Knudson, "PARFUME Theory and Model Basis Report", INL/EXT-08-14497, September 2009.
3. CEGA Corporation, "Material Models of Pyrocarbon and Pyrolytic Silicon Carbide", CEGA-002820 Rev. 1, July 1993.
4. J. T. Maki, D. A. Petti, D. L. Knudson and G. K. Miller, "The challenges associated high burnup, high temperature, and accelerated irradiation for TRISO-coated particle fuel," *Journal of Nuclear Materials*, 371 (2007) 270-280.

RAI MST-78: Since the German UO_2 test data used for developing TRISO-coated particle diffusion rates are largely based on post-irradiation heating tests as will be the data used to develop the diffusion rates for AGR UCO fuel, discuss why additional irradiation testing for UO_2 particle fuel and UCO particle fuel are not considered mechanistic source term knowledge gaps.

Comments: Table 5-1 does not include any gaps in fuel fission product release from intact and failed fuel particles during normal operations. Yet in TEV-1022, INL states: "To accurately model fission product transport in TRISO-coated particle fuel under high temperature irradiation, use of 'effective' diffusion coefficients for the kernel and coatings (as presented in the IAEA TECDOC 978) obtained from post-irradiation heating tests is not recommended because those coefficients do not consider the irradiation effects, either implicitly or explicitly."

Response MST-78:

The omission of additional irradiation testing of coated particle fuel (of any kernel composition) from Table 5-1 of the Mechanistic Source Terms White Paper is primarily a reflection of the way in which the scope of the Fuel Qualification and Mechanistic Source Terms White Paper s was divided.

When the FQ and Mechanistic Source Terms White Paper s were being written, it was recognized that the two papers are closely related, and considerable discussion was held regarding how to draw a line between the scopes of the papers. It was decided that the scope of the FQ White Paper would include coated particle fuel performance up to the point of fuel particle failure, while the scope of the Mechanistic Source Terms White Paper would include fission product transport behavior in the core, primary circuit, and reactor-building following fuel particle failure. The Mechanistic Source Terms White Paper also considers behavior of fission products in the fuel particle itself to the extent that this affects fission product release from either intact or failed particles.

Because additional irradiation testing of coated particle fuel was being thought of during preparation of the white papers as primarily a fuel qualification activity, it was not included in Table 5-1, which is a summary of major mechanistic source term knowledge gaps related to fission product transport. Additional irradiation testing to enhance understanding of fission product behavior in both intact and failed fuel particles and to support development of mechanistic source terms is, however, recognized as an important activity and is discussed elsewhere in Section 5 of the Mechanistic Source Terms White Paper.

Two irradiations devoted to fission product transport measurements are described in Table 5-2 of the Mechanistic Source Terms White Paper. Test AGR-3/4 is designed for the measurement of fission product transport from designed-to-fail (DTF) fuel particles (with intentionally missing buffer coatings and with a thin PyC seal coat). Measurements of fission gas release (R/B) will be made on line, and measurements of metallic fission product effective diffusivities through compact matrix and fuel element graphite will be made under both normal operating conditions (post irradiation examination) and accident conditions (safety testing). The results of these measurements will be used for fission product transport model development. Irradiation test AGR-8 will use DTF fuel particles, and fission product release and transport results under both normal operation and accident conditions will be used to validate fission product transport codes.

In addition to the two irradiations devoted to fission product transport within the fuel element, irradiation tests AGR-1, AGR-2, AGR-5/6, and AGR-7 will produce results on fission product transport under both normal operation and accident conditions. Mass balance measurements of radioisotopes within the irradiation capsules (capsule metallic surfaces, graphite fuel holders, fuel compacts, and fuel particles)

will reveal whether metallic fission products have been released from fuel particles under normal irradiation conditions and, if so, where they have been deposited.

RAI MST-79: Figure 4-1 identifies the Cleanup in the Helium Purification System and the Reactor Building HVAC System as important decontamination elements in the mechanistic source term calculations. However, Table 5-1 does not include these decontamination elements as source term knowledge gaps. Discuss the sources and the applicability of the data that is to be used to model the performance of these systems in the mechanistic source term analysis.

Response MST-79:

The data to be used to model the performance of the Helium Purification System (HPS) in the mechanistic source term analysis is derived primarily from the operation of the HPS in seven prior or current HTGRs (Dragon, Peach Bottom Unit 1, Fort St. Vrain, AVR, THTR, HTR-10, and HTTR) as well as from operation of the HPS for in-pile loops. This experience has been obtained over the last 50 years of HTGR technology development. The design and operation of the HPS for the modular HTGR is essentially the same as that of FSV, except for the use of improved materials in the oxidizer beds for hydrogen and tritium removal. HPS technology is well established and hence was not included in Table 5-1 as a major knowledge gap.

Similarly, reactor-building HVAC system technology is well developed, and modeling of its performance in mechanistic source term analysis is expected to be relatively straightforward. Although behavior of radionuclides in the reactor-building is included in Table 5-1 as a major knowledge gap, this uncertainty primarily affects the calculation of relative radionuclide content in the reactor-building atmosphere and on reactor-building surfaces. While concentrations of radionuclides in the reactor-building atmosphere are uncertain as a result of the knowledge gap acknowledged in Table 5-1, the performance of the reactor-building HVAC system for a range of concentrations is relatively well known. Hence, performance of the reactor-building HVAC system in and of itself was not included in Table 5-1 as a major knowledge gap.

RAI MST-80: Discuss the plans, if any, to conduct phenomena identification and ranking tabulations (PIRTs) for the water-ingress and dust elements of the NGNP mechanistic source term calculation. If PIRTs are not planned, describe how the phenomena important to these conditions will be identified and assessed.

Comments: PIRTs have not been conducted to identify the important phenomena and gaps to model the effects of water ingress on the mechanistic source terms, nor have PIRTs been conducted to identify the important phenomena and gaps to model the effects of graphite dust on the mechanistic source terms.

Response MST-80:

Water Ingress

The NGNP Project conducted an assessment of the effects of moisture ingress on the HTGR on February 16-17, 2011. The assessment was similar to a formal PIRT and is being referred to as a "Conceptual PIRT". The assessment team consisted of personnel from ORNL, INL, and various universities and subcontractors, including one of the reactor suppliers. Three staff members from the NRC attended as observers under the auspices of the DOE-NRC Memorandum of Understanding. Given the current plans for having a steam generator in the NGNP primary loop, the potential for moisture ingress is deemed to be much higher than was considered during previous phenomena identification and ranking tables (PIRTs) earlier in the NGNP Project. The reference design for this assessment was the General Atomics

modular high temperature gas-cooled reactor (MHTGR) design that was submitted to the NRC in the late 1980s. This design was selected because this concept has the most information regarding the physical configuration and system performance for a steam cycle facility at this time. Results of the assessment were documented in the report, "*Assessment of NGNP Moisture Ingress Events*" (INL/EXT-11-21397), which was transmitted to DOE on April 25, 2011, and has also been provided to the NRC staff. The major phenomena and issues of concern that were identified, categorized, and generally agreed upon as being of high importance and requiring more attention were:

- Characterization of graphite properties and performance under both short and long-term exposure to moisture
- Investigation into the importance of the plate-out and resuspension of radionuclides in the primary cooling system
- Development of a systems accident code capable of simulating phenomena associated with moisture ingress
- Additional scoping analysis to further identify phenomena and sequences that are important to the plant performance.

Dust

Dust effects have been the subject of ongoing interaction within a working group between experts from the NRC and NGNP Project staffs, including expert representatives from INL, HTGR vendors, universities, and various international organizations. Workshops on dust phenomena in the HTGR were conducted in November 2009, and in March 2010. A Dust Issues Assessment meeting was held from March 14-16, 2011. Each of these meetings was attended by staff from the NRC and the NGNP Project under the auspices of the DOE-NRC Memorandum of Understanding.

A document that describes potential modular HTGR dust safety issues and research and development needs was drafted in advance of the March 2011 assessment and was revised based upon the discussions of that meeting [1]. The document supplements the AGR Fuel Qualification and NGNP Methods Technical Development Plans and will also be used in the development of Calls for Proposals for DOE-funded university research. It is expected that these ongoing interactions will lead to a consensus regarding which dust phenomena are important to each HTGR design variant.

Although the comments that accompany this RAI note that there have not been PIRTs conducted to identify the important phenomena and gaps to model the effects of graphite dust on the mechanistic source terms, Reference [2] does contain the results of PIRT evaluations of the effects of dust on fission product transport. The document that was prepared based on the working group meetings [1]addresses the results of those PIRTs.

Reference:

1. INL/EXT-11-21097, "HTGR Dust Safety Issues and Needs for Research and Development", June 27, 2011
2. USNRC, "*Next Generation Nuclear Plant Phenomena Identification and Ranking Tables (PIRTs) Volume 3: Fission-Product Transport and Dose PIRTs*", NUREG/CR-6944, March, 2008.

RAI MST-81: Discuss the plans, if any, to conduct fuel testing to model the effects of potential air ingress on the NGNP MST calculation.

Comments: Section 5.2 states that: "The NGNP/AGR Program recognizes that additional tests to characterize the effects of water ingress on fuel performance and fission product transport will need to be added to the program." In addition, a footnote to this sentence states that: "Future white papers on LBEs, including air and water ingress, are planned." However, there is no mention in Figure 4-1 of fuel testing to develop data to model the effects of potential air ingress.

Response MST-81:

This RAI is very similar to RAI FQ-34/MST-38. Please see the response to RAI FQ-34/MST-38, which summarizes the plans for air ingress testing.

RAI MST-82: Discuss whether and/or how the mechanistic source term definition (and associated phenomena) would change should the Commission decide that: (1) a traditional low leakage containment building will be required or, (2) a reactor-building structure with an as yet to be defined fission product retention functional capability will be required (e.g., high efficiency filters in the vent path.

Comments: The proposed HTGR source term definition and associated accident condition phenomena reflect the proposed use of a vented low pressure confinement building. However, the Commission has not yet made a policy decision on HTGR containment functional performance requirements. HTGR containment functional performance requirements is also considered a defense-in-depth licensing policy issue.

Response MST-82:

The *definition* of the source term presented in the Mechanistic Source Terms (MST) White Paper would not be affected by either of the posited Commission decisions. The Mechanistic Source Terms White Paper presents the following definition of the source term (page 1, footnote b):

Source term – Refers to the quantities, timing, physical and chemical forms, and thermal energy of radionuclides ..... released from the reactor-building to the environment during postulated accidents.

The source term is referred to as "mechanistic" because it is calculated with a mechanistic approach that takes into account the inherent characteristics of the HTGR technology that provide multiple barriers to fission product transport to the environment using models that use first principle methods supported as needed by empirical confirmation to represent the mechanisms (phenomena) that affect the generation and transport of radionuclides in the HTGR facility. The definition of the source term is also discussed in the response to RAI MST-84.

Although the *definition* of the mechanistic source term would not be affected by these posited decisions, the manner in which the effects of the reactor-building as the final barrier to radionuclide release are mechanistically modeled and the resulting quantitative offsite dose consequences, i.e., the "*associated phenomena*" referred to in this RAI, would be affected by the design changes to the reactor-building that would be needed to comply with the posited decisions. The risk to the public would have to be reassessed with the change to the design, not only including the impact on the consequences of the Licensing Basis Events (LBEs) associated with the current conceptual design, but also including any new LBEs involving failure of the design modifications that would result from the Commission decisions. Impacts on operations, maintenance, availability, investment protection, and economic viability would also need to be assessed.

Studies on this topic for the MHTGR and for the NGNP PBMR Process Heat Plant have both found that a vented reactor-building is the best choice for the modular HTGR for this spectrum of considerations. Most importantly, a vented reactor-building is superior to an LWR-type, pressure retaining, low leakage containment for public safety. In primary system depressurization events, venting of the initial blowdown with minimal radionuclide concentrations limits the driving force for release of additional radionuclides during the subsequent heat-up phase. This also releases the pressure that would build up in a low leakage reactor-building by release of the non-condensable helium. Studies described in more detail below have shown that hold up of the initial blowdown has the potential to result in a higher total radionuclide release during these events than that released for the vented reactor-building design.

In traditional LWRs, the reactor-building design is chosen to retain radionuclides inside the building – the containment function of the reactor building. Additionally, the reactor-building must function to protect SSC inside the building from external hazards and threats.

Many of the technical bases for choosing a modular HTGR reactor-building design do not differ from those for current LWR technology (e.g., structural protection of the reactor from external events). However, since the modular HTGR and LWR technologies are fundamentally different in their approach to design for retention of radionuclides, these differences lead to differences in the design of the reactor building.

The modular HTGR reactor-building design considers a complete range of mechanistic event sequences and the associated response of the radionuclide barriers. The reactor-building design reflects the fundamental characteristics of the fuel as the primary barrier to radionuclide release, and the role of the coolant, the moderator, and the other functional containment barriers to radionuclide release. In modular HTGRs the high temperature capability of the reactor core, with its highly retentive coated fuel particles, and the configuration of the core allow the reactor to be cooled with the chemically inert helium coolant, pressurized or depressurized and with or without active cooling systems.

Consistent with this safety design approach, past studies dating back to the modular HTGR in the 1980s have confirmed the efficacy of venting the reactor-building during the blowdown. These studies also identified the need for the reactor-building to structurally perform the safety functions of maintaining the geometry of the reactor vessel internals and removing heat from the reactor vessel to the reactor cavity cooling system (RCCS). The reactor-building also needs to support functions of control of core heat generation, removal of core heat, and control of chemical attack. Additionally, while not required during Design Basis Events, the reactor-building performs a radionuclide retention function that provides additional margin for meeting the offsite dose requirements. Note that although some of the modular HTGR reactor-building safety functions correspond with some of the safety functions for the LWR reactor building, the primary focus for the modular HTGR reactor-building is on structural protection rather than radionuclide retention.

MHTGR Reactor Building Study

Because of the traditional emphasis on radionuclide retention in LWRs, the implications of imposing the LWR pressure-retaining reactor-building design on the modular HTGR has been examined over the years. The MHTGR reactor-building study conducted in 1989 and summarized in this RAI response considered the following design alternatives:

1. Vented, 100% per day leak rate (reference design)
2. Vented, filtered, 100% per day leak rate

3. Vented, filtered, 5% per day leak rate
4. Unvented, moderate pressure, 5% per day leak rate
   a. Air-cooled RCCS
   b. Water-cooled RCCS
5. Unvented, high pressure
   a. 5% per day leak rate
   b. 1% per day leak rate

Advantages and disadvantages of these alternatives are summarized in Table 1. The MHTGR 1989 study concluded that the reference baseline design (Alternative 1) would meet all safety requirements with large margins and was the only option that was likely to meet economic goals.

Table 1. Summary of Advantages and Disadvantages for Alternative Reactor Building Concepts

Reactor Building Alternative	Alternative Advantages	Alternative Disadvantages
1	- Provides acceptable level of public safety - Maximizes reliance on simple, passive features - Least complex design	- Depends on successful completion of technology programs for fuel fabrication, fuel performance, fission product behavior, physics, graphite, thermal performance, metals, and major components
2	- Provides acceptable level of public safety	- Increases cost, may not meet economic goal - Increases radionuclide containment system complexity - Depends on successful completion of essentially the same technology development program as for Alternative 1
3	- Provides acceptable level of public safety	- Adds substantial cost, not likely to meet economic goal - Increases radionuclide containment system complexity
4	- Provides acceptable level of public safety, including mitigation of hypothetical fuel failure accidents, in the event the passive safety design assumptions are not validated - Reduces reliance on technology development program	- Prohibitive cost, reduced plant availability, does not meet economic goal - Substantial increase in complexity of radionuclide containment system requires active systems - Decreases reliability of long term heat removal - Reduces module interdependence - Introduces enhanced radionuclide transport mechanisms for low frequency events for

Reactor Building Alternative	Alternative Advantages	Alternative Disadvantages
		Alternative 4a
5	- Same as 4	- Prohibitive cost, reduced plant availability, does not meet economic goal - Substantial increase in complexity of radionuclide containment system requires active systems - Decreases reliability of long term heat removal - Introduces enhanced radionuclide transport mechanisms for low frequency events for Alternative 4a

In response to an NRC request to evaluate alternative source terms for the MHTGR, in 1993 reactor-building alternatives were re-evaluated with respect to their impacts on radiological consequences during accidents. Specifically, the NRC requested that source terms be evaluated that considered

- lower quality fuel with higher defect fractions,
- very rapid hydrolysis of defective fuel during water-ingress events,
- "weak" fuel, i.e., fuel that performs as expected during normal operation but fails at higher rates than expected during accidents, and
- higher than expected release of plateout activity during accidents.

