September 21, 2011

CCN 225363
NRC Project #0748

U.S. Nuclear Regulatory Commission
Document Control Desk
Washington, D.C. 20555-0001

SUBJECT: Contract No. DE-AC07-05ID14517 — Next Generation Nuclear Plant Project Submittal — Response to Nuclear Regulatory Commission Request for Additional Information Letter No. 003 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 (RAI) regarding Next Generation Nuclear Plant (NGNP) Project white papers. The attachment contains the NGNP Project's responses for RAIs received in NRC RAI Letter Number 003 (Request for Additional Information No. 5895 Revision 0), dated July 25, 2011. The associated Fuel Qualification and Mechanistic Source Terms 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,

G. A. Gibbs

Greg Gibbs, Project Director
Next Generation Nuclear Plant Project

JK:CN

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Enclosure:

1. NGNP Response to NRC RAI No. 003.

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's. 5771 and 5772, Revision 0), June 7, 2011
d) NRC RAI Letter Number 003 (Request for Additional Information No. 5895, Revision 0), July 25, 2011
e) 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
f) Idaho National Laboratory, Next Generation Nuclear Plant Project Submittal – Response to Nuclear Regulatory Commission Request for Additional Information Letter No. 002 Regarding Next Generation Nuclear Project Fuel Qualification and Mechanistic Source Terms, August 10, 2011, CCN 224915

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Request for Additional Information No. 5895 Revision 0
2nd Set of RAI's

Acronyms

AGC	advanced graphite capsule
AGR	Advanced Gas Reactor
AOOs	anticipated operational occurrences
ATR	Advanced Test Reactor
AVR	Arbeitsgemeinschaft Versuchsreaktor (German)
B&W	Babcock and Wilcox
BDBE	beyond design basis events
BOL	beginning of life
CI	confidence interval
CVD	chemical vapor deposition
DLOFC	depressurized loss of forced circulation
EAB	Exclusion Area Boundary
EOL	end of life
FDL	Fuel Development Laboratory
FIMA	fissions per initial metal atom
FQ	fuel qualification
GA	General Atomics
GDC	General Design Criteria
HFR	High Flux Reactor
HM	heavy metal
HTGR	high temperature gas-cooled reactor
IAEA	International Atomic Energy Agency
iLTI	inner low temperature isotropic - alternative terminology for the inner pyrocarbon coating layer
I-NERI	International Nuclear Energy Research Initiative
INL	Idaho National Laboratory
IPyC	inner pyrocarbon
LBL	leach burn leach

LEU	low-enriched uranium
MCNP	Monte Carlo Neutral Particle computer code
MOL	middle of life
MTR	material test reactors
NGNP	Next Generation Nuclear Plant
NPR	New Production Reactor
NUCO	natural uranium oxycarbide
oLTI	outer low temperature isotropic - alternative terminology for the outer pyrocarbon coating layer
OPyC	outer pyrocarbon
ORNL	Oak Ridge National Laboratory
PAG	protective action guide
PARFUME	Particle Fuel Model computer code
PFP	Pilot Fuel Plant
PIE	post irradiation examination
PMBR	Pebble Bed Modular Reactor
PSID	Preliminary Safety Information Document
PWR	pressurized water reactor
QC	quality control
R/B	release-to-birth rate ratio
RGA	Regulatory Gap Analysis
SA	South African or South Africa
TC	thermocouple
TRISO	tristructural –isotropic

RAI FQ-B1/MST-B1: In stating that the fuel performance is "equivalent to or better than" the German fuel, what are the full set of parameters or figures of merit that are considered (i.e., fuel failure vs. burnup, temperature, varying reaction rates, etc.)? Does the statement mean that all considered figures of merit equal or exceed those of the German fuel?

Comment: The white paper statement noted above is expected to be revised in a future update to the white papers that will reflect the evolving position recently stated by INL [Ref] that the German data will not be used for determining statistical failure rates, but only to provide useful information in determining material properties such as diffusion rates and aid in fuel design (including the calibration of models used in the PARFUME fuel performance code).

Ref: Letter from INL to NRC, CCN 223977, Supplemental Information to the NGNP Fuel Qualification and Mechanistic Source Terms White Papers, May 3, 2011.

Response FQ-B1/MST-B1:
The response to this Request for Additional Information (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-B2/MST-B2: A list of failure mechanisms is given in FQ section 3.1.2, page 19, but no data or images are presented to support these. Are they theoretical or have examples of each failure mechanism been observed in representative fuel particles?

Comment: Given the large number of particles, it does not seem possible to identify most in-pile failure mechanisms. In fuel compacts or pebbles, particle failures are identified by the detection of fission products; however, in most cases, the particle responsible for the release is not identified.

Response FQ-B2/MST-B2:
Fuel failure mechanisms for tristructural–isotropic (TRISO) fuel have been established based on 50 years of testing around the world (Germany, United States, Japan, China, Europe, and Russia). The failure mechanisms identified in the white paper have been observed in irradiation and accident condition testing. Pictures of the mechanisms are shown in Reference 1. In modern high temperature gas-cooled reactor (HTGR) TRISO fuels, changes in fuel specifications (e.g., limits on anisotropy, limits on thin buffers, limits on defects), and/or limits on design service conditions (e.g., time at temperature, temperature gradients, fast fluence) have been used to minimize or eliminate these mechanisms from occurring to the extent practical. Module 7A of the HTGR Training Course provided to the NRC in May 2010 addressed this issue.

Reference:

1. D. A. Petti et al., "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," *Nuclear Engineering and Design*, 222 (2003) pp. 281-297.

RAI FQ-B3/MST-B3: The HM contamination is given as $\sim 10^{-5}$. What are the units on this number? Is it fraction of exposed kernels or particle defects or particle failures? Does it distinguish between enriched UCO/ UO_2 fuel, natural uranium, thorium, etc., and their respective sources and chemical forms?

Response FQ-B3/MST-B3:
Heavy metal contamination is a dimensionless number. It is defined as the ratio of the total amount of the uranium that is not encapsulated by a fission gas retentive coating layer to the total amount of uranium in as-manufactured fuel. Heavy metal contamination is measured by chemical analysis of the leachate from compact deconsolidation and leaching prior to the oxidation step of the leach-burn-leach procedure, and dividing the resulting uranium mass by the total mass of uranium in the compacts. It does not distinguish between enriched and natural uranium or their chemical forms.

RAI FQ-B4/MST-B4: The FQ White Paper states "...fuel particle failures in these irradiation experiments were caused by irradiation induced failure (cracking) of anisotropic IPyC leading to tensile stress intensification in the adjacent well-bonded SiC layer causing subsequent cracking in the SiC layer." Is the failure caused by mode I or mode II crack growth through the SiC? How is this included in the analytical models?

Comment: Materials can have significantly different fracture toughness, crack propagation, and fatigue crack growth for mode I and mode II cracks. From the description of the geometry, whether the crack is mode I or mode II may depend on if the particle layers have debonded. In general, mode II crack growth is not common and cracks will turn to become mode I. However, in some highly constrained geometries, mode II growth can occur. This may have implications for the material test program should accurate measurements of mode II toughness be needed.