For these pessimistic assumptions, the vented, filtered alternatives offered the potential for consequence mitigation.

NGNP PBMR Reactor Building Evaluation

More recently, an NGNP study, summarized in this RAI response, examined a range of reactor-building options for the NGNP PBMR Process Heat Plant. A summary of the alternative concepts that were studied is provided in Table 2. The study examined two alternatives for a vented and unfiltered building, three alternatives for a filtered and vented building, and two alternatives for a leak-tight or pressure-retaining reactor building.

To gain insight into alternative strategies for implementing the safety functions assigned to the NGNP reactor building, a set of alternative design concepts were developed and evaluated. The evaluations were supported by scoping calculations, including an assessment of the capabilities of each alternative to mitigate the releases from a broad spectrum of LBEs involving a Depressurized Loss of Forced Cooling (DLOFC) condition.

A comparison of the offsite doses associated with the alternatives from this study is shown in Figure 1. The event scenario evaluated is an assumed leak or break in the helium pressure boundary (HPB) followed by the loss of the two independent forced convection cooling systems, the heat transport system (HTS) and the shutdown cooling system (SCS). This event is alternately known as a Depressurized

Conduction Cooldown (DCC) or a Depressurized Loss of Forced Cooling (DLOFC). There are two components to the radionuclide release:

1. an early release of the radionuclides in the helium from normal operation and a fraction of the plateout that extends for minutes to hours depending on the size of the leak or break and
2. a delayed radionuclide release due to the core heat up that begins hours to days into the event.

These alternatives were assumed to have the same Reactor Building vented volume, approximately 10,000 m^3. A range of HPB break sizes from 2 mm to 1,000 mm effective diameter was evaluated. As seen in Figure 1, the filtered vented reactor-building Alternative 3a provides the most effective radionuclide retention capability among the options considered, including the pressure retaining Alternative 4a. When alternatives with larger vented volumes are included, Alternative 3b, which is also a vented filtered configuration, was found to provide superior retention capability among all the alternatives considered. Alternative 4a performs more poorly than 3a or 3b because that option does not remove the pressure driving force that exists following HPB depressurization. Alternatives 3a and 3b on the other hand, despite having much higher leak rate than Alternative 4a or 4b, manage the pressure relief function and eliminate the pressure driving force for the delayed fuel release after the time of depressurization, while using filtration to mitigate the blow-down release and the part of the release from the fuel that occurs prior to depressurization.

Table 2. Alternative Reactor Building Strategies for Performing Safety Functions

No.	Design Description	Vented Area Leak Rate Vol% /day	Pressure Relief Design Features	Post blow-down re-closure of PRS shaft?	Radionuclide Filtration	
					Blow-down phase	Delayed fuel release phase
1a.	Unfiltered and vented	50-100	Open vent	No	None	Passive
1b	Unfiltered and vented with rupture panels	50-100	Internal + External rupture panels	No	None	Passive
2	Partially filtered and vented with rupture panels	25-50	Internal + External rupture panels	Yes	None	Active HVAC
3a	Filtered and vented with rupture panels	25-50	Internal + External rupture panels	Yes	Passive	Active HVAC
3b	Filtered and vented with rupture panels and expansion volume	25-50	Internal + External rupture panels + expansion volume	Yes	Passive	Active HVAC
4a	Pressure retaining with internal rupture panels	0.1-1	Internal rupture panels	N/A	Passive	Passive
4b	Pressure retaining with internal rupture panels and expansion volume	0.1-1	Internal rupture panels + expansion volume	N/A	Passive	Passive

As seen in Figure 1, all of the evaluated alternatives meet the EPA Protective Action Guideline limit and the dose limits for DBA from 10 CFR Part 50.34 for design basis breaks up to 100 mm effective diameter as well as the beyond design basis breaks from 100 mm to 1,000 mm effective diameter by large margins based on these scoping calculations. The break size of 1,000 mm corresponds to the equivalent single ended break size for a double ended guillotine break of the largest pipe in the HPB, which has an internal diameter of 710 mm for the PBMR NGNP design.

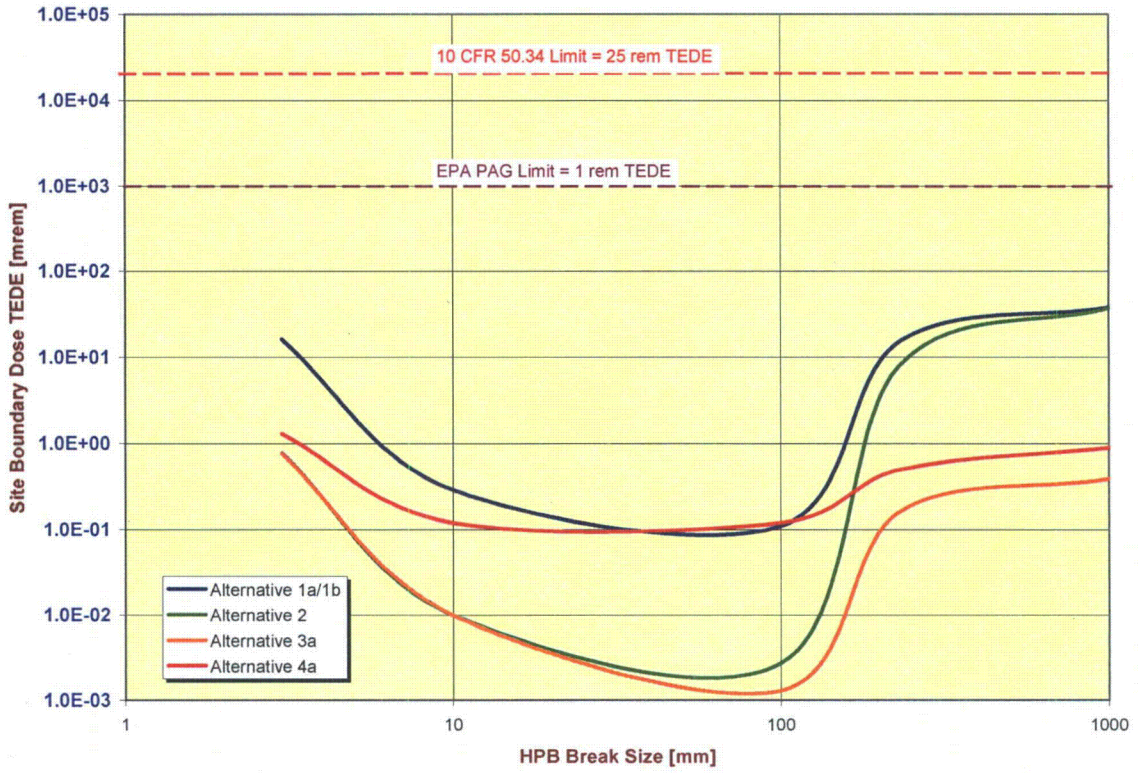

Figure 1. Site Boundary Dose (TEDE) vs. HPB Break Size (Effective Diameter) for LBEs Involving DLOFC for Alternative Reactor Building Concepts with Comparable Volumes

In general, the vented options that were considered (1a, 1b, 2, 3a, and 3b) were found to be superior to the pressure retaining options (4a and 4b) based on the following considerations:

- Greater compatibility with a non-condensable and inert primary coolant.
- Venting of the primary coolant inventory to atmosphere with or without filtration eliminates a driving force for subsequent fission product transport of the delayed fuel release source term.
- When used with filtration (2, 3a, and 3b) provides more effective retention of radio-nuclides for the design basis event spectrum up to a break size of 100 mm effective diameter. Alternatives 3a and 3b provide superior retention for BDBE break sizes up to 1,000 mm effective diameter as well.
- Lower capital and operating costs.

- Easier and less costly to engineer interfaces with RCCS and other systems.

The highest overall rating for alternatives examined was Alternative 2 (partially filtered and vented with rupture panels) followed by Alternative 3a (fully filtered and vented with rupture panels).

- Both Alternatives 2 and 3a provide superior radionuclide retention capability for design basis HPB breaks with DLOFC relative to the pressure retaining Alternatives 4a and 4b.
- Alternative 3b closely followed by Alternative 3a is superior to all evaluated alternatives across the entire HPB break spectrum including AOOs, DBEs, and BDBEs.
- Alternatives 2, 3a, and 3b are expected to have greater licensability than either of the open vented options (1a and 1b) due to their superior capability to mitigate releases and air ingress (lower building leak rates needed to accommodate filters).
- Another alternative for future study is a vented building with a passive re-closable damper without a filter. This is expected to have delayed release retention capabilities approaching that of Alternative 2 due to the capability to achieve a lower leak rate.
- Results of the radionuclide retention study show that all the evaluated alternatives provide sufficient margins to offsite dose limits based on inherent and passive safety characteristics of the NGNP.
- Added engineered features such as filters and re-closable dampers add additional margins.
- This study confirmed that radiological retention is not a required safety function but rather a supportive safety function for the NGNP reactor-building according to how these terms are defined in the NGNP risk-informed and performance-based licensing approach.
- It is noted that the required safety functions of the reactor-building that involve the structural protection of the reactor and its inherent and passive safety characteristics serve to maintain the fundamental safety function of controlling radionuclide releases.

Thus, the above studies concluded that a vented reactor-building meets the safety requirements with margin. The underlying technical motivation for a vented design is to allow the relatively clean helium inventory to be relieved prior to the delayed release from the fuel, thus eliminating the pressure that would enhance transport of radionuclides during the delayed release. The PBMR study quantified the superior performance of the vented designs relative to the traditional LWR low leakage containment buildings as depicted in Figure 2. As shown, Alternative 4a (low leakage containment building), which has the continued leakage of the non-condensable helium with the delayed fuel release source term, is marginally inferior to the vented Alternatives 2 and 3a. However, as shown in the far right column, if the low leakage containment fails after the delayed fuel release, Alternative 4a does not meet requirements. In fact, an alternative with *no reactor-building at all*, shown on the far left of Figure 2, is better than a low leakage containment building that has a delayed failure.

As the NGNP design and technology development efforts proceed, alternatives for the design of the reactor-building will continue to be examined and evaluated to determine the best trade-off among the various design considerations.

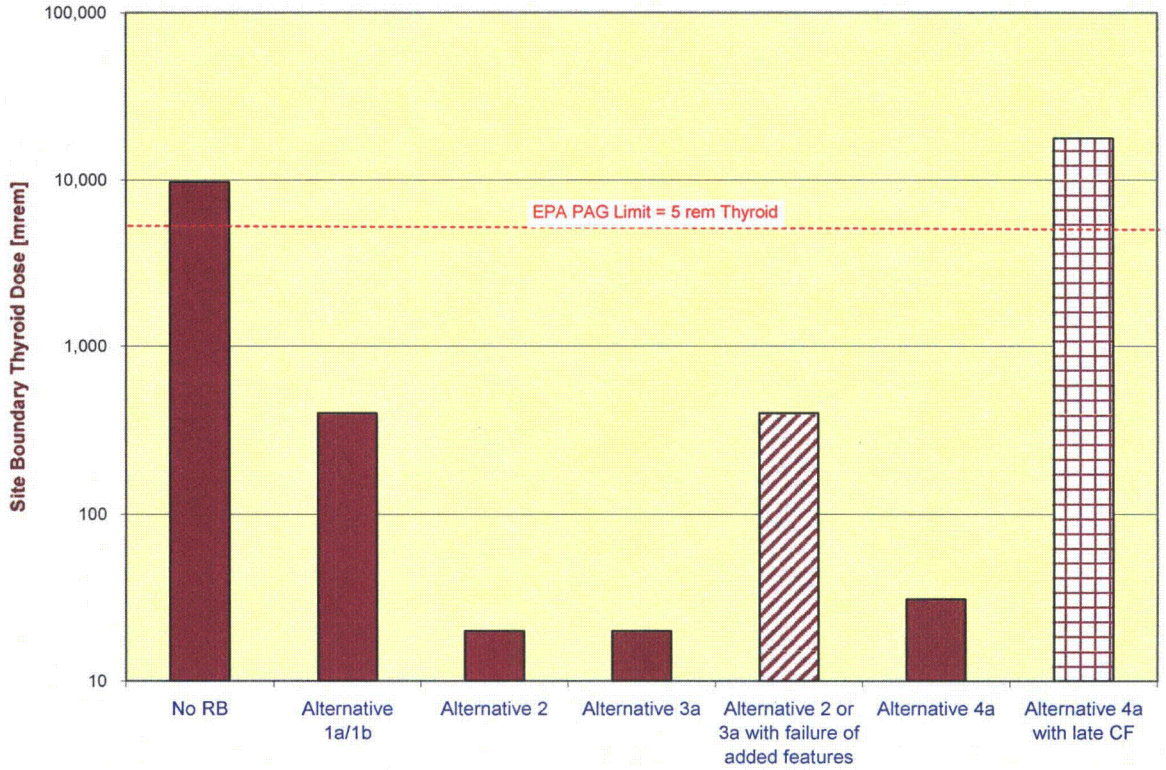

Figure 2. Comparison of PBMR NGNP Reactor Building Alternatives

RAI MST-83: Discuss the approach that will be taken to ensure that NGNP "severe accidents" and the resulting severe accident source terms are bounded.

Comments: The NGNP licensing strategy states: "Given the current state of VHTR technology design, development and experience, and the quality and completeness of the associated NGNP design-specific PRA, Option 2 is the preferred option for licensing the NGNP prototype, which makes primary use of deterministic judgment and analysis complemented by NGNP-specific PRA to establish the licensing basis and requirements." The white paper does not explicitly describe how deterministic judgment complemented by the NGNP PRA will be used to ensure that the events considered result in a set of source terms that bound the NGNP "severe accidents."

Response MST-83:

"Severe accidents" for light water reactors are those that result in significant core damage (significantly larger than expected fuel rod cladding failure and/or loss of cooling geometry and fuel melting). Credible accident scenarios that involve core damage as it is defined for LWRs have not been identified for the HTGR, even for BDBE scenarios with a frequency of occurrence as low as 5×10^{-7} per reactor year. In particular, HTGR fuel, being comprised of all ceramic materials, does not melt and is subject only to insignificant incremental degradation during BDBEs. Hence, HTGR bounding source terms are

calculated for a spectrum of BDBEs that are of comparable likelihood but have different consequences relative to LWR "severe accidents".

The approach taken to select and evaluate BDBEs for the HTGR is described in the Licensing Basis Event (LBE) Selection White Paper [1] and in the forthcoming Probabilistic Risk Assessment White Paper. These white papers include a discussion of how deterministic judgment complemented by the NGNP PRA will be used to ensure that the events considered result in a bounding set of source terms.

The LBE Selection White Paper discusses examination of incredible accident sequences with frequency of occurrence below the current cutoff of 5×10^{-7} per reactor year in the frequency and consequence criteria. The approach to be taken to the plant Probabilistic Risk Assessment will ensure that there are no events with a frequency of occurrence just below the current cutoff that are not taken into account. This will assure the adequacy of the lower cutoff frequency. Events below the 5×10^{-7} per plant-year BDBE region are examined to assure that the residual risk is negligible with respect to the latent mortality safety goal, and to provide general assurance that no potentially high consequence events go unnoticed.

Reference:

1. INL/EXT-10-19521, "Next Generation Nuclear Plant Licensing Basis Event Selection White Paper", September 2010

RAI MST-84: Please provide a clear and concise definition of event-specific mechanistic HTGR source terms.

Comments: The white paper solicits NRC agreement that the definition of event-specific mechanistic source terms for HTGRs is acceptable. A concise definition of "event-specific mechanistic source terms" is not provided. Source term is defined in general (footnote (b) on page 1) as release of radionuclides from the *reactor-building to the environment* – a departure from the traditional LWR source term. Event specificity is implied with regard to calculating the source terms for selected licensing basis events (LBEs). Mechanistic is interpreted in the earlier section of the text (page 2) as an approach that takes into account the multi-barrier DiD concept and distinctly different from the LWR "deterministic" approach. It is not clear if the implication is that the mechanistic approach is not deterministic which, of course, is not true. Later in the text (page 3 and also page 11, section 2.3.1), a better discussion of mechanistic approach is provided in terms of different fission product release and transport phenomena which are modeled presumably in a mechanistic manner supported by data. Also, elsewhere in the text (page 21, section 3.3.2), some elaboration is provided on what is meant by "deterministic source term." Perhaps, a better characterization is "prescriptive source term" as in TID-14484 and NUREG-1465.

The requested definition of event-specific mechanistic source terms should clearly state whether credit is taken for radionuclide retention in the reactor-building and whether the MSTs are to be calculated using deterministic phenomenological models, or applying other (e.g., probabilistic) models. The supporting discussion should clearly state why the source terms should be linked to "release outside the reactor building."

Response MST-84:

The following definitions apply:

- *HTGR Source Term* – Radionuclides released from the reactor-building of a modular HTGR plant to the environment.

- *Mechanistic HTGR Source Term* – A modular HTGR Source Term that is calculated using models that use first principle methods supported as needed by empirical confirmation to represent the mechanisms (phenomena) that affect the generation and transport of radionuclides in the plant.

- *Event Specific Mechanistic HTGR Source Term* – A Mechanistic HTGR Source Term that is calculated for a specific Licensing Basis Event (LBE).

With regard to the comments that accompany this RAI, the following is offered:

The NRC staff correctly notes in its comments in that the source term is defined as radionuclides released from the reactor building. The NRC staff also correctly notes in its comments that, in calculating a mechanistic source term using models that represent the mechanisms (or phenomena) that affect the generation and transport of radionuclides in the HTGR plant, the multi-barrier defense in depth approach is taken into account. It would not be appropriate to focus on only one of the five radionuclide retention barriers (in this case the reactor building) in the mechanistic source term definition as requested by the NRC staff in its comments. However, taking into account the multi-barrier defense in depth approach does include taking credit as appropriate for radionuclide retention in the reactor building. Use of the models is, as noted by the NRC staff in its comments, supported by experimental data. This mechanistic approach results in a source term that is characterized not only by its radionuclide composition (quantitative isotopic content), but is also characterized by its timing, physical and chemical forms, and thermal energy. The NRC staff also correctly notes in its comments that mechanistic source terms are calculated for specific, selected LBEs.

The NRC staff comments also indicate that this RAI response should clearly state why the source terms should be linked to "release outside the reactor building." This definition of the source term is judged by the NGNP Project to be appropriate for a reactor design that relies upon the fuel as the principal barrier to release of radionuclides to the environment and retains radionuclides at their source. Retention of radionuclides at their source enables the HTGR to have offsite dose consequences for LBEs that are very small at distances close to the plant as compared to other reactor types. Emphasis of this important capability for the HTGR focuses on release from the reactor-building to the environment. The definition of the source term used for the HTGR is consistent with that emphasis.

However, as noted in the Mechanistic Source Terms White Paper (page 1), the source terms developed with this approach, and the radionuclide inventories elsewhere in the plant that are determined during source term analysis, can also be used for other purposes, including equipment environmental qualification, control room habitability analyses, and assessments of BDBE risks in environmental impact statements. As discussed with the staff during the NGNP Project's presentation on the Mechanistic Source Terms White Paper on September 2, 2010, the radionuclide inventories elsewhere in the plant that are determined during source term analysis are analogous to other "source terms" typically referred to in LWR licensing documents.

It should also be noted that the definition of the source term as radionuclides released from the reactor-building to the environment is consistent with the definition used by the International Atomic Energy Agency (IAEA). On the IAEA tutorials web site

http://www.iaea.org/ns/tutorials/regcontrol/intro/glossarys.htm?w=1source+term,

the following definition is provided:

Source Term – The amount and isotopic composition of material released (or postulated to be released) from a facility. Used in modeling releases of radionuclides to the environment, particularly in the context of accidents at nuclear installations or releases from radioactive waste in repositories.

Based upon these considerations, the NGNP Project believes that linking the definition of the mechanistic source term to release outside the reactor-building is appropriate.

The NRC staff comments also indicate that this RAI response should state whether the MSTs are to be calculated using deterministic phenomenological models, or applying other (e.g., probabilistic) models. This information is provided in the response to RAI MST-88.

As used in the white paper, "deterministic" was intended to refer to analyses and analysis results containing prescriptive elements (models, assumptions, data, etc.) that may not be based on detailed, mechanistic phenomena. Hence, it was implied, as noted in the NRC staff comments, that deterministic analyses are, at least to some degree, not mechanistic to the extent intended for the calculation of mechanistic HTGR source terms.

RAI MST-85: Indicate where the HTGR supplier-proposed design conditions specified and clarify how they are used in relation to the NGNP design conditions in determining the mechanistic source terms?

Comments: In Section 2.1, the third paragraph makes reference to HTGR supplier-proposed design conditions. It is not clear whether these design conditions are less demanding (e.g., with regard to reactor outlet temperature, operating conditions, etc.) than the NGNP design conditions.

Response MST-85:

The portion of Section 2.1 of the Mechanistic Source Terms White Paper addressed by the RAI discusses the applicability of the approach to calculating mechanistic source terms and states the following:

> *"The general process described in this paper for establishing mechanistic source terms is intended to support design and licensing of the HTGR over a range of operating power levels and reactor outlet helium temperatures that meet as wide a range of potential end-user energy needs as achievable. This approach provides a wide margin for design and licensing at the lower HTGR supplier-proposed design conditions and supports future designs and licensing at higher temperature and power levels."*

The section in question notes that during NGNP preconceptual design, a range of reactor outlet temperatures up to 950°C was proposed by the reactor suppliers. The section then notes that, as a result of evaluations of technical risks and discussions with potential process heat end users, it was later proposed by the reactor suppliers that the NGNP plant be designed for a lower reactor outlet temperature of 700-800°C. These are the "lower HTGR supplier-proposed design conditions" referred to in the quote above. It should be noted that a final decision regarding the reactor outlet temperature for the NGNP has not yet been made.

The point of the white paper section in question is that the source term calculation methodology is applicable over a wide range of reactor outlet temperatures up to 950°C provided that appropriate input data are used for the various materials of construction over the range of reactor outlet temperature. The final selected NGNP reactor outlet temperature will be within that range.

RAI MST-86: What is considered core damage for NGNP? For example, is it kernel migration, breach of IPyC and/or OPyC layers, structural failure of SiC layer, failure of matrix retention capability, or some other definition?

Comments: The white paper indicates on page 11 that there will be no postulated condition of the plant that results in significant fuel particle degradation or any other significant core damage. The use of the term "core damage" is noted. For HTGRs, the term begs a definition. This discussion should clarify that even without "core damage" radionuclides released from HTGR fuel during normal operation will accumulate in the primary system over the reactor's operating lifetime and contribute significantly to release source terms in accidents.

Response MST-86:

"Core damage" in a light water reactor is typically associated with a significantly larger than expected fuel rod cladding failure rate, loss of coolable geometry, and fuel melting with an associated large radionuclide release from the fuel. Accident scenarios that entail significant core damage are the major contributor to large early release frequency in LWR probabilistic risk assessments (PRAs).

As noted in the Mechanistic Source Terms (MST) White Paper, credible accident scenarios that involve core damage as it is defined for LWRs have not been identified for the HTGR, even for LBE scenarios with a frequency of occurrence as low as $5x10^{-7}$ per plant year. In particular, HTGR fuel, being comprised of all ceramic materials, does not melt and is subject only to insignificant incremental degradation even during Beyond Design Basis Events.

Because of the different characteristics and material properties of the modular HTGR core, "core damage" as it is defined for LWRs and the resulting large early release of radionuclides are not meaningful terms for the modular HTGR. This position is consistent with Section 3.2 of SECY-10-0034, where it is noted that core damage frequency and large early release frequency are risk metrics that are not applicable to non-LWR small modular reactors [1].

As explained in the forthcoming Next Generation Nuclear Plant Probabilistic Risk Assessment White Paper, modular HTGR PRAs will be conducted in accordance with the draft ASME standard for non-LWR PRA. In lieu of the use of the term "core damage", the draft standard states that, "requirements in this standard are developed in terms of reactor specific end states called release categories."

Modular HTGR PRAs will not include the calculation of a core damage frequency risk metric, but they will include the frequencies of applicable release categories and their magnitude and timing. Event trees will be developed for the response of structures, systems, and components in the performance of plant-specific safety functions. Core damage and large early release end states will be replaced by modular HTGR-specific end states that include a full set of release categories. Subject to confirmation in these analyses, no modular HTGR release categories are expected, based on previous modular HTGR safety analyses, to constitute a "large early release" as the term is used in an LWR PRA context.

The following additional information is provided in response to the notes and comments that accompany this RAI.

"Core damage" in an HTGR should not be defined relative to any of the measures suggested in the RAI (i.e., kernel migration, breach of IPyC and/or OPyC layers, structural failure of SiC layer, failure of matrix retention capability). All of these measures are calculated parameters that in and of themselves do not correlate directly with modular HTGR release categories.

Consistent with the NRC staff comments that accompany this RAI and with information presented in the MST White Paper, limited coated fuel particle degradation is expected to occur during normal operation and under Licensing Basis Event conditions (including both Design Basis Events and Beyond Design Basis Events). The mechanisms of coated fuel particle degradation are discussed in detail in the Fuel Qualification White Paper. This limited fuel particle degradation will result in the release of some fission products from the core into the primary circuit and the accumulation of radionuclides in the primary circuit during the reactor's operating lifetime.

However, as noted in the MST White Paper the inventory of most radionuclides in the primary circuit will reach equilibrium relatively quickly during reactor operation. To clarify a point made in one of the NRC comments, only the plateout inventories of the longer-lived radionuclide inventories, such as cesium-137, will continue to increase during the reactor's operating lifetime in the manner noted by the NRC staff comments. The NRC staff comments state that the radionuclides released to the primary circuit during normal operation will contribute "significantly" to the mechanistic source terms. It should be clarified relative to this comment that, as discussed in Section 4.4.2 of the MST White Paper, the contribution to the accident source terms of the release of initial primary circuit circulating inventory and a fraction of the initial plateout inventory, relative to the contribution of the release of a fraction of the radionuclide inventory in the core during core heatup, depends on the specific accident scenario, but it is generally expected to be small.

In all cases, however, the total magnitude of the release from the fuel and/or the primary circuit will be insufficient to result in offsite doses that exceed the plume exposure EPA Protective Action Guides at the Exclusion Area Boundary. Hence, the limited fuel particle degradation during normal operation and under accident conditions and its contribution to the mechanistic source terms is not "significant", and these conditions would not constitute "core damage" as it is understood in LWR safety and risk analyses.

Reference:

1. SECY-10-0034, "Potential Policy, Licensing, and Key Technical Issues for Small Modular Nuclear Reactor Designs," U.S. Nuclear Regulatory Commission, March 28, 2010

RAI MST-87: Please explain Figure 2-5 more clearly in terms of examples of how it would relate to a proposed mechanistic source term calculation.

Response MST-87:

Standard design practice in the U.S. HTGR program has been to define a two-tier set of radionuclide design criteria—referred to as Maximum Expected and Design criteria—for allowable core releases for normal operation and Anticipated Operational Occurrences (AOOs). This practice has been followed since the design of Peach Bottom Unit 1 and Fort St. Vrain up through the design of modular HTGRs. The Design criteria are derived from externally imposed requirements, such as site-boundary dose limits, occupational exposure limits, etc. Once the Design criteria have been derived from the radionuclide control requirements, the corresponding Maximum Expected, criteria are derived by dividing the Design criteria by an uncertainty factor, or design margin, to account for uncertainties in the design methods. This uncertainty factor is typically a factor of four for the release of fission gases from the core and a factor of 10 for the release of fission metals from the core (see response to RAI MST-62 for further discussion of these factors). The fuel and core are to be designed such that there is at least a 50% probability that the fission product release will be less than the Maximum Expected criteria and at least a 95% probability that the release will be less than the Design criteria. The GA approach to implementing such radionuclide design criteria is illustrated in Figure 1, which is a reproduction of Figure 2-5 from the

Mechanistic Source Terms White Paper provided for easy reference. No particular scale is implied in this figure; it is simply a conceptual illustration of the approach.

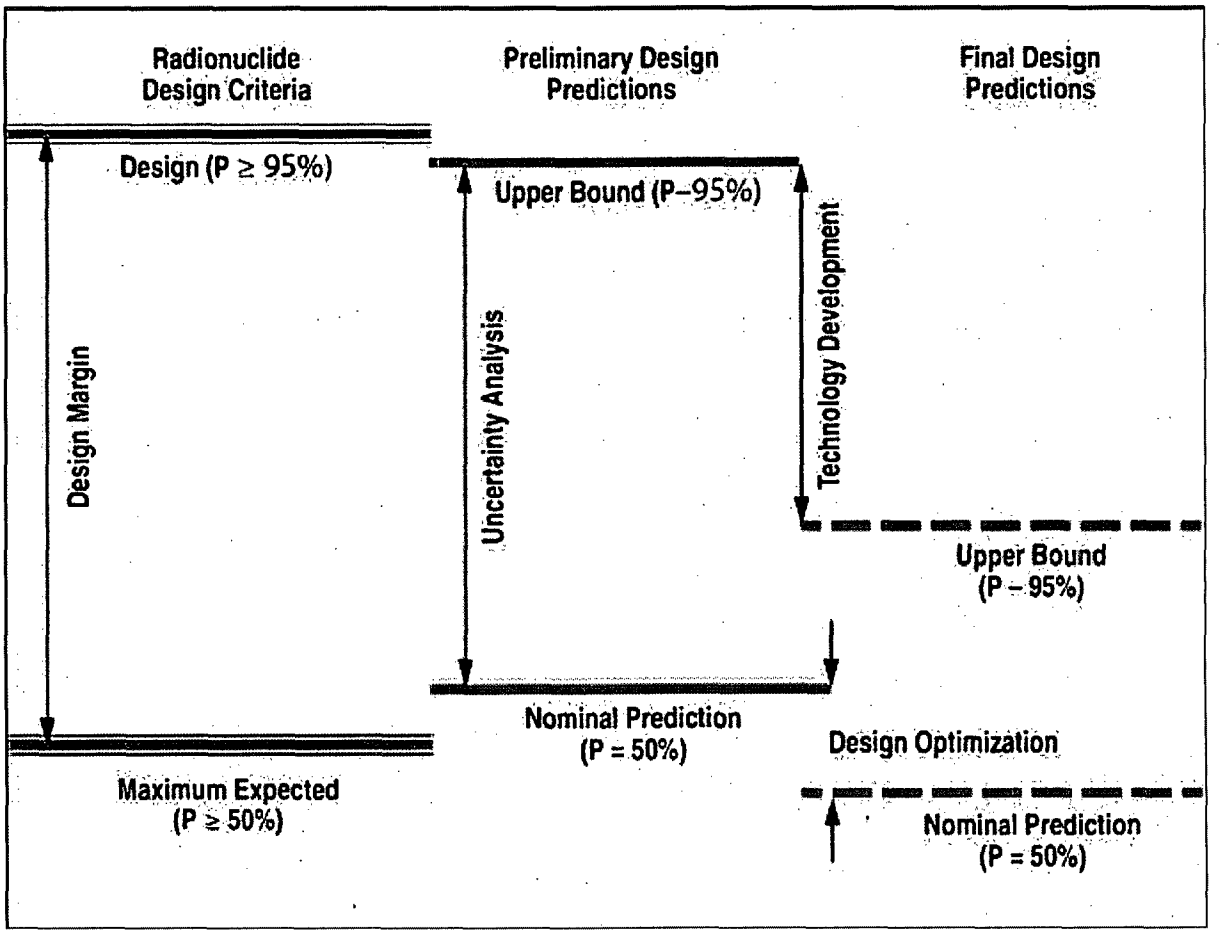

Figure 1. Radionuclide Design Criteria

These radionuclide design criteria have multiple uses in plant design and licensing. First, they provide a figure of merit for judging the acceptability of candidate core designs. Secondly, they allow the plant design and safety analyses to proceed while the core and fuel designs are being optimized as the overall plant design evolves through the conceptual, preliminary, and final phases.