Response FQ-B4/MST-B4:
Cracking of the IPyC layer can occur if the tensile stresses developed by irradiation-induced shrinkage exceed the strength of the IPyC layer. Debonding of the layers occurs when the interface is weak and tensile stresses exceed the bond strength. Both have been included in some fuel performance modeling (see References 1, 2 and 3) via detailed finite element modeling. However, these failure mechanisms have been overcome by changes in fabrication parameters by reducing the anisotropy of the IPyC and allowing for the formation of a very strong bond between the IPyC and SiC (see Reference 1). Proving that the changes in fabrication were effective was a key objective of the irradiating Advanced Gas Reactor (AGR) experiment AGR-1. The lack of failures in AGR-1 suggests that the changes were effective. Post irradiation examination (PIE) underway now will confirm this conclusion.

References:

1. D. A. Petti et al., "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," *Nuclear Engineering and Design*, 222 (2003) pp. 281-297.
2. G. K. Miller, D. A. Petti, D. J. Varacalle and J. T. Maki, "Consideration of the Effects on Fuel Particle Behavior from Shrinkage Cracks in the Inner Pyrocarbon Layer," *J. Nucl. Materials*, Vol. 295, 2001, p. 205-212

3. G. K. Miller, D. A. Petti, and J. T. Maki, "Consideration of the Effects of Partial Debonding of the IPyC and Particle Asphericity on TRISO-coated Fuel Behavior," *J. Nuclear Materials,* 334 (2004) 79.

RAI FQ-B5/MST-B5: Some of the properties listed in FQ White Paper Table 1 are path-dependent. This could have implications for testing and modeling. How is this path-dependence accounted for in the models? What is done to ensure that modeling and testing are covering the most severe service conditions for the material? For example, if the pyrocarbon shrinks from exposure to fast fluence and this shrinkage reduces stress in the SiC layer, higher burnup with lower fast fluence may be more limiting than high burnup, high fast fluence conditions. Table 6 shows a higher failure rate at lower burnup and lower fast fluence for heating tests at 1800°C. NGNP operating conditions are even lower burnup. What is done to guarantee that this data is conservative for NGNP?

Comment: Path-dependence could greatly increase the complexity of the modeling to cover all possible conditions of the particles.

Response FQ-B5/MST-B5:
The fuel performance/fuel failure mechanisms listed in Table 1 of the Fuel Qualification White Paper are modeled in fuel performance codes. The models in the codes define fuel performance in terms of the service conditions (e.g., burnup, fast fluence, temperature) upon which fuel performance phenomena are dependent, and calculate fuel performance incrementally (i.e., by time step) as part of the core depletion analysis such that operating history (i.e., path) is taken into account. The fuel irradiations, PIE, and safety testing described in the AGR Technical Program Plan [1] cover a wide range of irradiation and accident conditions that bound the conditions expected for both design options of the NGNP.

The results in Table 6 of the Fuel Qualification White Paper are, by themselves, misleading. This is because two fuel spheres (AVR 74/10 and AVR 70/33) included in the column titled 1,800°C that have low burnup (1.6 and 5.5% fissions per initial heavy metal atom [FIMA]) and account for 57 particle failures are suspected to have been heated to temperatures higher than the intended 1,800°C as a result of fouling of the pyrometer window (see FQ White Paper, Appendix A-1.3.2, pages 122-124). Excluding the results from AVR 74/10 and AVR 70/33 causes the average burnup value to increase and the number of particle failures to decrease. Because of the uncertainty regarding the heating temperatures for the fuel spheres AVR 74/10 and AVR 70/33, the comparison of the results in the two columns in Table 6 at 1,800°C is unreliable, so no conclusions regarding the influence (or lack of influence) of burnup on fuel particle failures at 1,800°C can be made. It should also be recognized that at temperatures ≥1800°C, the main issue is not mechanical performance of the TRISO coating system, but degradation of the SiC coating.

Reference

1. "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," PLN-3636, Rev. 0, INL, September 2010.

RAI FQ-B6/MST-B6: Describe quantitatively hypothetical particle power pulses that, based on currently available information and insights on transient fuel performance, would reasonably be expected to cause TRISO SiC failure fractions in the pulsed particles to increase significantly (e.g. by a factor of 10 or more).

Comment: The requested description of hypothetical particle power pulses should address the transient particle power history and resulting temperature histories of the particle kernels and coatings. While the

hypothetical power pulses considered should distinguish between those that are possible and those that are not physically possible in reactors moderated by graphite, or by graphite and moisture, this question does not concern the likelihood or plausibility of power transient events in any NGNP design, but rather seeks to understand the expected behaviors and failure thresholds of fuel particles in postulated severe transients (e.g., single or multiple rod ejection during or after massive moisture ingress) regardless of their likelihood or plausibility. The identification of power transients to be considered in NGNP licensing or risk analyses is a separate topic that will help determine any needs for transient fuel testing.

Response FQ-B6/MST-B6:
The response of TRISO fuel to power transients is discussed in the response to RAI FQ-33. In summary, power pulse testing (pulses of a few milliseconds duration) results indicate increasing particle failures as the kernel temperature approaches the melting point. The transient conditions of the tests are sufficiently far removed from bounding reactivity transient conditions for either NGNP concept that expenditure of resources to perform detailed thermal and structural analyses of particle response is not considered to be justified.

RAI FQ-B7/MST-B7: With respect to thermal-mechanical fuel performance, describe how the testing that has been or will be performed will show that lifetime power variations in NGNP will not result in TRISO layer separation and fuel failure?

Comment: Thermal-mechanical loadings in NGNP will include those due to load follow, power cycling, fuel sphere cycling, shutdown/restart, prismatic fuel block shuffling, and potential local power oscillations (e.g., as experienced in Fort Saint Vrain). The response should thus address how such loadings are represented in the MTR irradiation cycles for fuel qualification testing.

Response FQ-B7/MST-B7:
Typical materials test reactor (MTR) coated particle fuel irradiations have involved numerous power variations including startup and shutdown for each irradiation cycle, mid-cycle shutdowns, control rod motion, and burnup variations. The operational thermal transients during MTR irradiations are in general much more rapid than would be the case in an NGNP core because of the large thermal inertial of an HTGR core relative to that of a water-cooled MTR irradiation assembly containing coated particle fuel materials.

Extreme power and temperature variations during irradiation were experienced by the large number of spheres loaded and recycled during the Arbeitsgemeinschaft Versuchsreaktor (AVR) operations. Each time a sphere was loaded into the AVR, it was dropped cold onto the top of the up-flow core, where local coolant temperatures in the hotter regions were well above 1100°C during 950°C nominal mixed-mean coolant outlet temperature operation, and the particle power immediately rose to the level corresponding to the thermal flux at the top of the core. No indications of particle failures associated with these large rapid power and temperature variations have been observed in any TRISO particle irradiations in the AVR. Thermal transients experienced by fuel spheres during loading into a pebble bed NGNP would be substantially reduced because of the much lower temperatures at the top of the down-flow core.

Because of the rapid thermal response of the TRISO particles, on the order of tens of milliseconds, as discussed in the response to RAI FQ-33, particle temperature response to operational power variations can be addressed as quasi-static variations with no significant transient thermal stresses.

RAI FQ-B8/MST-B8: Describe how the design analysis of coating layer stresses addresses non-spherical particles.

Comment: Although fabrication techniques discuss the removal of non-spherical particles, there is no discussion of tolerances and the effects of non-sphericity on predicted stress and strain in the coating layers.

Response FQ-B8/MST-B8:
Particle asphericity has been explicitly modeled using PARicle FUel ModEl (PARFUME) and a detailed multidimensional finite element code (Reference 1). Based on this analysis, values for asphericity that resulted in particle failure were established and translated into critical limits in the fuel specification to limit failures from this potential mechanism to values less than 10^{-5}. The allowable aspect ratio (max to min diameter) is a function of the size of the particle. The current specification is that no more than 1% of the particles can have an aspect ratio greater than 1.14 as measured at the SiC layer for UCO fuel and 1.10 for UO_2 fuel. Current TRISO fuels being fabricated in the AGR program are much more spherical than the specification allows and much better than historical German fuel. AGR-2 UCO fuel has a mean aspect ratio of 1.040 with no more than 1% of the particles exceeding 1.068, and no inspected particles observed to exceed the specification limit of 1.14. AGR-2 UO_2 fuel has a mean aspect ratio of 1.035 with no more than 1% of the particles exceeding 1.062, and no inspected particles were observed to exceed the specification limit of 1.10. These values are lower than historical German fuel that had a mean aspect ratio of 1.07. Thus, asphericity is not anticipated to be significant contributor to fuel failure in AGR-2.

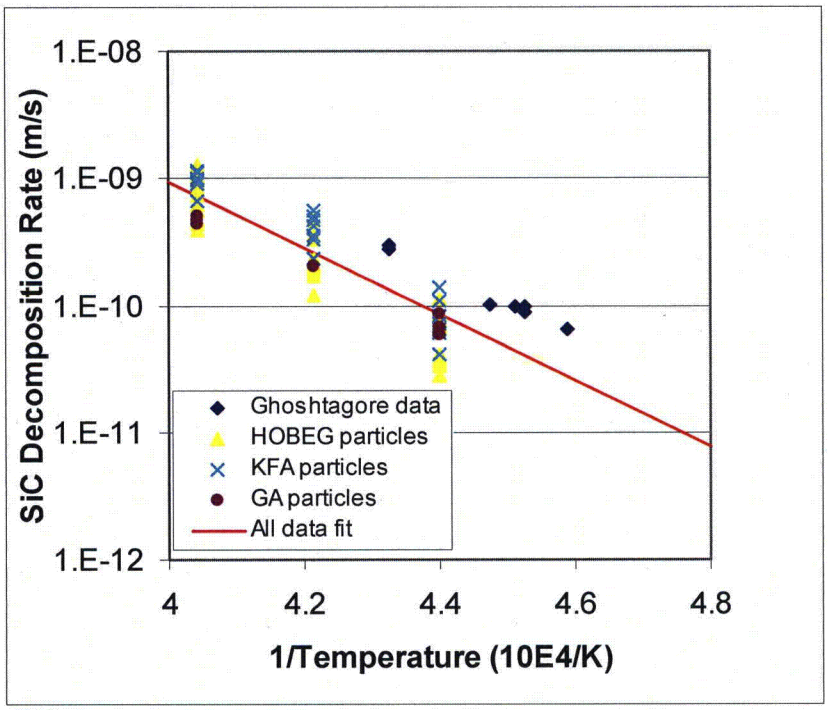
Reference:
1. G. K. Miller, D. A. Petti, and J. T. Maki, "Consideration of the Effects of Partial Debonding of the IPyC and Particle Asphericity on TRISO-coated Fuel Behavior," *J. Nuclear Materials*, Vol. 334, 2004, p. 79.

RAI FQ-B9/MST-B9: Quantify the degree of SiC decomposition that occurs during the 1950°C fabrication annealing of fuel compacts and fuel spheres.

Comment: Based on the temperatures quoted in these two sections it would seem that the SiC in the TRISO particles would start to decompose during the fabrication annealing step.

Response FQ-B9/MST-B9:
SiC thermal decomposition becomes a significant fuel failure mechanism when held for a sufficiently long duration at high temperatures, generally above 1800°C. A thermal decomposition model for PARFUME was developed by fitting all known data to an Arrhenius function, as shown in Figure 1. Each measurement series was conducted in an inert, non-oxidizing atmosphere on exposed SiC. Since SiC that is covered by a layer of pyrolitic carbon decomposes more slowly than exposed SiC [2], correlations based upon these data may be considered to be conservative. Using the data fit correlation for AGR fuel that undergoes a final heat treatment of 1800°C for one hour, SiC is calculated to decompose to a depth of 0.024 μm. For German fuel spheres that experience a 1950°C final heat treatment for one hour, the correlation indicates that SiC decomposes to a depth of 0.17 μm. Realizing that SiC is covered by pyrolitic carbon and graphitic matrix material in both the AGR compacts and German fuel spheres, lesser amounts of SiC is expected to decompose than that calculated above. In fact, there are no reported instances of SiC decomposition observed in as-manufactured heat treated fuel.

Figure 1. SiC Decomposition Rate versus Reciprocal Temperature



References:

1. Development of Improved Models and Designs for Coated-Particle Gas Reactor Fuels, Final Report Under the International Nuclear Energy Research Initiative (I-NERI), INEEL/EXT-05-02615, December 2004.
2. R. J. Price, "Properties of Silicon Carbide for Nuclear Fuel Particle Coatings", Nuc. Tech. 35, September 1977, pp.320-336.

RAI FQ-B10/MST-B10: What is the basis for using Kr 85 to determine that the fission product gaseous source term will be effectively retained in the fuel kernel?

Comment: While the note is true for Kr 85 with a short half life, other fission products may decay to gaseous or high volatile materials that have sufficient half lives to migrate from the kernel; Xe to Cs for example. A more rigorous fission product release treatment should include the production and decay chains of the fission products and their mobility to justify such a white paper conclusion.

Response FQ-B10/MST-B10:
Footnote "h" on page 26 in Section 3.2 of the Fuel Qualification White Paper refers to Kr-85m, which has a half-life of 4.48 hours. By contrast, Kr-85 has a half-life of 10.73 years and is one of the principal noble gas radioisotopes measured in post irradiation safety tests. Codes used for analysis of fission product release incorporate models for production and decay chains of the fission products and their mobility. Noble gas radioisotopes that are measurable under irradiation conditions and under safety testing conditions are used in the evaluation of fission gas releases from fuel.