Finally, these radionuclide design criteria provide conservative initial conditions for a spectrum of postulated accidents (e.g., primary coolant leaks) that involve the partial release of circulating and plateout activities from the primary coolant circuit ("conservative" in the sense that they are 60-year end-of-plant life inventories, etc.). The Design activities are used whenever conservative deterministic analyses are required. The Maximum Expected activities are used whenever best estimates are appropriate. Furthermore, since confidence levels are associated with each criteria level and the design margin is specified (e.g., factor of 10 for metal release), a standard deviation can be calculated and used in probabilistic risk assessment (PRA) analysis (assuming a log-normal distribution).

RAI MST-88: How will the uncertainties be treated in the NGNP approach for mechanistic source term calculation?

Comments: The NGNP approach for mechanistic source terms relies on calculation of source terms and dose rates based on the current understanding of generation and transport phenomena. While the approach, in theory, is reasonable, it is not evident how uncertainties will be addressed.

Response MST-88:

The uncertainty evaluation of the of the mechanistic source terms proceeds as follows:

- The detailed calculational tools described in Section 4.5 and in Appendices D and E of the Mechanistic Source Terms (MST) White Paper are used to predict the best estimate, time-dependent mechanistic source term for a given Licensing Basis Event (LBE). These include separate computer codes for calculating the initial radionuclide inventories within the fuel and within the helium pressure boundary and for modeling the off-normal event phenomena as described in the Mechanistic Source Terms White Paper.

- A simplified integrated model is constructed for use in the mechanistic source term and consequence uncertainty evaluation. Best estimate values for the input parameters are utilized in this consequence uncertainty model to predict the mechanistic source terms for comparison, those obtained with the detailed calculational tools in Step 1.

- When there is confidence that the results of the simplified model of Step 2 are within reasonable convergence with the results obtained using the detailed tools, uncertainty distributions are selected for each of the independent input parameters.

- The simplified consequence uncertainty model is then run tens of thousands of times in a Monte Carlo fashion to construct the uncertainty distribution for the mechanistic source terms.

An example of Steps 2 thru 4 of the approach as utilized for the MHTGR in the 1990s for a leak/break in the helium pressure boundary with loss of forced cooling is described in the following paragraph. The basic approach is expected to be used for the NGNP Project, with updates as appropriate to reflect improved understanding of fuel performance and fission product behavior based primarily on the results of the NGNP technology development activities.

The consequence uncertainty model accounts for the release and transport of radionuclides from the fuel barriers, the helium pressure boundary, and the reactor building and finally to the atmosphere. The model treats the fuel elements, the helium pressure boundary, the reactor building and the plateout (deposition) in the reactor building as four separate volumes as depicted in Figure 1.

For Volume 1, the initial inventories of the key radionuclides in the fuel compacts and fuel element graphite as a result of normal operation are determined. Radionuclide release mechanisms, which are individually accounted for in the model, include:

- release by diffusion from fuel particles with intact coatings,

- release from particles with defective silicon carbide (SiC) coatings,

- release from particles with the SiC and both pyrocarbon coatings failed, referred to as exposed or bare kernels, and

- release from heavy metal contamination.

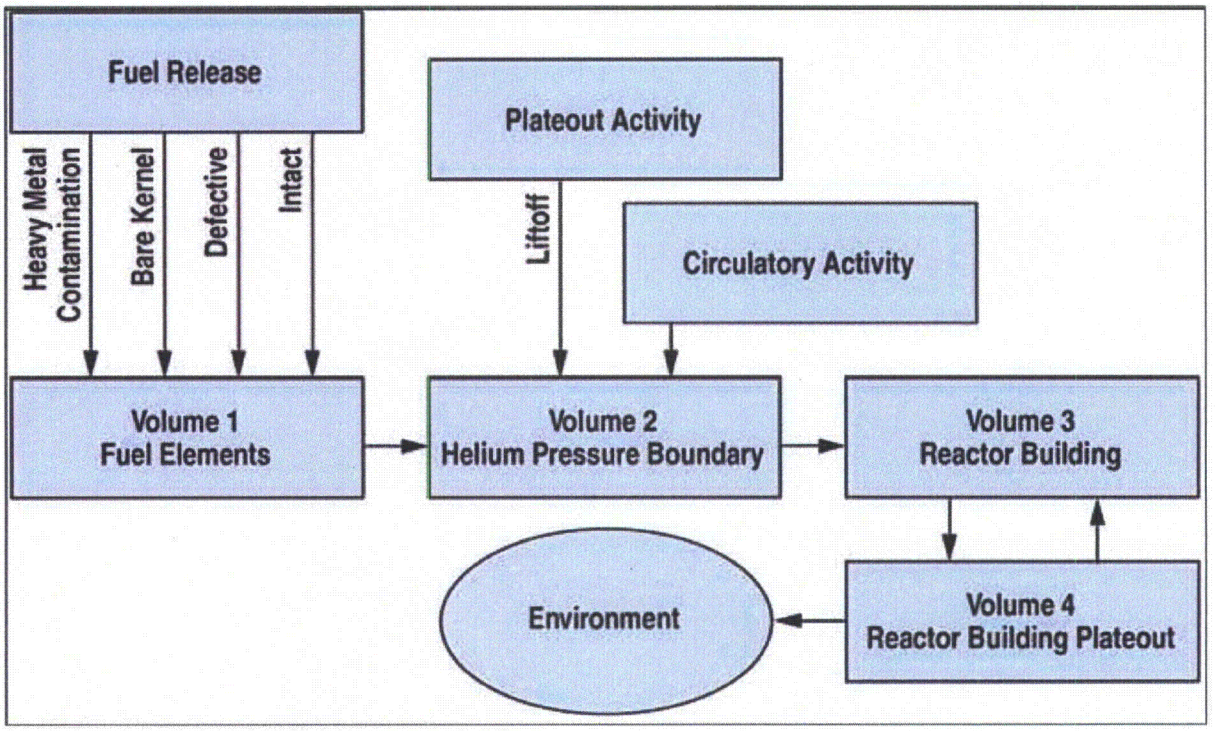

Figure 1. Consequence Uncertainty Model Volumes

The activity released from exposed kernels due to the temperature transient during any time interval is given by the following:

$$Q_{ek} = Q_f(f_{ex,i} R_{ex,i} + f_{ex,t} R_{ex,t})$$

where

Q_{ek} = Cumulative activity released from exposed kernels, Ci

Q_f = Fuel body activity, Ci

$f_{ex,i}$ = Initial fraction of fuel with exposed kernels

$R_{ex,i}$ = Time-dependent cumulative temperature induced fractional release from particles with exposed kernels at start of transient

$f_{ex,t}$ = Time-dependent cumulative fraction of fuel that fail during the transient as exposed kernels

$R_{ex,t}$ = Time-dependent cumulative temperature induced fractional release from fuel that fails during the transient with exposed kernels

The activity released from particles with defective SiC coatings, but intact OPyC coatings, is given by the following:

$$\mathbf{Q_{def} = Q_f(f_{def,i} R_{def,i} + f_{def,t} R_{def,t})}$$

where

Q_{def} = Cumulative activity released from particles with defective SiC, Ci

Q_f = Fuel body activity, Ci

$f_{def,i}$ = Initial fraction of fuel with defective SiC coatings

$R_{def,i}$ = Time-dependent cumulative temperature induced fractional release from particles with initially defective SiC coatings

$f_{def,t}$ = Time-dependent cumulative fraction of kernels which develop defective SiC coatings due to the transient

$R_{def,t}$ = Time-dependent cumulative temperature induced fractional release from particles whose SiC coatings fail during the transient

The activity released from particles with intact coatings is given by:

$$\mathbf{Q_{int} = Q_f R_{int}}$$

where

Q_{int} = Cumulative activity released from intact particles, Ci

Q_f = Fuel body activity, Ci

R_{int} = Time-dependent cumulative temperature induced fractional release from particles whose SiC coatings fail during the transient

The fraction of fuel with intact particles is assumed to be 1.0 throughout the transient, since the fraction of particles with defects on any kind is $<1x10^{-3}$ at all times during the transient. The activity released from heavy metal contamination is given by:

$$\mathbf{Q_{hme} = Q_f f_{hme} R_{hme}}$$

where

Q_{hme} = Cumulative activity released from heavy metal contamination, Ci

Q_f = Fuel body activity, Ci

f_{hme} = Fraction of heavy metal contamination

R_{hme} = Time-dependent cumulative fractional release due to heavy metal contamination

For simplicity, the time and nuclide dependencies have been neglected in the above equations describing the radionuclide release from the fuel, transport to the atmosphere, and offsite dose. The sum of the release due to each of these release mechanisms (exposed kernels, defective particles, intact particles, and heavy metal contamination) is the total amount of activity released to the Volume 1 Fuel Elements at any given time.

Similarly, for Volume 2, the helium pressure boundary, there is an initial inventory of radionuclides that is circulating and plated out on the primary circuit surfaces as a result of normal operation. The circulating and plateout activities are dependent on the fuel body inventory, the fraction of exposed kernels, and the heavy metal contamination fraction during normal operation. The model assumes that all of the circulating and plateout activity accumulated during normal operation is a result of these two release mechanisms (exposed kernels and heavy metal contamination). Then as a result of the leak/break of the helium pressure boundary, the circulating activity is released and a fraction of the plateout is lifted off, depending on the size and location of the leak/break. Radionuclides released from the fuel elements as a result of the core heatup are transported into the helium pressure boundary (primary circuit). Volume 3, the reactor building, receives the radionuclides released from the helium pressure boundary.

Each volume receives and releases activity at rates calculated by the model for specified time steps. Radionuclide decay and buildup is accounted for in each volume. Deposition and radionuclide decay are not considered enroute from the reactor building to the Exclusion Area Boundary (EAB).

The solutions for the nuclide activities in each volume are dependent on all input parameters being constant during the time interval of interest. For example, the vent flow from the helium pressure boundary to the reactor building must be constant over a time interval. Activity that has settled or plated out in the reactor building is treated as a fourth volume. Noble gases, which result from radioactive decay of precursors in Volume 4, are modeled as a release to the reactor building. Halogens, metallics, and all other nuclides remain in Volume 4 if they are the result of radioactive decay precursors.

Once the time dependent amount of activity in the reactor building is determined, integrating the product of the activity and the reactor building leakage rate over the time interval of interest yields the amount of activity released to the atmosphere in that interval, i.e.

$$\mathbf{S_{atm}} = \int \mathbf{(L_b A_b)}$$

where

S_{atm} = Activity released to atmosphere, Ci

A_b = Airborne activity in the reactor building, Ci

L_b = Reactor building leakage rate, hr^{-1}

The release from the reactor building is converted to a dose by multiplying by the weather dilution factor, the breathing rate, if applicable, and the dose conversion factor of the isotope for each time interval. The thyroid dose, whole body gamma dose, cumulative effective dose equivalent (CEDE), and total effective dose equivalent (TEDE) are calculated for an unsheltered person who is continuously standing at the edge of the EAB. TEDE, which is equal to the sum of the CEDE and whole body gamma doses, is used for comparison with the Plume Exposure Protective Action Guide (PAG). In general, the equation for dose is given by:

$$\mathbf{D} = \mathbf{S_{atm}}\ \chi/Q\ \mathbf{BR}\ \varepsilon$$

where

D = Dose, rem

S_{atm} = Activity release from the building, Ci

χ/Q = Weather dilution factor, s/m^3

BR = Breathing rate, m^3/s

ε = Dose conversion factor, rem/Ci

To obtain benchmark results for the MHTGR in the 1990s, the consequence uncertainty model was run with all best estimate, or median, values and without any uncertainty distributions. Benchmark calculations for the leak/break in the helium pressure boundary with loss of forced cooling using the consequence uncertainty model showed good agreement with intermediate results at each volume and for the final dose at the EAB that were calculated using the more detailed analytical methods.

For each uncertain variable in the consequence uncertainty model, the Monte Carlo uncertainty propagation software uses an uncertainty factor and a distribution to change the variable's value within a bounded range. A listing of these variables and their uncertainty distribution types is given in Table 1. Independent uncertainties are applied to the initial inventories for each of the 29 isotopes in the uncertainty model. Uncertainties for circulating and plateout activity account for the uncertainties in helium purification removal and plateout during normal operation. The variable uncertainties for circulating and plateout activity are correlated on an individual element basis. The remaining variable uncertainties in the model are the same for all isotopes. Also, the variable uncertainties remain constant with time for each sampling.

Table 1. Variable Uncertainties

Variable	Description	Distribution Type	Ratio of 95th to 50th percentile
1-29	Fuel inventory (29 isotopes)	lognormal	1.0 – 7.3
30-58	Circulating inventory(1) (29 isotopes)	lognormal	1.0 - 1123
59-87	Plateout inventory(1) (29 isotopes)	lognormal	1.0 - 2.5
88	Fraction of circulating and plateout inventories due to exposed kernels for gases	Linear 0-1 $\mu = .5$	1.9
89	Fraction of circulating and plateout inventories due to exposed kernels for metals	Linear 0-1 $\mu = .9$	1.1
90	Initial fraction of fuel particles with defective SiC coatings	lognormal	4
91	Fraction of fuel particles with defective SiC coatings due to transient	lognormal	4
92	Initial fraction of fuel particles with exposed kernels	lognormal	4
93	Fraction of fuel with exposed kernels due to transient	lognormal	4

Variable	Description	Distribution Type	Ratio of 95th to 50th percentile
94	Initial fraction of fuel contamination	lognormal	2
95	Fraction of Ag initially in fuel elements	lognormal	10
96	Liftoff fraction	lognormal	5
97	Release from particles with defective SiC	lognormal	2
98	Release from particles with exposed kernels	lognormal	4
99	Release from fuel contamination	lognormal	2
100	Release from intact particles	lognormal	2
101	Radionuclide diffusion rate in the fuel elements	lognormal	1.3
102	RB flow rate	normal	1.2
103	RB and HPB flow rate	normal	1.3
104	RB plateout rate for halogens	lognormal	10
105	RB settling rate for particulates	lognormal	10
106	Not used	lognormal	10
107	Weather model dilution factor χ/Q	PSID	10
108	Release due to temperature	lognormal	1.2

(1) A factor of four to account for upper bound design values is applied

RAI MST-89: Should the onset of prompt fatality be a consideration in the construct of frequency and consequence criteria for licensing basis events? If prompt fatalities were to be included, how would the NGNP approach for mechanistic source terms be affected? What is the basis for meeting the protective action guide (PAG) at a 425-meter exclusion area boundary (EAB)?

Comments: White paper Section 4.2 discusses NGNP top-level radionuclide control requirements. It is noted the top level requirements do not include onset of prompt fatality (300 – 500 rem dose). Also, it is noted that the requirements are discussed in relation to meeting the PAG at a 425-meter EAB.

Response MST-89:

The onset of prompt fatality is a consideration in the construct of the frequency and consequence criteria for licensing basis events. The approach to consideration of the onset of prompt fatality for the HTGR is discussed in the Licensing Basis Event Selection White Paper [1]. Consideration of prompt fatality has no effect on the NGNP approach for mechanistic source terms.

An HTGR that meets the top-level radionuclide control requirements presented in Section 4.2 of the Mechanistic Source Terms White Paper for a complete range of Licensing Basis Events, including Beyond Design Basis Events (BDBEs), would not produce offsite TEDE dose consequences of the 300-500 rem magnitude required to result in a prompt fatality for any credible LBE scenarios (i.e., those with a frequency of occurrence larger than 5×10^{-7} per plant year). Based on earlier analyses of accident

consequences for modular HTGRs, which included analysis of BDBEs, it is expected that the most limiting offsite dose requirement will be that associated with the design requirement of meeting the 5 rem thyroid PAG and the 1 rem TEDE PAG at the exclusion area boundary (EAB). These dose limits are more than two orders of magnitude below the prompt fatality dose level. Since the prompt fatality dose limit is applied at a distance of one mile from the plant rather than at the exclusion area boundary, there is an even larger difference between the PAG doses and the prompt fatality doses at the exclusion area boundary. This previous HTGR accident analysis experience is the basis for focusing analysis efforts on the PAG at the exclusion area boundary.

Figure 1, taken from [1], presents the frequency-consequence (F-C) curve with the top level regulatory criteria and further illustrates the points made in the previous paragraph. It is clear from this figure that in both the DBE and BDBE regimes the requirement to meet the PAGs at the EAB is significantly more restrictive than the requirement to meet the prompt mortality safety goal.