RAI FQ-B11/MST-B11: The FQ White Paper states that "In HTGR fuel, coated particles provide the main barrier against release of fission products; thus, attention is primarily focused on performance of the coated particles. Although the fuel sphere in the pebble design provides additional fission-product retention through diffusion and trapping and adsorption effects, the principal function of the sphere is to protect the embedded coated particles against external environmental and mechanical effects and to facilitate fuel handling. This means that all irradiation test results obtained on fuel samples containing coated particles of a design similar to that for the pebble-bed design can be included in the experience database when considering particle performance." A similar logic seems to be applied in determining the applicability of fuel tests to TRISO particles in prismatic block fuel compacts and fuel elements. It is not obvious from this explanation why all the fuel samples can be included in the fuel experience database. Please clarify this statement.

Comment: The third sentence of the statement does not seem to be a logical conclusion from the first two sentences. To support such a conclusion, information would also be needed on the effectiveness of the respective fuel forms at protecting the fuel particles over specified ranges of conditions (loadings, temperatures, fluences, chemical attack by water, air, fission products, impurities, etc.).

Response FQ-B11/MST-B11:
The term "fuel performance", as used in the papers, is associated with the structural performance of the fuel particles as determined by the fraction of particle failures. The Fuel Qualification White Paper addresses fuel performance, while the Mechanistic Source Terms White Paper addresses fission product transport in the fuel. The particle failure fraction is indicated by noble gas fission product release during irradiation or during heating tests. Noble gas released from a particle is not retained or significantly delayed by the surrounding matrix in a fuel sphere or compact. Thus, indications of particle failures during prior irradiation or heating tests of particles in spheres, compacts, or loose particles are relevant to the understanding of fuel particle performance, with variations in particle design, fuel form, and irradiation conditions considered as appropriate.

RAI FQ-B12/MST-B12: The FQ White Paper states "coated particles are evenly distributed in this inner, fuel containing region to prevent the development of hot spots in the fuel sphere." Hot spots may develop from clusters of particles in fuel pebbles or fuel compacts and this may be a statistical phenomenon. It is necessary to either ensure that hot spots cannot develop or analyze the hot spots to ensure that they do not impact performance. The same comment applies to other statistical anomalies such as a cluster of defective particles or of spheres with low burnup. How are these statistical anomalies analyzed?

Response FQ-B12/MST-B12:
In the compacting process used by General Atomics (GA) to make fuel compacts for the Fort St. Vrain HTGR, fissile and fertile fuel particles (two sizes of each) and graphite shim particles were all metered into the mold cavities and then a liquefied thermoplastic matrix was injected into the mold to fill the interstitial voids between the particles. The use of different particle sizes and the matrix injection process created the potential for segregation of particles and matrix, although significant inhomogeneity was rarely observed.

In the fabrication process developed by the NGNP/AGR Fuel Development and Qualification Program for prismatic HTGR fuel compacts, only overcoated particles are loaded into the mold cavities. All of the particles are of a single particle type, and all the matrix material used to bind the particles together in a compact is included in the overcoatings on the particles. The hot pressing process causes the matrix to soften and flow to fill the interstitial voids between the particles, and although the compacting process

involves a reduction in the length of the particle bed, it does not involve a significant redistribution of the fuel particles within the mold. Thus, there is no potential for obtaining significant inhomogeneity in the distribution of fuel particles within a fuel compact that could result in a localized hot spot.

With respect to a cluster of as-manufactured defective particles within a prismatic fuel compact, the statistical probability of such an occurrence is extremely low given the very low defect fraction specifications for the fuel particles, which effectively allow less than one defective particle per fuel compact. The only credible mechanism that might result in a cluster of defective particles would be some event during compact fabrication that causes localized fuel damage. However, such a mechanism in the fuel compacting process is precluded because the very low defective SiC and heavy metal contamination specifications applied to the fuel compacts require a compacting process that does not damage fuel particle coatings.

RAI FQ-B13/MST-B13: The FQ White Paper states "The release of silver from intact particles was observed at all heating-test temperatures, with the rate of release increasing substantially between 1600and 1800°C. Cesium was seen to be effectively retained in the intact particles at 1600°C and for the first 100 h at 1700°C, but intact particles dominated the cesium release at 1800°C. Strontium was retained within the sphere at 1600 and 1700°C but was released at 1800°C." Are such releases of Ag, Cs, and Sr considered "particle failures" when the particles are intact?

Comment: Any release of fission products is a challenge for the fuel system. Intact particles are not providing a barrier if through wall diffusion of any of the fission products is occurring. More clarity may be needed in describing fuel performance in terms that distinguish between the mechanical integrity of kernels and coating layers (e.g., intact versus failed) and the functional retentiveness of intact kernels and coating layers.

Response FQ-B13/MST-B13:
As discussed in the response to RAI FQ-24/MST-29, diffusion/release of certain metallic fission products from intact particles is a known phenomenon, and the analysis methods used to calculate mechanistic source terms for the HTGR take into account this phenomenon as well as fission product release from degraded (or functionally failed) and totally failed fuel particles under both normal operating and accident conditions. The response to RAI FQ-13/MST-18 provides a discussion of each of the metallic fission product elements that can diffuse through intact particle coatings at various temperatures.

Diffusion/release of certain metallic fission products from the coating layers of intact fuel particles (either at predicted or higher than predicted operating temperatures) is not considered to be representative of particle failure because the particle coatings are intact and have not become degraded. The majority of fission products, including noble gases and halogens, are retained under the conditions at which diffusion/release of certain metallic fission products from intact particles takes place.

RAI FQ-B14/MST-B14: What justification can be given for the statement that " ...all irradiation test results obtained on fuel samples containing coated particles of a design similar to that for the pebble-bed design can be included in the experience database when considering particle performance" ?

Comment: Although the FQ White Paper makes an argument that examination of the German fuel has resulted in identifying needed changes to the US fabrication process, a possible way to approach proof positive that all important dissimilarities have been corrected would be to show that for all testing the US fuel gives the same results as the German tests. If that is so, it should be addressed, possibly by reference to another section of the FQ White Paper, to substantiate the claim of "similarity."

Certainly even the German fuel has evolved, and fabrication techniques employed are very important in demonstrated performance. Fuel performance has so many synergistic effects that implying fuel performance equivalence by claiming "similarity" may be a questionable use of the term.

Response FQ-B14/MST-B14:
See the response to RAI FQ-B11/MST-11. A primary objective of the NGNP AGR fuel development and qualification program [1] is the demonstration of fuel performance meeting the NGNP requirements without reference to prior testing.

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

RAI FQ-B15/MST-B15: With regard to checking for monodispersity in spherical shape, is testing performed at each stage of particle coating?

Comment: It is not clear from the FQ White Paper whether the monodispersity test is done for each CVD step.

Response FQ-B15/MST-B15:
Sphericity specifications exist for kernels and for particles at the SiC and OPyC layers. Kernels are tested for size and shape prior to coating. A final sample is obtained after OPyC coating, and sphericity is measured on the OPyC layer and on SiC after the OPyC layer is burned off to check compliance with the specification.