Reference [1] discusses examination of incredible accident sequences with frequency of occurrence below the current cutoff of 5×10^{-7} per plant year in the frequency and consequence criteria. The approach to be taken to the plant Probabilistic Risk Assessment will ensure that there are no events with a frequency of occurrence just below the current cutoff that are missed. This will assure the adequacy of the lower cutoff frequency. The approach being taken for the HTGR of meeting the PAGs at the exclusion area boundary for Beyond Design Basis Events ensures that no credible accident sequence can result in prompt fatalities outside the exclusion area boundary. This approach meets the intent of the NUREG-0654, Section 1.D.2, planning considerations, which requires that for unlikely accident sequences there be negligible early fatality risk outside the EAB.

As discussed in [1], in addition to meeting the F-C curve limits for individual events, the overall integrated risk of all event sequences from the plant PRA is to be evaluated to show that the calculated plant risk meets the reactor safety goals.

The qualitative reactor safety goals are discussed in NUREG-0880 in terms of quantitative health objectives (QHOs). The following QHOs are used in determining satisfactory achievement of the safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of one percent (0.1%) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of one percent (0.1%) of the sum of cancer fatality risks resulting from all other causes.

The prompt fatality QHO limits the increase in an individual's risk of prompt fatality to 0.1% of that from all "other accidents to which members of the U.S. population are generally exposed," which is about 5×10^{-4} per year. Therefore the incremental prompt cancer risk goal established by the Reactor Safety Goals is 5×10^{-7} fatalities/year. The "vicinity" of a nuclear power plant is understood to be a distance extending to 1 mile from the plant site boundary.

The latent fatality QHO limits the increase in an individual's annual risk of death to 0.1% of 2×10^{-3} per person-year, or an incremental increase of no more than 2×10^{-6} per person-year. For this QHO, the "area" is understood to be an annulus of 10-mile radius from the plant site boundary. The cancer risk is also

determined on the basis of an average individual risk, i.e., by evaluating the number of latent cancers (societal risk) resulting from all accidents to a distance of 10 miles from the plant site boundary, weighted by the frequency of the accident, dividing by the total population to 10 miles, and summing over all accidents.

The NGNP project has a design goal to meet the PAGs at the EAB. By meeting this design goal, it is clear that the prompt fatality safety goal will be met by orders of magnitude. By meeting the PAG (1 rem for an early release) at the EAB for all LBEs, the sum of the average risk of a prompt fatality from all accidents to an individual will be less than the prompt fatality goal with significant margin. Similarly, by meeting the PAG at the EAB for all LBEs, the sum of the average risk of a latent fatality from all accidents to an individual will be less than the latent fatality goal.

Another way of stating the result is that if all events including the BDBEs meet the dose limits at the EAB, then the QHOs within the larger distances from the site, and within the closer EAB, are met with large margins.

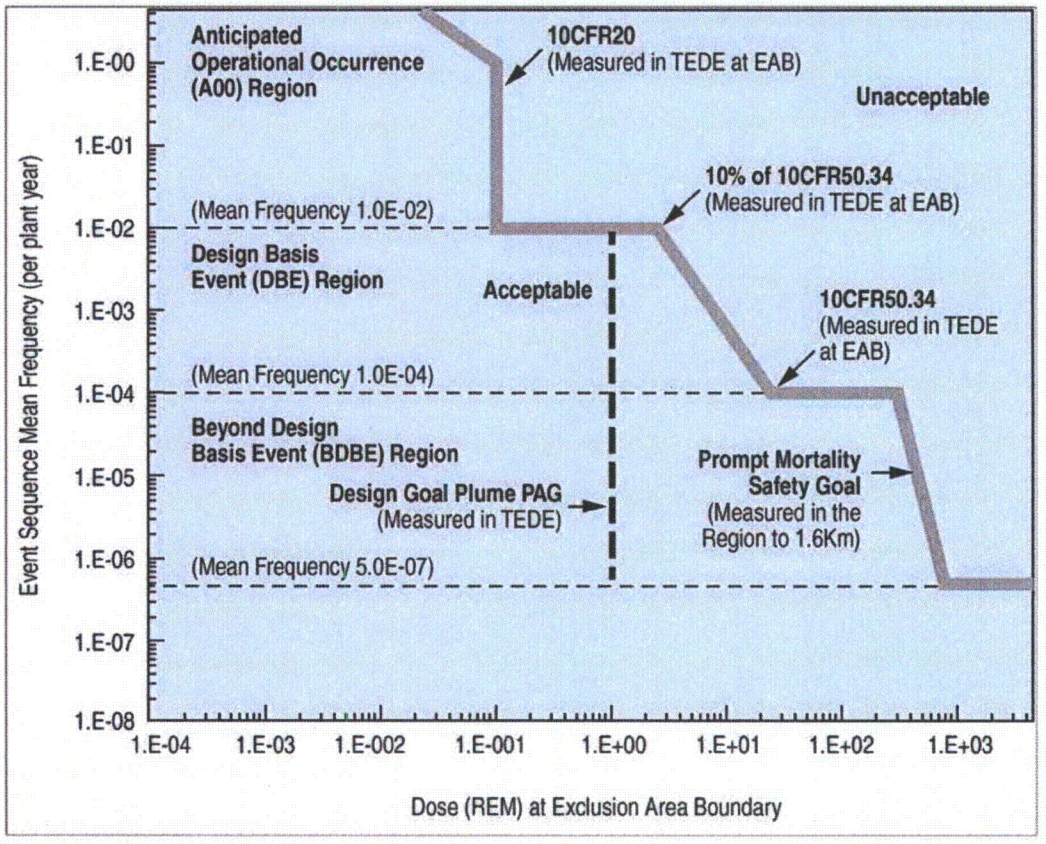

Figure 1. Frequency-Consequence Chart with Top Level Regulatory Criteria

Reference:

1. INL/EXT-10-19521, Next Generation Nuclear Plant Licensing Basis Event Selection White Paper, September 16, 2010

RAI MST-90: The white paper states that Table 4-3 is based on several earlier analyses. Please provide references to these earlier analyses.

Response MST-90:

The references are historical, internal GA and PBMR documents. The values in the table are "typical" for previous modular HTGRs with design details that are, in some cases, different from those of the NGNP. The values were presented in the white paper solely to provide background information and a sense of the order of magnitude to be expected. Unique values will be calculated for NGNP, and it is expected that they will be consistent in magnitude with those from the earlier modular HTGR analyses.

RAI MST-91: SORS is an accident analysis code in the line of the NRC code MELCOR and TAC2D is a two-dimensional heat transport code. Clarify how the accident analysis will be done and how the NGNP mechanistic source term be calculated from these codes and methods.

Response MST-91:

How the NGNP mechanistic source term will be calculated is both design dependent and designer dependent. TAC2D and SORS are GA codes for predicting the thermal response, fuel performance, and radionuclide release from a prismatic core during depressurized conduction cooldown accidents. As new methods are developed by the NGNP Project, they will be evaluated by GA as replacements for SORS and/or TAC2D. Analogous codes for use in source term calculations for prismatic core designs are in various stages of development at AREVA. Appendix D and E of the Mechanistic Source Term White Paper provide additional detail on the currently used codes for GA and PBMR, but at this stage of the design, methods are subject to improvement or replacement.

The TAC2D code is a general purpose two-dimensional finite-difference heat transfer code that is used to model the depressurized conduction cooldown in the prismatic HTGR design using r-z geometry. The geometric model encompasses the active core; the inner, outer, top, and bottom graphite reflectors; the graphite core support floor; the core support plates; the core barrel; the insulated upper-plenum structure; the reactor vessel; radiation shielding material above and below the reactor vessel; and the air-cooled RCCS panels along with the concrete behind them. Heat transfer within this model is principally by conduction through the core and reflectors to the top and bottom core surfaces and to the core periphery adjacent to the core barrel. Heat is transferred by thermal radiation and conduction across the gas spaces separating the core surfaces and metal support structures and shrouds, and across the gas spaces to the reactor vessel. Free convection from heated surfaces is represented by placing a multiplicative factor on the thermal conductivity of the gas in the spaces between surfaces. Heat is transferred predominately by thermal radiation across the gas spaces separating the reactor vessel and the reactor cavity cooling panels. A convective flow of air through the cooling panels, which removes the heat from the panels, is calculated. SINDA/FLUINT and ANSYS are additional heat transfer codes that have been and may be used for analyzing temperature transients. Time-dependent core temperatures are used by the SORS code to predict fuel performance and fission product release during the transients.

The SORS code was developed to calculate the release of fission products, transuranics, and other radionuclides from the reactor fuel to the coolant for postulated core heat-up scenarios in an HTGR core. SORS calculates the fractional release of fission products from the fuel particles based on diffusion in a one-dimensional spherical geometry. In calculating the fractional release of fission products from the fuel, SORS considers:

- the presence of various types of defects in the fuel particles at the onset of the accident,
- failure of one or more of the fuel particle coating layers by a number of mechanisms during the accident,
- the heavy metal contamination fraction,
- diffusion of volatile and metallic fission products through the fuel kernel,
- diffusion of metallic fission products through intact SiC coatings, and
- diffusion of volatile fission products through intact OPyC coatings. The calculations account for the influence of burnup, fast-neutron fluence, irradiation and accident condition temperatures, and radionuclide concentrations.

The fuel particle defect fractions at the onset of an accident (or heating phase) are provided as input to SORS based on the fuel performance calculated using SURVEY code. SORS also considers fission product release from heavy-metal contamination.

The accident scenario parameters provided as input data to SORS include the time-averaged temperature during the irradiation phase preceding the accident, the burnup and fast-neutron fluence at the onset of the accident, and the time-dependent temperature during the accident. SORS calculates the SiC failure probability during the accident as a function of time, temperature, burnup, and fast fluence, and uses the values along with the initial failure probabilities and particle defect fractions input to the code to calculate the fraction of failed fuel particles.

Because there is essentially no thermal gradient during accidents, particle failure by kernel migration does not occur. There also is no particle failure by corrosion of the SiC by fission products because of the relatively short reaction times involved during accident scenarios. The dominant failure mechanism for SiC coating failure during accident conditions is SiC thermal decomposition.

The fuel particle failure fractions calculated in SORS provide the input for fission product release from the particles, which is calculated using the fission product release models that are summarized in Section 4.4 and Appendix C of the Mechanistic Source Terms White Paper. These include models for diffusion of gaseous and metallic fission products from the kernel in failed fuel particles, diffusion of gaseous and metallic fission products through intact OPyC coating layers in particles having a defective SiC layer, and diffusion of metallic fission products through the SiC in intact fuel particles.

The SORS code model for transport of fission products that are released from the fuel particles through the fuel compact matrix, through the fuel element graphite, and into the helium coolant is described in the response to RAI MST-76.

RAI MST-92: It appears that GAUGE, BURP, and few other codes listed in Section 4.5.1 are legacy codes from the 1960s for which the validation data base must be very limited. Does the NGNP project have a plan for verification and validation of these legacy codes? If so, what is the schedule for such

activities? What data will be required and generated for verification and validation? If verification and validation is not planned, provide a justification.

Response MST-92:

All codes used for the design of safety-related SSCs in the modular HTGR demonstration plant, including any legacy codes, will be verified and validated to NQA-1-2008/2009 standards and with consideration of NRC Regulatory Guide 1.203. The existing validation data base is independent of the vintage and genealogy of any particular code.

In particular, the NGNP/AGR fuel development and qualification program will provide the design-specific data (e.g., performance data for the reference LEU UCO fuel) necessary to complete the verification and validation of the design methods used to generate source terms. The schedule for these validation and verification activities will parallel the schedule for the AGR fuel program and the NGNP license application. The code validation and verification effort will continue through startup testing and initial operation of the NGNP demonstration plant.

Industry-standard codes, such as ANSYS, RELAP5, MCNP, etc., are used whenever such codes have the requisite capabilities for modular HTGR plant design and safety analysis. However, for certain aspects of prismatic core design and analysis, especially regarding fuel performance and fission product transport, the legacy codes are still the best available analytical tools.

Moreover, some of these legacy codes have an ongoing development history that extends well beyond their initial development date. For example, the initial version of the SURVEY code, which performs a full-core fuel performance and fission gas release analysis, was developed in 1978 for the analysis of large HTGRs with HEU UC_2 TRISO/ThO_2 BISO fuel. Since then, the SURVEY code has been modified multiple times to add new component models for different fuel types, including LEU UCO TRISO fuel. The most recent modifications to SURVEY were made in 2009 when the capability to model fuel "shuffling" in the core was added (fuel elements are moved from one core location to another during their core residence time). Moreover, these upgraded legacy codes are now operational on a variety of modern computing platforms, ranging from desk-top workstations to massively parallel supercomputers.

RAI MST-93: In Section 4.6.1, it is stated that significant reactor surveillance data will become available from the Japanese HTTR. Please describe (a) the status of any agreements under which such data will be provided, and (b) details on the surveillance data that will be acquired from the HTTR and when it would be available.

Response MST-93:

The JAEA and the DOE are working toward a bilateral agreement covering nuclear research and development, of which one major component is HTGR technology. Given the pace of progress to date toward ratification, this Agreement should be considered as applicable for long-term Very High Temperature Reactor (VHTR) R&D activities.

Of more immediate interest are efforts to acquire data from a recent campaign of HTTR safety tests. JAEA has sought and received Organization for Economic Co-operation and Development (OECD) sponsorship of three safety (loss of forced cooling) tests. The U.S. NRC is a party to this project, and the NGNP Project will acquire the data and analyses of these tests through inter-governmental transfer. The NGNP Project has also drafted Statements of Work to acquire data by contract directly with JAEA. Tritium permeation data from HTTR has already been acquired in this manner. Data from critical experiments and recent safety tests not covered under the OECD project will be acquired as funds permit.

Under OECD-CNSI (Committee on the Safety of Nuclear Installations) sponsorship, JAEA is expected to perform the following three safety tests at HTTR:

Test 1: Trip of all three Helium Gas Circulators (HGCs) starting from reactor power of 9 MWt and coolant inlet/outlet temperatures of 180C/320C.

Test 2: Trip of all three HGC starting from reactor power of 30 MWt (full power) and coolant inlet/outlet temperatures of 395C/850C.

Test 3: Trip of all three HGC starting from reactor power of 9 MWt and coolant inlet/outlet temperatures of 180C/ 320C and all reactor vessel cooling system (VCS) pumps disabled.

These LOFC tests are initiated by tripping all three HGCs of the HTTR while deactivating all reactor reactivity control to disallow reactor scram due to abnormal reduction of primary coolant flow rate. The tests will demonstrate HTTR performance for an anticipated transient without scram (ATWS) with occurrence of reactor re-criticality. The tests will be conducted with and without active function of the VCS. The vessel cooling system in the HTTR corresponds to the RCCS in the modular HTGR and cools the reactor pressure vessel mainly by radiation. The tests are designed to provide data for the validation of system and safety codes, particularly with regard to reactor kinetics, core physics, and thermal hydraulics.

The three safety tests have been scheduled, under the test agreement, to be conducted over a three year period beginning in April 2010. Test 1 was conducted in December 2010. As the U.S. NRC is a party to the project, the data will be available to all agencies of the U.S. Government. A final report will be prepared and delivered to conclude the project.

The NGNP Project is also pursuing a long-term research collaboration with JAEA. Preliminary discussions are underway to establish the technical content. A bilateral agreement covering cooperation in nuclear science and technology is currently the subject of discussions between the U.S. and Japanese government. In the event that a bilateral agreement cannot be signed, INL will pursue a Cooperative Research and Development Agreement (CRADA) with JAEA. Informal discussions with DOE management have indicated that a bilateral agreement could be signed in 2011.

Statements of Work to acquire data from past HTTR and related experiments have been drafted and approved by INL. Under one contract, data and reports from the Simple Homogenous Experiment (SHE) and Very High Temperature Reactor Critical facility criticality tests would be delivered. Under the other contract, data from three (non-OECD) safety tests performed on HTTR in 2010 and 2011 would be delivered to the NGNP Project and shared with the NRC. The tests include:

Test 1: Operational data from HP-11: 50 day steady state run at full power (30 MWt) and outlet temperature of 950C.

Test 2: Temperature coefficient measurements taken in December 2010

Test 3: Data from RS-13: LOFC test (trip of all three HGC and one of two VCS pumps.

JAEA has been registered by INL as a qualified supplier for this information. However, delivery of this information to the NGNP Project is on hold pending resolution of funding issues.