RAI FQ-B16/MST-B16: How is the desired degree of uniformity in the random distribution of particles in a fuel compact or fuel pebble determined, specified, and verified?

Response FQ-B16/MST-B16:
With regard to pebble fuel, the response to this Request for Additional Information (RAI) will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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.

With regard to prismatic fuel, historically, GA imposed a uranium homogeneity specification on prismatic fuel compacts made for the Fort St. Vrain HTGR using its thermoplastic matrix injection process to control the homogeneity of the uranium within the fuel compacts. Each half of randomly selected compacts was analyzed for U-235 content by gamma spectroscopy, and the compact was not allowed to have more than a specified fraction of the specified compact U-235 loading in either half of the compact to be acceptable. For the reasons discussed below and in the response to RAI FQ-B12/MST-B12, such a specification on prismatic fuel compacts is unnecessary for compacts made using the compacting process developed by the NGNP/AGR Fuel Development and Qualification Program.

Proper spacing of particles is achieved inherently in the fabrication process. Currently, a very uniform overcoating of finely jet-milled matrix material and water is applied to the particles using a fluidized bed overcoating system adapted from the pharmaceutical industry. The process rapidly makes very

consistent, highly spherical overcoatings with sufficient strength for material handling. The mass of matrix is applied in a well-controlled and programmable manner where the amount of matrix applied to the particle is set to obtain the desired packing fraction. Process yields are essentially 100%, allowing direct feed of the overcoated particles into the compactor after drying and allowing elimination of sieving, tabling, or riffle splitting. Overcoated particles are very spherical based on cam sizing measurements after the overcoating step. The overcoated particles are then loaded into a compacting die, heated, and pressed to form the fuel compact with minimal redistribution of the particles within the die. This is a major improvement over the historical rotary drum overcoating process, for which a broader distribution of overcoating thicknesses was obtained.

RAI FQ-B17/MST-B17: Explain why some of data entries in Table 7 are blank.

Response FQ-B17/MST-B17:
The blank cells in the table correspond to temperature for fuel spheres irradiated in the AVR. The temperature of individual fuel spheres was not measured and would vary as a function of radial, azimuthal and axial position in the core during each pass. Since the temperature is not known, the cells were left blank. Following resolution of the requests for additional information associated with the Fuel Qualification White Paper, the paper will be updated to clarify this matter.

RAI FQ-B18/MST-B18: It is stated that "Fuel-performance models would predict that the isothermal testing should be more challenging for the fuel..." Please elaborate on how this statement is justified and applied, as much of the experimental data is said to be isothermal.

Comment: It is not clear that all failure modes are bounded by isothermal testing. Relevant failure modes and the corresponding bounding isothermal temperatures should be given and justified. Furthermore, it is not clear that test conditions are truly isothermal, given that MTRs may have frequent shutdown cycles for refueling, etc.

Response FQ-B18/MST-B18:
The referenced statement is in the context of post irradiation heating tests, comparing isothermal tests to tests that simulate the time dependence of a depressurized loss of forced circulation (DLOFC) with a peak temperature approximately the same as the isothermal test temperature. Fuel performance, as defined and used in the fuel qualification white paper, is expressed in terms of particle failure fraction. The particle failure mechanisms listed in Table 1 that are relevant to post-irradiation heating include pressure vessel failure from additional diffusive release of gases from the kernel and/or formation of CO, diffusive release through intact layers, chemical attack of the SiC from fission products or CO, and SiC thermal decomposition.

These failure mechanisms are driven by diffusion and chemical reaction mechanisms. In general, fuel performance models address diffusion and chemical reaction mechanisms with rate constants expressed as Arrhenius functions of temperature, with the magnitude of the effects increasing with increasing time at elevated temperature. Since the isothermal tests have a considerably higher integrated time at temperature than transient simulation tests with a comparable maximum temperature, the models would predict stronger effects of the diffusion and chemical reaction mechanisms for the isothermal tests.

RAI FQ-B19/MST-B19: Are the results presented in FQ Figures 7 and 8 characteristic of one or more particle failures in a sphere? How do Figures 7 and 8 support the conclusion that there were defective coated particles?

Comment: The release rate to birth rate ratio (R/B) values for Kr and Xe do not identify which isotopes of these nobles are being considered, and therefore it is difficult to give a meaning to the half-life.

Response FQ-B19/MST-B19:
The material in Section 3.2.4 of the Fuel Qualification White Paper summarizes information provided in more detail in Reference 38 of the white paper. As discussed in Reference 38, after investigating the fission gas release during irradiation the German program concluded that fuel sphere HFR-K6/4 contained two as-manufactured defective particles. The results presented in Figures 7 and 8 are intended to illustrate the ability of the NOBLEG code to calculate release from defective particles as a function of half life for krypton and xenon. The Kr isotopes for which data are reported, in order of increasing half-life (hours) are: Kr-89 (5.3×10^{-2}), Kr-87 (1.27), Kr-88 (2.84) and Kr-85m (4.48). The Xe isotopes for which data are reported, in order of increasing half-life (hours) are: Xe-137 (6.36×10^{-2}), Xe-138 (0.235), Xe-135m (0.255), Xe-135 (effective half-life 1.2) and Xe-133 (126). Note that the effective half-life of Xe-135 during irradiation is substantially reduced from the isotope half-life of 9.14 hours by neutron absorption, because of exceptionally large thermal neutron absorption cross section.

RAI FQ-B20/MST-B20: With once through coating how are QC/QA assured at each coating step?

Response FQ-B20/MST-B20:
Hot samples are pulled during the TRISO coating process after buffer and IPyC layers to allow these layers to be characterized. This is accomplished by lowering a sampling cup into the particle bed at temperature after a coating layer is completed. Another sample is taken after OPyC deposition. Mass flow controllers regulate the flow of gases during each deposition stage, and temperature is monitored and controlled with a thermocouple. TRISO fuel particles are sieved after coating to remove any undersized or oversized particles. This helps ensure that reject particles (bare kernels, particles with missing layers) are not passed along with good product. Samples of the coating layers from coated particles are measured for density in a density column to confirm that IPyC, SiC, and OPyC layers are within acceptable density specifications. Layer thickness are also measured from particle cross sections. TRISO fuel samples are also tested by leach-burn-leach (LBL) to analyze for number of defective particles; any batch that does not pass at or below the specification value is rejected. Sample size for LBL is currently 50K to 120K particles.

RAI FQ-B21/MST-B21: In the thermosetting-matrix-based compacting process, why is it important to assure monodispersity? Are the TRISO particles still checked for monodispersity before overcoating? Does the overcoating material have significant property differences from the "additional" matrix material that could result in discontinuities during the fuel's lifetime? How thick is the overcoating, and how does this affect homogenization of the active fuel region in the compact and the volume fraction of the "additional" matrix material?

Comment: The prismatic fuel is going to switch to a thermosetting process using resin and overcoating of the TRISO particles versus the thermoplastic process used for FSV. The question is asked because it is not obvious why the coated particles need to attain monodispersity. This would be more important than checking this parameter after overcoating. Or, is it a concern that the thermosetting process will upset the previously assured (QC'd) particle monodispersity?