In 2010, tritium concentration data from the HTTR were acquired under contract from JAEA with General Atomics acting as an intermediary independent reviewer. These data were obtained during the HP-11 operation described above. The tritium data are being used by INL to validate calculations of

tritium permeation throughout the plant using TPAC (Tritium Permeation Analysis Code). A technical report containing the results of the TPAC validation was completed on June 30, 2011. The document is designated proprietary and Official Use Only and has been provided to the NRC with appropriate restrictions on further distribution.

RAI MST-94: Fort Saint Vrain (FSV) is noted as another source of data. FSV used lower-quality TRISO-coated fissile/fertile particles with kernels of HEU-carbide/thorium-carbide as opposed to NGNP's higher-quality TRISO-coated fuel particles with kernels of LEU-oxide and LEU-oxycarbide as described in the FQ White Paper. In view of the differences associated with these and any other key attributes of FSV and NGNP fuel, please characterize the applicability of FSV data on fuel behavior and fission product transport to the validation/assessment of codes for predicting NGNP fuel behavior and fission product transport.

Response MST-94:

NRC correctly notes that FSV fuel consisted of kernel materials that are different from those of the NGNP and that initial fuel quality was lower than that to be specified for NGNP fuel. However, other aspects of FSV operation — including the role of the core graphite and primary circuit surfaces in sorption and plateout of various radionuclides, the role of the helium purification system in maintaining reactor coolant chemistry, and the use of circulating activity monitors, a plateout probe, and moisture monitors to characterize primary helium impurities and radionuclide content— were effectively the same as those of the NGNP. The methods used to model coated-fuel particle performance and fission product transport in the core and the primary coolant system are applicable to a wide range of fuel materials, core conditions, and primary circuit surface conditions provided that appropriate material property input data are used. Therefore, FSV data on fuel performance and fission product transport are applicable to the validation and assessment of codes for predicting NGNP fuel performance and fission product transport. The information provided in Section 4.6.1 of the Mechanistic Source Terms White Paper on FSV experience indicates that these methods can predict fission-product transport in the HTGR with an acceptable degree of conservatism.

RAI MST-95: Air ingress effects are not specifically mentioned in Table 5-1 under "Gap Description" in the fission product release category. Are air ingress effects going to be addressed related to the calculation of mechanistic source terms?

Response MST-95:

Air ingress effects will be addressed in the calculation of mechanistic source terms. They were not included in Table 5-1 because they are not considered to be a "major" source term knowledge gap.

RAI MST-96: Describe in detail the experimental data used to evaluate effective diffusion coefficients under NGNP operating and accident conditions of temperature, neutron flux, and neutron fluence.

Comments: Table 5-1 states the following: "The suitability of using an "effective" diffusion coefficient in a model based on Fick's Law of diffusion needs to be demonstrated to ensure accurate calculations of fission product release to the primary circuit for use in shielding analyses and source term calculations."

Response MST-96:

Many approaches have been used to characterize radionuclide transport in HTGR core materials. While the approaches have been diverse, the transport models and material property correlations used to predict radionuclide transport in support of reactor design and safety analysis are, in general, based upon

experimental data that have been correlated with phenomenological models that are based upon first principles, rather than simply nth-order polynomial fits of the data. Often, correction factors are added to the first-principles model to account for irradiation effects. The technical bases for the reference models and material property correlations for predicting radionuclide transport in prismatic fuel HTGRs are summarized in [1].

The detailed experimental data used to evaluate effective diffusion coefficients under NGNP operating and accident conditions of temperature, neutron flux, and neutron fluence are described in [2]. While the international data bases for characterizing radionuclide transport in HTGR core materials are extensive, there are gaps that will be addressed by NGNP/AGR fuel development and qualification program. In particular, additional data are needed for the reference UCO TRISO particle and for the new matrix and graphite grades being qualified by NGNP Project. The planned AGR-3/4 test is designed to address these data needs, and the refined radionuclide transport models and correlations derived from these AGR-3/4 data will be independently validated by the planned AGR-8 test.

As discussed in the response to RAI MST-9, the design goal is to be able to predict fission metal release from the core with a factor of 10 accuracy at 95% confidence. It is anticipated that a Fick's 2nd Law model (transient homogeneous diffusion) using effective diffusion coefficients will be adequate to achieve and demonstrate that goal. While it may be conceptually ideal to employ more rigorous mechanistic transport models (e.g., a surface diffusion-trapping model for transport in graphite), such an approach is judged to be unnecessary if, as expected, the design goal can be met with a transport model based upon Fick's 2nd Law. The requisite single-effects data to derive the material property (e.g., surface diffusivities, trap densities as a function temperature and fast fluence, etc.) that are required for more complicated transport models are not available from the existing international data base. Once the AGR-3/4 test data become available, alternative transport models would be considered to correlate the data if the current model based on Fick's 2nd Law is determined to be inadequate. If a more complex model is ultimately adopted, supplemental testing would be needed to obtain the required material property data.

Once the AGR-3/4 data become available and the reference radionuclide transport models for core materials have been refined accordingly, it is anticipated that one or more Licensing Topical Reports on this subject will be prepared by the license applicant.

References:

1. Martin, R.C., "Compilation of Fuel Performance and Fission Product Transport Models and Database for MHTGR Design," ORNL/NPR-91/6, Oak Ridge National Laboratory, October 1993.

2. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," TECDOC-978, International Atomic Energy Agency, November 1997.

RAI MST-97: Table 5-1, it is stated that significant reactor surveillance data will become available from the Japanese HTTR. Please describe (a) the status of any agreements under which such data will be provided, and (b) details on the surveillance data that will be acquired from the HTTR and when it would be available.

Response MST-97:

See the response to RAI MST-93. RAI MST-93 and this RAI (MST-97) are identical.

RAI MST-98: The white paper mentions that calculations are underway to determine the impact of dust on the behavior of fission products in the system. Please describe the calculations and the details of how they address the impacts of dust on release source terms.

Comments: Important factors to be addressed by such analytical studies would include:

a. how the dust is produced and in what quantities, chemical compositions, and grain sizes, as suggested by related comments under RAI MST-14 and MST-16,

b. the effects of various postulated break sizes, including large (e.g., cross-duct) breaks, on how much dust exits the primary boundary,

c. the effects of variable peak fuel operating temperatures, including temperatures in postulated core hot spots, and fission product diffusivity uncertainties on the dust activity acquired from metallic fission products (e.g., cesium) from exposed fuel kernels and intact CFPs, and

d. the effects of water-ingress on dust activity and release.

Response MST-98:

The calculations referred to on page 48 of the Mechanistic Source Terms White Paper are calculations that were underway at PBMR to evaluate the impact of dust in the AVR on fission product behavior. As a result of the cessation of activities on the pebble bed reactor program in South Africa, these calculations are no longer underway. Details regarding the calculations, including those requested in this RAI, are no longer available to the NGNP Project. Following resolution of the RAIs, the white paper will be revised to reflect these developments.

RAI MST-99: When will the NGNP project provide its recommendations on the extent of additional testing to validate fission product liftoff and plateout models? Would there be any merit to soliciting external input (e.g., from NRC input under the existing DOE/NRC interagency agreement on NGNP) during the formulation of recommendations?

Comments: The white paper states that the extent of additional testing needed for fission product liftoff and plateout models is still under discussion within the NGNP Project Team.

Response MST-99:

The additional testing referred to in this RAI and in the Mechanistic Source Terms White Paper is addressed in the response to RAI MST-102.

RAI MST-100: Please provide technical basis for the white paper's stated conclusion that PHEBUS and DEMONA are generally not applicable to HTGR fission product transport in reactor building.

Response MST-100:

No direct measurements have been made of radionuclide removal from contaminated helium by condensation, settling, and plateout under the conditions expected in the HTGR reactor-building during a core heatup transient. There is an extensive LWR and Canadian Deuterium Uranium (CANDU) database on the behavior of radionuclides in water-cooled reactor containment buildings, and major experimental programs have been conducted to characterize further the behavior of radionuclides in LWR containment buildings (e.g., the international PHEBUS test program in France and the DEMONA test program in Germany). Some of these LWR data, especially those that relate to radionuclide partitioning between

steam and liquid phases in steam-water mixtures, may be applicable to radionuclide behavior in the HTGR reactor building. Consequently, the extensive literature on radionuclide transport in water-cooled reactor containment buildings was reviewed by staff at General Atomics, and an assessment was made as to the utility of this database to support analysis of radionuclide transport in the HTGR reactor building. The results of this assessment are summarized in Table 1.

Severe core damage accidents in water reactors are characterized by extremely complex physical and chemical phenomena that occur in the degrading (melting) core, reactor coolant system, and containment building. The materials released from the core include fission products, heavy metals (such as uranium and plutonium), structural materials, and complex compounds thereof. This material is typically present in the reactor coolant system and containment building as reactive aerosols. Much of the international research has focused on characterizing the transport behavior of these aerosols during various accident scenarios. The other major focus has been characterizing the transport behavior of radioiodines in the containment building, including the partitioning of iodine between steam and liquid water.

The transport behavior of radionuclides during core heatup accidents in modular HTGRs is much more simple by comparison. Relatively few radionuclides (primarily, radioisotopes of Kr, Xe, I, Te, Ag, Cs and Sr) are released from the core; no heavy metals or core structural materials are released, and the radionuclide mass concentrations are so low that no aerosols are expected to form, although fission products may condense or sorb onto existing aerosols. Aerosols ("dust") present in the primary coolant circuit during normal operation may contribute to release of plateout activity from the primary circuit into the reactor-building during rapid depressurization accidents, especially in pebble-bed HTGRs, which are expected to have higher dust concentrations, but the chemical and physical nature of this aerosol is quite different from the aerosol produced in an LWR severe core damage accident. Consequently, the water-reactor aerosol transport data appear to be of little direct relevance to the behavior of radionuclides in the HTGR reactor building. However, the aerosol transport codes might be adoptable for analyzing depressurization transients in HTGRs, provided the requisite material property data were available. Some of the water-cooled reactor data on iodine behavior in steam-liquid water systems might be applicable to HTGRs that use a steam cycle power conversion system and can experience moisture ingress events.

In summary, the extensive, experimental water-cooled reactor database for radionuclide transport in low leakage containment buildings is judged to be of limited value for refining and independently validating the design methods used to predict radionuclide transport in the HTGR reactor building.

Table 1. Applicability of Water-Cooled Reactor Data to Modular HTGR Reactor Building Fission Product Transport Analysis

Water-Cooled Reactor Containment Data	Applicability to Modular HTGR Reactor Building	Comments
Radionuclide release from core during normal operation (initial conditions)	Limited: Fission product release data from UO_2 pellets may be relevant to release from oxide-bearing fuel kernels.	Fuels forms are very different. Applicability of UO_2 pellet data would need confirmation.
Initial inventories and chemical speciation of radionuclides in reactor coolant system (initial conditions)	None	Different coolants; hence, different chemistry. Different mix of radionuclides (activation products prominent in water-cooled reactors).

Radionuclide release rates from core during unrestricted core heatup (core melt) accidents	None	Heavy metal and structural materials not released from modular HTGR cores.
Aerosol formation in reactor coolant system and containment building	None	Aerosols (dust) in modular HTGRs composed of graphite and metal oxides. No evidence of aerosol formation during core heatup accidents.
Aerosol transport in reactor coolant system and containment building	Limited. Aerosol deposition and reentrainment models might be applicable.	Data on modular HTGR-specific aerosol characteristics and material properties needed.
Iodine chemistry (compound formation) in reactor coolant system and containment building	Fundamental thermochemical data should be applicable.	Iodine concentrations much higher and chemical systems more complicated in water-cooled reactors.
Iodine partitioning in steam-liquid water systems	Applicable to steam cycle modular HTGRs, provided water chemistry is similar.	

RAI MST-101: Please provide clarifying information on the NGNP approach for calculating a event-specific mechanistic source term in the following specific areas:

1. Treatment of uncertainties in source term calculations;
2. Efficacy of the legacy codes for source terms calculations; and
3. Validation/assessment data bases for codes.

Comments: This summary-level RAI seeks integration of the responses to related RAIs above.

Response MST-101:

1. As described in the response to RAI MST-88, the best-estimate analysis of the mechanistic source terms proceeds as follows: For each independent input parameter the best estimate and its uncertainty distribution is determined for the event sequence of interest. The dependent output parameter and its uncertainty distribution are calculated with a Monte Carlo routine. The uncertainty distributions of the output parameter can have a range of shapes depending on the algorithm and the distributions of the input parameters (e.g., log normal, Weibull, etc.).

2. As described in the response to RAI MST-92, industry-standard codes are used whenever such codes have the requisite capabilities for HTGR plant design and safety analysis. However, for certain aspects of prismatic core design and analysis, especially regarding fuel performance and fission product transport, the legacy codes are still the best available analytical tools. Moreover, some of these legacy codes have an ongoing development history that extends well beyond their initial development date because many of these legacy codes have been upgraded multiple times to incorporate improved component models. Moreover, these upgraded legacy codes are now

operational on a variety of modern computing platforms, ranging from desk-top workstations to massively parallel supercomputers.

3. As also described in the response to RAI MST-92, all codes used for the design of safety-related structures, systems and components (SSCs) in the modular HTGR demonstration plant, including any legacy codes, will be verified and validated to NQA-1-2008/2009 standards and with consideration of NRC Regulatory Guide 1.203. The verification and validation effort will consider the use of all credible integral test data for code validation, including existing Fort St. Vrain startup and surveillance data and anticipated future data from the Japanese HTTR, which is described in the response to RAI MST-93. The on-going NGNP base technology programs will be relied to provide HTGR design-specific integral validation data, especially in the areas of UCO fuel and graphite.

RAI MST-102: The approach for planned fission product tests focuses primarily on release and transport phenomena within fuel (i.e., coated particles, pebbles, and compacts). Provide more information on tests for other transport phenomena, including within the helium loop, reactor building, etc., to judge the adequacy and completeness of the proposed approach.

Response MST-102:

As explained in the Mechanistic Source Terms White Paper, the decision regarding additional testing that may be needed to obtain information on fission product transport phenomena depends, among other things, on allocation of uncertainty in the effectiveness of each of the five barriers to fission product transport. For example, if fuel qualification testing were to demonstrate that the effectiveness of the coated particle fuel is higher than currently expected with relatively low uncertainty, it might be possible to demonstrate compliance with Top Level Regulatory Criteria (TLRC) and design goals while taking relatively little credit for retention of radionuclides in the reactor building. In such an example, technology development resources could be focused on developing information on the performance of radionuclide transport barriers other than the reactor building. Alternatively, development of information on the performance of the reactor-building could focus on approaches to obtaining adequate data that are more cost and schedule effective than other possible approaches.

A recent review was conducted within the NGNP Project to assess:

1. the available data that pertain to fission product transport phenomena in the primary circuit and in the reactor building,
2. additional such data that will be obtained under the currently planned AGR Fuel Development and Qualification Program,
3. the need for further data to support the NGNP design and safety analyses relative to the TLRC and design goals for offsite dose performance,
4. potential alternatives to acquisition of additional data on transport in the primary circuit and in the reactor building, such as design changes or reducing uncertainties in fuel performance and fission product transport phenomena within the fuel, and
5. alternative approaches to acquisition of any needed additional data on transport in the helium loop and in the reactor building.

The available data that pertain to fission product transport in the primary circuit and in the reactor building are summarized in Section 4 and Appendix C of the Mechanistic Source Terms White Paper.

More detailed information is contained in [1] and [2]. These data have been obtained by a combination of separate effects testing, irradiation testing and post irradiation safety testing of coated particle fuels, out-of-pile testing, integrated in-pile loop testing, and examinations of earlier operating HTGRs such as Peach Bottom Unit 1, Fort St. Vrain, and the AVR

Additional such data will be obtained under the AGR Fuel Development and Qualification Program. The AGR program is also expected to provide data that will support use of lower statistically significant in service particle failure rates than those assumed to date in HTGR core analyses under normal operating and accident conditions. Startup testing and operational testing and surveillance during the first few years of operation of the demonstration plant will be conducted to confirm these results for normal operations. Startup testing and operational testing and surveillance are discussed in the responses to RAI FQ-3/MST-3 and RAI FQ-23/MST-28.