Also, the overcoating comprises another particle layer that could result in compatibility concerns with the matrix material, especially in terms of separation from the particle or matrix material.

Response FQ-B21/MST-B21:
Compaction with the automated press at Babcock and Wilcox (B&W) utilizes a volumetric feed system to automatically load a charge weight of overcoated product into the press die cavity. Consistent charge weights can be achieved with a monodisperse overcoated material as has been demonstrated in compaction scale up testing where weight standard deviation on green compacts is less than 0.03 grams. More consistent weight per compact provides a higher consistency in uranium loading per compact and more consistent lengths of heat treated compacts such that the length range is within the anticipated specification. No additional matrix is added during the compacting step. Matrix is only added to the particles at the overcoating stage.

Current activities measure particle size on TRISO-coated fuel particles prior to overcoating. This information is needed to determine the proper amount of matrix to achieve the desired packing fraction. The TRISO particle size is checked on each composite lot.

Overcoating thickness will be dependent on the packing fraction in the fuel compact. Overcoating thicknesses are very uniform ensuring a good uniform distribution in the compact.

RAI FQ-B22/MST-B22: There appears to be little true transient testing of the TRISO particle. Have the temperature ramps of up to 190°C/hour mentioned been qualified against the accident envelope?

Comment: Although it is discussed that non-isothermal testing has been performed, and that further testing will be required, the past testing may not be indicative of the rapid heatups associated with reactivity excursions, and there is a concern that the FQ White Paper seems to have unduly discounted such excursions.

Response FQ-B22/MST-B22:
As discussed in Section 3.3.3 of the Fuel Qualification White Paper, the heating rates noted were representative of those expected in the large HTGR designs under consideration at the time. The planned heating tests for NGNP fuel will include simulation tests qualified against the accident envelope. As discussed in the response to RAI FQ-33, there are no heat-ups associated with reactivity excursion transients that could be considered rapid relative to the fuel particle thermal time constant, and the thermal conditions relevant to fuel performance (time at temperature) associated with reactivity excursion events are bounded by the DLOFC event in terms of effect on fuel particle integrity.

RAI FQ-B23/MST-B23: What are the justifications for assuming that the two concluding statements (numbered) at the end of Section 3.3.3 are applicable to the technical bases for the NGNP fuel design?

Comment: A stronger argument should be made about the acceptability of the German or other non-UCO transient tests to validate the proposed NGNP TRISO fuel design. In particular, the fuel design has been evolving with respect to not only the kernel material but the coating process and compact or sphere formation. It is difficult to keep track of the variations with respect to the different tests offered as representative. Maybe a detailed table would help, where known differences amongst the fuels undergoing tests are clearly laid out. There may be too many variable parameters that affect fuel performance and their synergistic effect to give full credit to any fuel test, which does not fully replicate the material content and fabrication process, even to the extent of requiring that the samples come from the same fabrication facility.

Response FQ-B23/MST-B23:
The two numbered statements at the end of Section 3.3.3 on pages 46 and 47 of the Fuel Qualification White Paper are justifications for the conclusion that safety testing data for German UO_2 fuel are generally applicable to the UCO fuel being developed in the NGNP/AGR Fuel Program. As discussed in Section 3.3.3 of the FQ White Paper, the relatively few safety testing data on UCO fuel and the numerous data on UO_2 fuel indicate that kernel composition does not have a large effect on fission product release under accident conditions. Overall, the data suggest that the performance of the SiC coating is the controlling factor for fuel performance under accident conditions and is effectively independent of the kernel composition. However, it is important to understand that the NGNP/AGR Fuel Development and Qualification Program has extensive safety testing planned and, although comparisons will be made with previous results on UO_2 fuel, the Program will rely on its own safety test data and will not rely on previous results of German safety testing of UO_2 fuel to demonstrate satisfactory fuel performance.

RAI FQ-B24/MST-B24: How well have the analytical methods (fuel performance codes) been able to predict blind test data, and what data has been used to verify or tune the codes?

Comment: Experience has shown that unless the fabrication tolerances are very tight, there can be a large uncertainty in a fuel code's prediction. This is especially true here as the commercial fabrication facility for the NFNR has yet to come to fruition.

Response FQ-B24/MST-B24:
A large number of fuel performance codes from around the world were recently subject to a large benchmarking activity under the auspices of the International Atomic Energy Agency (IAEA) [1]. The benchmarks include comparisons against TRISO fuel performance data under both normal operation and accident conditions. The benchmarking activity concluded that the current predictive capability is limited by uncertainties in material properties.

There are a number of fuel performance codes in use by high temperature gas cooled reactor vendors. All of the codes are essentially built on the world international database as described in [2]. In the NGNP/AGR Fuel Development and Qualification Program, the AGR-7 irradiation and follow on safety tests will be used to provide data for validation of fuel performance models. Also, as noted in the response to RAI FQ-26/MST-31, it is anticipated that a fuel and fission product surveillance program will be conducted during initial operation of NGNP that will provide additional data to demonstrate that the fuel is performing as predicted by the codes.

With regard to the comment that accompanies this RAI; tight fabrication specifications are applied to all critical parameters, but are not needed on all of the attributes of TRISO fuel. The fuel performance modeling to support development of the specification has identified that many of the specifications when evaluated over a wide range do not strongly affect the failure probability of TRISO particles. Results of a detailed analysis [3] suggest that the three most important specifications for TRISO coating thermomechanical performance are buffer thickness, sphericity, and anisotropy of the IPyC layer followed by SiC and IPyC thicknesses. OPyC properties, SiC density and IPyC density had little impact on predicted fuel performance. (See response to RAI FQ-B52/MST-B52 for more details.) Current U.S. fabrication at industrial scale has consistently met these specifications at 95% confidence. Today the improved measurement technology and automatic control of the CVD coating process is producing material with lower population standard deviations than in the prior generation of TRISO fuel. These well-characterized data sets remove potential sources of uncertainties in fuel performance predictions. Actual data from AGR-2 is presented in the response to RAI FQ-B63/MST-B63.

References:
1. IAEA, High Temperature Gas Cooled Reactor Fuels and Materials, IAEA-TECDOC-CD-1645, March 2010.
2. G. K. Miller, D. A. Petti, J. T. Maki, D. L. Knudson, "PARFUME Theory and Model Basis Report," INL/EXT-08-14497 September 2009.
3. D. A. Petti, "NGNP/VHTR Technology Development," presentation to DOE Nuclear Energy Advisory Committee, April 2011.

RAI FQ-B25/MST-B25: How does the "sphere center temperature" or the compact center temperature relate to the kernel temperature and failure data?

Comment: The kernel temperatures may be different from the sphere or compact center temperatures.