Additional data on fission product plateout and reentrainment in the helium loop and the reactor-building are desired, particularly in the presence of dust and/or moisture. It is currently projected that adequate data can be obtained with laboratory scale and out-of-pile separate effects testing and limited integrated out-of-pile loop testing. Data obtained in these tests will be compared with that obtained in earlier in-pile loop tests (particularly the COMEDIE in-pile loop tests discussed in Appendix C of the Mechanistic Source Terms White Paper) to confirm the adequacy of this approach. A more detailed review of existing data will be conducted, and scaling analyses will be performed to support development of the specifications for an out-of-pile loop. Tracers will be used in the initial testing in this loop, and evaluation of the results of the tracer testing will be used to establish if additional testing is required using irradiated fuel. Testing using irradiated fuel would likely be conducted in hot cells.

The AGR Fuel Development and Qualification Plan will be updated to reflect this approach at the next regularly scheduled biannual update (late FY12).

References:

1. Martin, R.C., "Compilation of Fuel Performance and Fission Product Transport Models and Database for MHTGR Design," ORNL/NPR-91/6, Oak Ridge National Laboratory, October 1993.

2. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," TECDOC-978, International Atomic Energy Agency, November 1997.

RAI MST-103: How much dust is generated in the reactor during operation? How is it generated, and what is it's size distribution?

Comments: This and following questions about dust are mostly concerned about one thing: although potential dust issues are identified pretty thoroughly in the paper, what is not addressed are quantitative questions like "How much?" and "What is the impact?" etc. Also not addressed is how INL plans to answer those questions. I get the impression that INL is going to rely on historical operation data and HTTR data, when it becomes available. I see no experiments planned to get further data. Regarding dust generation, the impression I got during the Dust Workshop was that most dust is formed by graphite-metal abrasion, i.e., the fuel handling system in the HTR-10 and AVR, or the circulator piston rings in the HTTR. In the PMR design, I get the impression that the designers think that there will be little dust, so the discussion about it is moot. I heard a lot about radiation-induced spalling – I have no idea how they would even measure that. Another possibility is from refueling while handling the graphite blocks.

Presumably any dust from this mechanism would be between blocks and not available to the coolant channels.

Response MST-103:

The amount of dust generated in the reactor during operation, the manner in which it is generated, and its chemical composition and size distribution are all dependent upon the specific design of the modular HTGR. After a design selection is made, this information will be provided by the license applicant, as needed, in topical reports in support of the NGNP license application.

RAI MST-104: Does this dust deposit in the primary or core, and where?

Response MST-104:

In principle, particulate matter ("dust") in the primary coolant can deposit on all surfaces in the primary coolant circuit, including the reactor core. The distribution of the deposited dust will depend, in particular, upon the chemical composition of the dust, its particle size distribution, and the primary circuit geometry. These attributes are all dependent on the specific design of the modular HTGR. After a design selection is made and necessary analyses are completed, this information will be provided, as needed, by the license applicant in topical reports in support of the NGNP license application.

As discussed in the response to RAI MST-14, HTGR operating experience strongly suggests that dust is generated in pebble bed reactors (as observed in AVR) primarily from abrasion between fuel spheres and the fuel handling system and that prismatic core reactors are expected to generate significantly smaller amounts of dust. The primary consequence of dust in the primary circuit will be to potentially influence the transport behavior of radionuclides released from the core during normal operation and postulated accidents. Any dust circulating in the primary coolant will be substantially removed by the helium purification system. The primary location of the dust that deposits in the primary coolant circuit will be design dependent. In a steam-cycle modular HTGR, the primary location will be the steam generator since it has the largest surface area and the highest mass transfer rates of the ex-core primary circuit components.

RAI MST-105: How much fission products accumulate on the dust via adsorption orplateout, versus deposition on metal surfaces?

Response MST-105:

The degree to which radionuclides that are released from the core would become associated with particulate matter ("dust") that may be present in the primary circuit would depend upon a number of design-dependent factors, the most important of which are:

- the concentrations of dust in the primary coolant and deposited on the surfaces in the primary circuit,
- the chemical and physical nature of the dust, and
- the chemical identities of the various radionuclides released from the core.

Conceptually, it is evident that if the concentrations of circulating and deposited dust in the circuit are very low (i.e., such that the collisions between fission product atoms and circulating particulates are infrequent compared to collisions between fission product atoms and helium atoms and/or fixed surfaces), then dust effects should be unimportant. In the other extreme, if the concentrations of dust are very high,

then dust effects should play a dominant role in determining the transport, deposition and reentrainment behavior of condensable radionuclides.

As discussed in the response to RAI MST-14, HTGR operating experience strongly suggests that dust is generated in pebble bed reactors (as observed in AVR) primarily from abrasion between fuel spheres and the fuel handling system and that prismatic core reactors are expected to generate significantly smaller amounts of dust. Consequently, any dust in the primary circuit of a modular HTGR is likely to be carbonaceous (i.e., having the composition and characteristics of pebble matrix and/or core graphite). However, the presence of friable surface films on metallic components cannot be ruled out completely. In fact, in the case of Fort St. Vrain the small quantity of dust observed in the primary circuit was primarily iron oxide (rust) as a consequent of water ingresses from malfunctioning of the circulator water bearings.

Given that the dust in the primary circuit of a modular HTGR will be predominantly carbonaceous, the relative affinities of such dust compared to that of metal surfaces can be inferred from the relative sorptivities of metals and bulk matrix and graphite. The relative affinities vary with chemical element [1, 2]. For those radionuclides with the greatest radiological consequences for a modular HTGR, the following qualitative behavior is expected. Cesium (an alkali metal) is strongly sorbed on oxidized metal surfaces; it is also strongly sorbed on graphitic materials at primary circuit temperatures. Strontium (an alkaline earth), which is much less volatile than cesium, is strongly sorbed by matrix and graphite and by metal surfaces. Silver (a noble metal) is strongly sorbed on metal surfaces at temperatures below about 600°C. It is also sorbed on matrix and graphite but to lesser extent than strontium or cesium. Iodine (a highly volatile halogen) is sorbed on reduced metal surfaces at temperatures below about 400°C; the iodine sorptivity of oxidized metals and graphitic materials is significantly lower.

References:

1. Hanson, D. L., "Plate-Out Phenomena in Direct-Cycle High Temperature Gas Reactors," 1003387, Electric Power Research Institute, June 2002.

2. Wichner, R. P., "Fission Product Plateout and Liftoff in the MHTGR Primary System: A Review," NUREG/CR-5647 (ORNL/TM-11685), Oak Ridge National Laboratory, April 1991.

RAI MST-106: How much dust is available for liftoff in the event of an accident such as a DLOFC? Is all or part of the deposited dust immobilized by forming a crust? How much is in so-called dead zones?

Comments: There has been a lot of discussion attributing the formation of "crusts" to the introduction of organics, i.e. oil, into the primary. There is also mention about dust accumulating in dead zones. In general, dead zones will likely remain "dead" during an accident, so there will be no re-entrainment from these areas. So I think it is useful to identify these dead zones, and how much dust might be in them.

Response MST-106:

The amount of dust available for liftoff, the amount of dust that might be immobilized by forming a crust such as that observed in the AVR (or by other means), the amount that settles in "dead zones", and the locations of "dead zones" in the primary circuit are all dependent on the specific design of the modular HTGR. After a design selection is made, this information will be provided, as needed, in topical reports in support of the NGNP license application.

RAI MST-107: What is the effect of depressurization during an accident, for instance a DLOFC, on iodine adsorbed on steel and graphite in respect to desorption?

Comments: In C-6.2, there is discussion of re-entrainment of condensable fission products, I'm assuming including iodine. This appears to concentrate on metal surfaces in the primary. I see no discussion in the paper of the effect of depressurization on iodine adsorbed on steel and graphite, namely, it will desorb.

Response MST-107:

The extent to which the small quantity of iodine that is deposited in the primary circuit during normal operation may desorb during depressurization transients is accident scenario-specific. Iodine preferentially deposits during normal operation on metal surfaces in the primary circuit with surface temperatures below about 400°C [1]. For example, in a steam-cycle prismatic fuel modular HTGR, approximately 90% of the iodine plateout is on the evaporator-economizer sections of the steam generator, and less than 1% is deposited on graphite structures. Consequently, the extent to which iodine may desorb during depressurization transients depends upon the time-temperature history of the steam generator tube bundle following the initiation of the transient.

If feedwater flow to the steam generator is maintained after a transient is initiated and the reactor is shutdown, which is the most likely case, the surface temperatures of the tube bundle will decrease, and no iodine desorption will occur. If feedwater flow to the steam generator is terminated, there may be certain scenarios wherein the surface temperatures of the evaporator-economizer sections gradually increase. Under these circumstances, a fraction of the deposited iodine could desorb and be transported to other locations within the primary circuit, depending on the convective flow rate through the steam generator, if any, and the flow path. The desorbed iodine would then resorb when the convecting coolant reached colder locations or locations with low surface concentrations of iodine. Some of the coolant containing the desorbed iodine could egress from the primary circuit into the reactor building. Under these circumstances, the iodine would tend to sorb on the relatively cooler surfaces in the building. In the limit, a fraction of the desorbed iodine could be vented to the environment. Again, the quantity of desorbed iodine that could be released to the environment is accident scenario-specific.

The potential for the desorption of plateout activity was investigated in the COMEDIE BD-1 in-pile loop test, which included four *in situ* loop depressurizations over a range of successively higher shear ratios (the ratio of the wall shear stress during blowdown to that during normal operation) [2]. The first depressurization was done at shear ratio of 0.72 expressly to determine if desorption of plateout activity was a significant contributor to remobilization of plateout activity during depressurization transients. During the almost six hours required to reheat the loop to normal service temperatures with electrical heaters prior to performing the first depressurization test, a fraction of the iodine plateout desorbed and subsequently resorbed at colder locations, but only 0.0077% of the iodine plateout reached the blowdown filters. Moreover, at a peak shear ratio of 5.6, only 0.13% of iodine plateout was re-entrained. The peak shear ratio during a rapid depressurization accident in a modular HTGR is typically less than 1.1. No other deposited radionuclides desorbed during this prolonged reheating period.

References:

1. Hanson, D. L., and A. S. Shenoy, "In-pile Loop Tests to Validate Fission Product Transport Codes," Proceedings HTR2006: 3rd International Topical Meeting on High Temperature Reactor Technology, October 1-4, 2006, Johannesburg, South Africa.

2. Hanson, D. L., "Plate-Out Phenomena in Direct-Cycle High Temperature Gas Reactors," 1003387, Electric Power Research Institute, June 2002.

RAI MST-108: Please clarify the statement about ">95% retention of Kr and Iodine by the kernel

Comments: On p.69, para.2, it talks about >95% retention of Kr and Iodine by the kernel under normal conditions. Hossein Esmaili has calculated about 15% release for Kr-85 at 1200 K under "normal conditions", and I got similar results (17% at 1250 K). This doesn't seem very "retentive", and is not ">95%". Possibly they are referring only to Kr-88 and I-131, which have short halflives, but the writing gives the impression that 95% applies to all Iodine, Xe, and Kr isotopes.

p.72, C-5, describes matrix and graphite as barriers. Diffusion rates in TECDOC-978 for Kr and Iodine are so high that they indicate that the resistance of matrix/graphite is ignored for these gases, so is a barrier only for metals. This is actually stated in Section 4.4.3, bottom of p.33. A further comment: both C-4 and C-5 have titles referring to "Radionuclide Transport in ..." whereas they only refer to metals.

In C-6, reference is made to "condensable fission products" as plating out on the primary surfaces. There is some danger of confusion here, as plateout can be driven by several mechanisms, including condensation, aerosol deposition, and adsorption.

Response MST-108:

The exact quote from page 69 of the Mechanistic Source Terms (MST) White Paper is "Under normal operating conditions, the kernel retains a substantial fraction (>95%) of the radiologically important, short-lived fission gases such as Kr-88 and I-131." The statement refers explicitly to "short-lived fission gases such as Kr-88 and I-131." The fractional releases of fission gases, including iodine isotopes, from coated fuel particles with through coating failures (also referred to as exposed kernels), which are often expressed a release-to-birth rate (R/B) ratio, have been measured experimentally. The R/Bs for fission gases have been determined to be functions of the chemical element, radioactive half life, temperature, and to a lesser extent burnup [1]. Using the correlations for fission gas release from exposed kernels [2], the typical R/B for normal operating conditions is about 0.5% for 2.8-hr Kr-88, and the R/B is about 2% for 8-day I-131. This is the basis for the statement cited in the RAI, which was confined to short-lived fission gases. The fractional releases for long-lived fission gases, such as 10.7-yr Kr-85, are larger, certainly greater than 5%, as indicated in the comments that accompany this RAI.

As subsequently stated in the same paragraph in the Mechanistic Source Terms White Paper, at elevated temperatures the effectiveness of the fuel kernels for retaining fission gases can be reduced. In addition, fractional releases of fission gases from exposed fuel kernels under irradiation can be enhanced if the kernel is hydrolyzed by reaction with water vapor, which may be present in the helium coolant during some off-normal events.

As discussed in Appendix C-2 of the Mechanistic Source Terms White Paper, the two dominant sources of fission gas release from a modular HTGR core are:

- as manufactured heavy-metal contamination, and
- particles whose coatings are defective or fail in service.

Full-core analyses are done with codes such as SURVEY to predict the amount and distribution of fuel particle failure and the subsequent fission gas release. The models used to predict fuel failure and fission gas release from heavy metal contamination and from failed fuel particles are given in [1]. Typically, heavy metal contamination is the dominant source of fission gas release, including iodine release.

Regarding the other matters raised in the comments that accompany this RAI, the following responses are offered. It is correctly noted that no credit is taken for fission gas holdup, including iodines, by the fuel-compact matrix or fuel-element graphite in the calculation of mechanistic source terms. "Plateout" simply refers to radionuclides that have been released from the core and subsequently deposited on surfaces in the primary coolant circuit. No particular deposition mechanism is implied. However, it is noteworthy that the partial pressures of condensable radionuclides in the primary coolant of a modular HTGR are much too low for bulk condensation to occur. In addition, "plateout" does not include in situ neutron activation products, which are limited and fixed in a modular HTGR.

References:

1. Martin, R.C., "Compilation of Fuel Performance and Fission Product Transport Models and Database for MHTGR Design," ORNL/NPR-91/6, Oak Ridge National Laboratory, October 1993.

2. "Fuel Performance and Fission Product Behavior in Gas Cooled Reactors," TECDOC-978, International Atomic Energy Agency, November 1997.

RAI MST-109: There is currently no database to validate the use of the usual LWR systems codes for radionuclide transport in the reactor building. How is INL going to address this?

Comments: Regarding the reactor building, there is discussion (C-7) that the mechanisms for calculating FP transport are the same as for LWRs (true as far as I know), but there is no database to support the calculations for HTGRs, so this is a problem. The source term is now being defined as "release from the reactor building", so this is a critical issue.

Response MST-109:

As discussed in Section C-7 of the Mechanistic Source Terms (MST) White Paper, no direct measurements have been made of radionuclide removal from contaminated helium by condensation, settling, and plateout under the conditions expected in the HTGR reactor building during a core heatup transient. There is an extensive LWR and CANDU database on the behavior of radionuclides in water cooled reactor containment buildings, and major experimental programs are in progress to further characterize the behavior of radionuclides in LWR containment buildings. Overall, based on a review and assessment of these data, the experimental water-cooled reactor database for radionuclide transport in LWR containment buildings appears to be of limited value for refining and independently validating the design methods used to predict radionuclide transport in the modular HTGR reactor building. The bases for this conclusion are discussed in the response to RAI MST-100.

The approach to be taken to develop a database for radionuclide behavior in the reactor building is discussed in the response to RAI MST-102. Additional data on fission product plateout and reentrainment in the helium loop and the reactor building are desired, particularly in the presence of dust and/or moisture. It is currently projected that adequate data can be obtained with laboratory scale and out-of-pile separate effects testing and limited integrated out-of-pile loop testing. Data obtained in these tests will be compared with that obtained in earlier in-pile loop tests (particularly the COMEDIE in-pile loop tests discussed in Appendix C of the Mechanistic Source Terms White Paper) to confirm the adequacy of this approach. A more detailed review of existing data will be conducted, and scaling analyses will be performed to support development of the specifications for an out-of-pile loop. Tracers will be used in the initial testing in this loop, and evaluation of the results of the tracer testing will be used

to establish if additional testing is required using irradiated fuel. Testing using irradiated fuel would likely be conducted in hot cells.