Response FQ-B25/MST-B25:
The sphere center temperature is generally defined as the matrix temperature at the center of the sphere. The difference between the maximum particle kernel temperature and the matrix temperature at the center of a sphere or compact depends on the power that is generated during the irradiation. Analyses performed with the matrix assumed to be held at a constant 1000°C regardless of power [1] indicate that a 350 μm diameter UCO fuel particle, operating at a typical prismatic power of 25 mW/particle, experiences a 25°C temperature rise from the outer PyC layer to the kernel center. Increasing the power to 75 mW/particle, corresponding to an acceleration factor of 3, the temperature rise is 100°C. For a 500 μm UO_2 fuel particle, operating at a typical pebble bed power of 50 mW/particle, the temperature rise from the outer PyC layer to kernel center is 20°C. Increasing the power to 150 mW/particle, for an acceleration factor of 3, results in a 60°C temperature rise. These calculations indicate that kernel center temperatures, depending upon fuel design, can be 60 to 100°C higher than the matrix when the particle power is three times that expected in a HTGR. At very high powers corresponding to highly accelerated irradiations kernel temperatures are calculated to be 150 to 300°C higher than the fuel element center temperature. The impact of these higher temperatures is examined in Reference 1 and concluded in Reference 2 to have probably contributed to the higher failure rates observed in historical very accelerated U.S. irradiation testing when compared to German irradiation testing. The U.S. program now has a limit on the particle power during the irradiation to ensure that kernel temperatures are not significantly greater than that expected in an HTGR. The difference between kernel temperature and matrix temperature is small for typical HTGR conditions.

References:
1. 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*, Vol. 371, 2007, pp. 270–280.
2. D. A. Petti, et al., "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," *Nuclear Engineering and Design*, Vol. 222, 2003, pp. 281–297

RAI FQ-B26/MST-B26: The FQ White Paper states that "the specified value of 6×10^{-5} for a single fuel lot will be used as the design value for the free-uranium fraction in the core for establishing the relationship between failure fraction and temperature described in Section 4.2.3.4" However, if more manufacturing defects are assumed, fewer particle failures are estimated. Can it be shown that this is a conservative assumption?

Response FQ-B26/MST-B26:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B27/MST-B27: What is the fuel time constant for heat transfer?
Comment: To better understand the discussion on transient response, it would be beneficial to know what fuel time constant is assumed for transients and accidents.

Response FQ-B27/MST-B27:
Analyses of transients and accidents will be conducted with thermal models incorporating a level of detail appropriate to the event being analyzed. The analyses will be based on material properties, not an assumed time constant. As discussed in the response to RAI FQ-33, the coated particle thermal time constant for the German particle properties is calculated to be approximately 40 ms. Transient thermal stresses within the particles could become significant for transients with time frames less than ~100 ms.

Variations in local fuel to coolant heat transfer geometry within a prismatic fuel assembly limit the applicability of a macroscopic thermal time constant. An effective time constant for a sphere with a homogenized fueled region varies with temperature over a range of about 15 to 30 seconds (decreasing with temperature primarily because of increasing matrix thermal conductivity). As discussed in the response to RAI FQ-B7/MST-B7, the ability of the spheres to withstand large rapid thermal transients was demonstrated during operation of the AVR.

RAI FQ-B28/MST-B28: What consideration has been given to possible discontinuities or physical separation between the TRISO particles and the matrix material with irradiation?

Comment: Load following, shutdowns, fuel shuffling, and sphere cycling can cause temperatures and reaction rates to vary greatly over time. Differences in physical properties between the matrix material and particles may result in the development of a void, gap, or other discontinuities. It is not clear whether the temperatures are enough to result in recrystallization, but radiation enhanced creep or radiation induced growth or shrinkage may also contribute to the concern.

Response B28/MST-B28:
Physical separation between fuel particles and the surrounding matrix has typically not been observed in the PIE of historical irradiations of TRISO particle fuels with matrix similar to German A3-3 or A3-27 matrix or in the ongoing PIE of fuel from the AGR-1 test. Therefore, no consideration of possible discontinuities between TRISO fuel particles and matrix material during irradiation has been given. A narrow discontinuity would have a minor impact on fuel temperatures and therefore fuel performance. Typically, irradiations in materials test reactors, such as the Advanced Test Reactor (ATR), involve numerous shutdowns and restarts with heating and cooling rates many times greater than would be experienced in a modular high temperature gas-cooled reactor, such as the one being envisioned by the NGNP Project.

RAI FQ-B29/MST-B29: What stresses on the spheres results from thermal expansion? Is there sufficient free volume in the core to accommodate expansion and no risk of pinning to cause stress on the spheres?

Comment: There appears to be no discussion on any effect that might prevent the free motion of the spheres. Nor is there a discussion of any mechanism that might contribute to stress on the spheres except for a mention that the authors are satisfied with the free drop test conducted on the spheres by the Germans.

Response FQ-B29/MST-B29:
The reactor cavity is sized for operating (hot) conditions and includes a gas plenum between the top reflector and the top of the pebble bed, including the conus that develops as fuel is dropped onto the bed. The volumetric coefficient of thermal expansion of the pebble matrix material is about 3.E-6/K [1] so that even though the vessel restricts all expansion to the axial direction, the difference in volume pebble bed between hot operation and cold shutdown is less than 1%. This translates into a height difference of 2 to 3 cm, much smaller than the void space itself. The anisotropy in thermal expansion is limited to 1.3.

The free motion of spheres is impeded by wall friction and friction between the pebbles. Wall friction creates a drag on the pebble flow so that the pebbles near the walls move more slowly than those in the middle of the pebble bed cylinder or annulus. This behavior was studied extensively in the early years of the German HTR program and simulated more recently using discrete element codes such as PFC^{3D} and PEBBLES. The instrument/control 'noses' in the AVR contributed significantly to the non-uniformity in the pebble velocity profile [2]. Simulations performed as part of this study indicate ratios of between 5:1 and 9:1 between the fastest and slowest pebbles in cores without noses. None of the candidate NGNP pebble bed designs feature noses so that this velocity profile is expected.

The drag force on the pebbles at the wall will support the pebble bed thereby relieving the pebbles in the lower region of the full weight of the upper pebbles. A single column of pebbles in the PBMR-400 (height = 1100 cm) contains 183 pebbles with a weight of $183*0.202kg/pebble*9.8m/s^2 = 365$ N. The maximum wall contact force on a pebble in the HTR-10 reactor was computed with the PEBBLES code to be about 50 N [2]. Both of these values are well below the specified breaking load of the element of ≥18000N [3].

'Bridging' of pebbles in a discharge chute has been observed only in chutes that are less than 8 pebble diameters across [4]. None of the proposed NGNP pebble bed designs feature chutes with diameters less than 8 pebbles.