The AGR Fuel Development and Qualification Plan will be updated to reflect this approach at the next regularly scheduled biannual update (late FY12).

RAI MST-110: How many experiments are envisioned to satisfy the objective to fill in the gaps in the fission product transport models -- for all nuclides, all base materials, all temperatures and effect of neutron fluence? For the total amount of experiments totaled up, how well estimated will the error bands be?

Response MST-110:

The DOE AGR Fuel Development and Qualification Program [1] has the mission to develop and qualify fuel for the NGNP Project. Validation of radionuclide source terms is also within the scope of the AGR Fuel Program. Reference [1] is essentially an umbrella document defining the overall scope and schedule for the various planned test programs. More detailed test specifications, test plans, and test procedures are prepared for the individual tests. For example, for the AGR-1 fuel irradiation test, a large number of programmatic documents were prepared which included:

- a fuel product specification defining the required attributes of the test fuel,
- a fuel irradiation test plan,
- a Postirradiation examination (PIE) plan, including a series of post irradiation heating tests.

These test specifications and test plans include test matrices defining in detail the number of tests and the individual measurements to be performed for each phase of AGR-1 test. Such test specifications and test plans will eventually be prepared for all of the programs described in [1]. These documents will become available as the Technology Development Program proceeds.

Two types of tests are needed to characterize TRISO fuel performance and radionuclide transport:

- single-effects tests to generate differential data for deriving improved component models and material property correlations, and
- independent integral tests to confirm the validity of the upgraded design methods. The early emphasis in the AGR Fuel Development and Qualification Program has been on characterizing TRISO fuel performance and radionuclide transport in core materials.

Consequently, the planned test programs designed for this purpose are well defined in [1] and are summarized in Table 5-2 of the Mechanistic Source Terms White Paper. For characterizing TRISO fuel performance, the single-effects tests are AGR-1, AGR-2, and AGR-5/6, and the validation test is AGR-7. For characterizing radionuclide transport in the core, the single-effects tests are AGR-3/4, and the validation test is AGR-8.

The planned test programs for characterizing radionuclide transport in the primary circuit and in the reactor building are discussed in the response to RAI MST-102. As discussed in that response, additional data on fission product plateout and reentrainment in the helium loop and the reactor building are desired, particularly in the presence of dust and/or moisture. It is currently projected that adequate data can be obtained with laboratory scale and out-of-pile separate effects testing and limited integrated out-of-pile loop testing. Data obtained in these tests will be compared with that obtained in earlier in-pile loop tests

(particularly the COMEDIE in-pile loop tests discussed in Appendix C of the Mechanistic Source Terms White Paper) to confirm the adequacy of this approach. A more detailed review of existing data will be conducted, and scaling analyses will be performed to support development of the specifications for an out-of-pile loop. Tracers will be used in the initial testing in this loop, and evaluation of the results of the tracer testing will be used to establish if additional testing is required using irradiated fuel. Testing using irradiated fuel would likely be conducted in hot cells. The AGR Fuel Development and Qualification Plan will be updated to reflect this approach at the next regularly scheduled biannual update (late FY12).

In summary, the exact number of experiments to be performed under the AGR Fuel Development and Qualification Program has not yet been determined. As explained in the response to RAI MST-58, the radionuclides of interest in HTGR design and safety analysis can be broken into classifications for analysis purposes. The properties of the radionuclide classes presented in Table 4-2 of the Mechanistic Source Terms White Paper are such that it is not necessary to collect data on all radionuclide species that are analyzed in the calculation of mechanistic source terms.

With regard to the "error bands" cited in this RAI, an objective in calculating radionuclide source terms is to be able to determine fuel performance and radionuclide transport within a specified uncertainty factor between the 50% and 95% confidence values. This uncertainty factor is typically a factor of four for the release of fission gases from the core and a factor of 10 for the release of fission metals from the core and for radionuclide transport in the primary system and the reactor building. See the response to RAI MST-62 for further discussion of the basis for these uncertainty factors.

A sufficient number of experiments will be conducted to confirm the ability to calculate fuel performance and radionuclide transport and release to within these uncertainty factors.

Reference:

1. Simonds, J., "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," PLN-3636, Rev. 0, Idaho National Laboratory, September 2010.

RAI MST-111: How have the nuclides been selected for transport analysis? Is it assured that the selected nuclides adequately cover the various dose estimate requirements? Should C-14 and tritium be included in your "fission product" listing as special cases? Why are Cs134 and Cs136 omitted on p. 38?

Response MST-111:

It is assumed that the reference cited in this RAI is intended to be Table 4-2 on page 28 of the Mechanistic Source Terms (MST) White Paper. The "Key Nuclides" listed in column 2 of Table 4-2 are only intended to be illustrative of that class of radionuclide; it is not meant to imply that other radionuclides in these classes are neglected as elaborated below.

The response to RAI MST-58 explains why the fuel irradiation and accident condition testing programs will focus on obtaining data for specific species of specific gaseous and metallic fission products and how these limited data will be used to model the transport of all of the radiologically significant species of fission products, thereby assuring that no radiologically significant species are omitted.

As described in Section 2.4 of the MST WP and elaborated in the responses to RAIs MST-87 and MST-62, standard design practice in the U.S. HTGR program has been to define a two-tier set of radionuclide design criteria—referred to as "Maximum Expected" and "Design" criteria. The "Design" criteria are derived from externally imposed requirements, such as site boundary dose limits, occupational exposure

limits, etc. After the "Design" criteria have been derived from the radionuclide control requirements, the corresponding "Maximum Expected" criteria are derived by dividing the "Design" criteria by a design margin to account for uncertainties in the design methods.

An essential element in evaluating the adequacy of a candidate core design for a modular HTGR is to determine whether the design meets these radionuclide design criteria at the required confidence levels. Design methods have been developed over the past four decades to make such an evaluation. These methods are introduced in Appendices B through E of the Mechanistic Source Terms White Paper for both prismatic and pebble-bed core designs. Three decades of modular HTGR design experience has demonstrated that it is sufficient to calculate the core release rates for a few key radionuclides to determine compliance with these radionuclide design criteria. The key radionuclides are Kr-88, I-131, Cs-137, Ag-110m, and Sr-90. Typically, these are also dominant dose contributors.

With regard to carbon-14 and tritium, both these radionuclides are included in the radionuclide design criteria. They received relatively little attention in the Mechanistic Source Terms White Paper because they are insignificant contributors to accident source terms.

The sources and transport behavior of carbon-14 in an HTGR are described in [1], and the standard modeling protocols used in the U.S. HTGR program are described Section 11.1 of [2]. Essentially, carbon-14 is permanently bound in the core graphite and is not released from the core except by graphite oxidation by air or water.

Calculation of an overall plant mass balance for tritium is a standard element of HTGR plant design. A separate white paper on tritium limits in modular HTGRs is in preparation by the NGNP Project and will be submitted to the NRC.

With regard to cesium-134 and cesium-136, they are also included in the radionuclide design criteria. The allowable and calculated core release fractions for 2.1-yr cesium-134 and 13-day cesium-136 (as well as all other cesium isotopes) are typically assumed to be the same as those calculated for 30-yr cesium-137. This assumption is slightly conservative for cesium-134 and very conservative for cesium-136 because decay in transit prior to being released from the core is neglected. Separate full-core calculations could easily be done for cesium-134 and cesium-136 that properly account for their individual half-lives, but such calculations are unnecessary because the more constraining requirement is to demonstrate compliance with the cesium-137 core release limit.

References:

1. Brooks, L. H., et al., "Carbon-14 in the HTGR Fuel Cycle," GA-A13174, General Atomic, November 1974.

2. Stone & Webster Engineering Corp., 1992, "Preliminary Safety Information Document for the Standard MHTGR," HTGR-86024, Rev. 13, September 1992.

RAI MST-112: Why is the direct cycle turbine not listed along with the steam generator as a possible special source?

Response MST-112:

The steam generator was referred to on page 27 of the Mechanistic Source Terms White Paper as a significant source of occupational exposure during facility maintenance for a steam cycle modular HTGR. The steam cycle modular HTGR was cited because it is currently the reference design for the NGNP

Project. For a modular HTGR that uses a direct Brayton cycle energy conversion system with a gas turbine, the steam generator would be replaced by the helium gas turbomachine and other heat exchangers (e.g., a recuperator, a precooler, and an intercooler). In this case, these components would be sources of occupational exposure during maintenance.

RAI MST-113: How do you define "barrier"? How many of the "barriers" listed are truly barriers?

Response MST-113:

The following definition applies in the context of the Mechanistic Source Terms White Paper:

Barrier – A feature that provides resistance to the transport and release of one or more radionuclides.

The definition does not imply impenetrability. The definition does not preclude the possibility that any given barrier may be more effective in retention of some radionuclide species and less effective in the retention of others.

In this context, all of the "barriers" listed in the Mechanistic Source Terms White Paper are truly barriers.

RAI MST-114: Is "plateout" used to signify aerosol deposition, both dust and molecular deposition e.g., sorption on surfaces? Are there separate model for each mechanism? Similarly for liftoff, does that refer only aerodynamic removal of aerosols, or also for desorption? Are there separate models for aerosol deposition and for chemisorptive (or condensation) deposition?

Response MST-114:

As described in response to RAI MST-108, "plateout" simply refers to condensable radionuclides that have been released from the core during normal operation and subsequently deposited on surfaces in the primary coolant circuit. No particular deposition mechanism is implied. However, it is noteworthy that the partial pressures of condensable radionuclides in the primary coolant of a modular HTGR are much too low for bulk condensation to occur. In addition, "plateout" does not include in situ neutron activation products, which are limited and fixed in a modular HTGR.

As discussed in the response to RAI MST-14, HTGR operating experience strongly suggests that dust is generated in pebble bed reactors (as observed in AVR) primarily from abrasion between fuel spheres and the fuel handling system and that prismatic core reactors are expected to generate significantly smaller amounts of dust. Only modest quantities of dust are expected to be present in the primary coolant circuits of future pebble-bed MHTGRs (due to redesign of the fuel handling systems), and insignificant quantities of dust are expected in future prismatic MHTGRs. As indicated in Appendix D of the Mechanistic Source Terms (MST) White Paper, the PADLOC code, which is used by General Atomics to predict plateout in prismatic HTGRs, does not explicitly model aerosol transport. The current assumption is that the dust concentrations in prismatic modular HTGRs will be so low that aerosol transport effects can be neglected (keeping in mind that the accuracy goal for predicting plateout distributions is a factor of 10). As indicated in Appendix E of the Mechanistic Source Terms White Paper, the DAMD code, which is used to predict plateout in pebble-bed HTGRs, includes separate component models for both molecular and aerosol transport.

Typically, "liftoff" is used to refer to mechanical reentrainment during off normal events involving depressurization due to leaks/breaks in the helium pressure boundary and does not include desorption. As indicated in Appendix D of the MST White Paper, the POLO code, which is used by General Atomics for prismatic modular HTGRs, contains an empirical liftoff model, which only addresses mechanical

reentrainment of plateout activity (the so-called shear ratio model). The PADLOC code, which is used by General Atomics to predict plateout distributions in prismatic modular HTGRs, models reversible chemisorption of volatile radionuclides; consequently, it can also model transient desorption of plateout activity. As indicated in Appendix E of the MST White Paper, the DAMD code, which is used to predict radionuclide transport in the primary circuit of pebble-bed modular HTGRs, includes separate component models for predicting both mechanical reentrainment and desorption.

RAI MST-115: Clarify the meanings of "fuel quality", "HM contamination", and "defective particles". Are these terms used consistently? Are HM contamination and fuel particles failures actually reported separately in PIEs, or is this a future intention?

Response MST-115:

The parameters in Table 4-3 are addressed in the response to RAI MST-73. As discussed in the response to RAI MST-73, there have been historical differences in the definition of fuel quality related parameters between the U.S. and German programs. As noted in the supplemental information provided to the NRC in INL letter CCN 223977, the current strategy for fuel acquisition for the NGNP does not involve replication of the German fuel. Thus the clarifications provided below are in the context of fuel fabrication and characterization by the NGNP project.

- *Fuel quality* is a measure of the ability of as-manufactured fuel to retain radionuclides under irradiation as indicated by a combination of fractional quantities of heavy metal (HM) contamination, missing or defective buffers, missing or defective IPyC, defective SiC, and missing or defective OPyC.
- *HM contamination* is the fraction of heavy metal (uranium in the case of NGNP particles) associated with contamination on coated particle surfaces and in the matrix, and in kernels of as-manufactured particles with connected cracks in the IPyC, SiC and OPyC layers.
- *Defective particles* can include particles with missing or defective buffers, missing or defective IPyC, defective SiC, and missing or defective OPyC.

Fuel quality parameters used for quality assurance purposes (fractional quantities of HM contamination, missing or defective buffers, missing or defective IPyC, defective SiC, and missing or defective OPyC) are defined and used consistently for the purposes of measurement and for use in fuel performance codes. The meaning of the term "defective particles" can vary depending on the context of its use.

HM contamination is not measured in PIEs. It is a term associated with as-manufactured fuel. However, deconsolidation and pre-oxidation leachate analyses are conducted on irradiated fuel as a means of determining the fraction of exposed uranium, including failed particles, at the end of irradiation or following heating tests. Fuel particle failures during irradiation can also be determined by on-line gas release analysis and post-irradiation measurement of fission products in the capsule components.

RAI MST-116: Since dust removal is cited as a release mechanism, fission product association with dust must be specified. Where in the White Paper is radioactivity association with dust considered?

Response MST-116:

The Mechanistic Source Terms White Paper includes multiple mentions of dust in the context of its role in fission product transport and release. These are predominantly to be found in Section 4.4 (including

Table 4-2 and Figure 4-3), Section 5 (particularly Table 5-1, which acknowledges the need for additional knowledge regarding dust effects on radionuclide behavior), and Appendix C.6 of the white paper.

RAI MST-117: What are the inputs, principal modeling assumptions, and outputs of each code? Particularly, what is the role of each program? Why does there appear to be duplication?

Response MST-117:

This request for additional information would require a lengthy response for each of the many codes discussed in Figure 4-3 and Appendix D of the Mechanistic Source Terms White Paper. Such a response is beyond the intended scope of the objectives of this white paper. The codes in Figure 4-3 and Appendix D are the codes used only by GA. The organization chosen as the design vendor for the NGNP Project will provide Licensing Topical Reports for its codes to the extent needed to support the associated license application. This will allow the applicant and the NRC staff to focus resources on the codes that will be used in support of the application – possibly the codes in Figure 4-3 and Appendix D, or possibly other codes.

The role of each program referred to in Figure 4-3 and Appendix D is discussed in Appendix D of the white paper. Regarding the appearance of "duplication" noted in the RAI, codes presented in Figure 4-3 are, as noted where the figure is called out on page 35 of the white paper, the principal design codes used by GA for predicting fuel performance and fission product release from a prismatic core during normal operation and core conduction cooldown accidents. As also noted on that page, Appendix D provides a table that summarizes these and other codes used by General Atomics to calculate radionuclide source terms for prismatic fuel HTGRs. The codes in Figure 4-3 are a subset of those in Appendix D. This explains the apparent duplication between Figure 4-3 and Appendix D.

RAI MST-118: Why are Xe135m and Kr88 releases rising with FIMA? What is it in NOBLEG that predicts this?

Response MST-118:

Figures 4-7 and 4-8 are taken from References [1] and [2]. The figures present gaseous fission product releases from two spherical test elements that experienced no particle failure. The releases are therefore determined by the uranium and thorium contamination of the OPyC layers of the coated particles and the matrix material of the fuel element. As explained in Reference 1, during irradiation the fertile uranium-238 and thorium-232 in the matrix material contamination capture neutrons to create fissile plutonium-239, plutonium-241, and uranium-233. Therefore, the effective fissile contamination increases with burn-up, and the release of fission gases from the fuel elements increases as well. The effect is modeled in NOBLEG and contributes to the good comparison between measured and calculated fission gas release for these fuel elements.

References:

1. van der Merwe, J.J, "Development and Validation of Fission Product Release Models and Software at PBMR," 2nd International Topical Meeting on High Temperature Reactor Technology, Beijing, China, September 22-24, 2004, Paper # C18.

2. van der Merwe, J.J., "A Method to Evaluate Fission Gas Release During Irradiation Testing of Spherical Fuel," Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology, September 28–October 1, 2008, Washington, DC, HTR2008-58184