References:
1. H. Nickel, H. Nabielek, G. Pott, A. W. Mehner, "Long time experience with the development of HTR fuel elements in Germany," *Nuclear Engineering and Design*, 217 (2002).
2. H. F. Niessen, PFC3D Validation with Experimental results: AVR as a complex test, ANABEK, FZJ-ISR-RC-5112, June 2009.
3. J. J. Cogliati and A. M. Ougouag, "Pebble bed reactor dust production model," *Proceedings of the 4th International Topical Meeting on High Temperature Reactor Technology, 2008*, Washington, D.C., U.S., September 28 to October 1.
4. Q. X. Liang, H. Jun, G. Guiju, Z. Kaifen, and Z. S. Xi, *Development of the Graphite Matrix Materials of the Fuel Elements for the 10MW HTR Test Module*, INET, Tsing Hua University, Beijing China.
5. C.-B. von der Decken, "Mechanical Problems of a Pebble Bed Reactor," *Nuclear Eng. Design*, Vol. 18, 1972, pp. 323–333.

RAI FQ-B30/MST-B30: What is the justification for claiming a depressurization accident is more limiting than a reactivity excursion accident? What is the justification for using point kinetics to model reactivity excursion accidents?

Comment: The FQ White Paper continues to take the position that a depressurization accident is the limiting condition. While this may be further explained in another document, it is not covered in the FQ White Paper to be sufficiently convincing. This accident would appear to establish the limiting temperatures that the fuel would need to be tested against. It is understood that in the pebble bed design, a large thermal capacity and limited excess reactivity are safety features. Still, the discussions on reactivity excursions for pebble bed and prismatic designs are unconvincing, especially when point kinetics are discussed as the modeling technique.

Response FQ-B30/MST-B30:
The basis for the DLOFC being more limiting on fuel performance than reactivity excursions is provided in the response to RAI FQ-33. As noted in Section 5.3.1 of the Fuel Qualification White Paper, a point kinetics model with a 2D thermal model was used in a 1987 analysis of specific beyond licensing basis events for the standard MHTGR. The justification for reactor kinetics and thermal models to be used for analysis of licensing basis events applicable to the NGNP will be addressed in future submittals in the course of the license application review.

RAI FQ-B31/MST-B31: No rod ejection accident is considered for these reactors. However, an inspection of the illustration showing the prismatic version of the core (Fig. 25) shows, the control rod drive elements are enclosed in housings, which operate at system pressure but are still vulnerable to failure (similar to a control rod drive flange failure in a PWR). How is the fuel expected to react to a sudden increase in power brought about by the addition of over one dollar of reactivity? Bear in mind that the initial power pulse is terminated by the Doppler effect alone. Are any pulsed power experiments planned for the fuel?

Comment: Analysis of a rod ejection accident is required for all current LWR designs. If this reactor design has advantages over competing concepts because of its particular kinetic parameters, then this will become clear in the course of the analysis. The behavior of the fuel will be important in this case.

Response FQ-B31/MST-B31:
The response of the fuel to reactivity excursions is addressed in the response to RAI FQ-33. For reasons discussed in that response, no pulsed power experiments are planned for the fuel. Additional discussion on the likelihood of a control rod ejection is provided in the response to RAI FQ-B32/MST-B32.

RAI FQ-B32/MST-B32: What are the technical bases for assuming that control rod ejection is not credible, and why do such reactivity insertions appear to be discounted?

Comment: For LWRs, control rod protection systems also exist, but design basis accidents include consideration of possible failure.

Response FQ-B32/MST-B32:
The NGNP control rod drive and housing design is expected to be similar to the design used in the MHTGR. The basis for assigning a low probability to a control rod ejection is summarized on page 6-16 of the MHTGR PRA [1] as follows:

The concern with a more rapid reactivity insertion is that core power may significantly overshoot the power levels reached in the slower withdrawal event before the negative temperature coefficient counteracts the overpower condition. Even though the overpower condition is brief, if more energy is deposited in the fuel than in the withdrawal case, higher fuel temperatures may be reached. The most rapid potential for reactivity insertion into the core may be associated with a control rod ejection. The likelihood of such an event in the MHTGR is extremely remote. The event would first require the total failure of the Class 1 vessel penetration, which houses the individual rod drives. Secondly, the ejection would have to be energetic enough to force the rods through the structures located above the rod housings. In particular, they would have to penetrate the fairly massive refueling floor above the upper vessel. And finally, a failure of the other rods or reserve shutdown control equipment (RSCE) to be inserted must be postulated.

In an LWR the control and safety rods are rigid, and the rod within 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 of 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 from an HTGR like the NGNP.

Fundamental differences with LWRs that lead to mitigation of the effects of reactivity excursions are further addressed in the response to RAI FQ-33.

Reference:

1. DOE, "Probabilistic Risk Assessment for the Standard Modular High Temperature Gas-Cooled Reactor," DOE-HTGR-86011, Rev. 5, Vol. 1, April 1988.

RAI FQ-B33/MST-B33: The approach of establishing limits for fuel quality does not seem conservative. Why not make the best fuel that is economical and have a larger margin?

Comment: It may be useful to consider a cost/benefit analysis of the specifications in addition to the top-level requirements. The fuel quality requirements will be given separately for contamination, coating defects and probably other variables. The interaction of these variables can be complex under operating and accident scenarios. The logic for deriving the fuel requirements is sound. However, for example, if increasing coating quality can be done at relatively low cost, it could be included in the fuel specification to increase the safety margin.

Response FQ-B33/MST-B33:
Limits on fuel quality are only one of many design selections that contribute to the safety capabilities of the modular HTGR, which is designed to provide safety margins that substantially exceed those available from other reactor technologies. It can be misleading to focus on one aspect of the design and thereby to conclude that it may not be conservative. All of the design trade-offs and selections that determine the safety characteristics of the modular HTGR and the cumulative effect of those design selections on safety margins need to be considered to reach a proper conclusion.

The design goal of the modular HTGR is that for all Licensing Basis Events, including Beyond Design Basis Events (BDBEs) with a frequency of occurrence as low at 5×10^{-7} per plant year, the offsite dose

consequences do not exceed the plume exposure Environmental Protection Agency Protective Action Guides (PAGs) at the facility Exclusion Area Boundary (EAB), which is located approximately 425 m from the reactor building. The design objective of meeting the PAGs at the EAB is the most limiting relative to the regulatory requirements for offsite doses, including the requirement to meet 10 CFR 50.34 dose limits for Design Basis Accidents. The ability to meet the PAGs at the EAB even for BDBEs provides substantial safety margins relative to the regulatory requirements for offsite doses. These margins are evident in Figure 1, which is taken from the Licensing Basis Event Selection White Paper [1]. As-manufactured fuel quality is one of the design selections that contribute to these margins, each of which entails considerations of cost and benefit.

With regard to the limits on fuel quality, the preliminary specifications on SiC defects and heavy metal contamination included in the Fuel Qualification White Paper already require large sample sizes for destructive testing to show compliance with the specifications. The statistical quality control (QC) approach used for HTGR fuel acceptance testing ensures larger margins for fuel performance than might be evident from an examination of the fuel specification. This is the case because the actual fuel quality must be considerably better than the specification limits. Otherwise, the sample sizes that would be necessary to show compliance with the specifications at the 95% confidence level would be prohibitively large, should the actual product quality level be close to the specification limits.

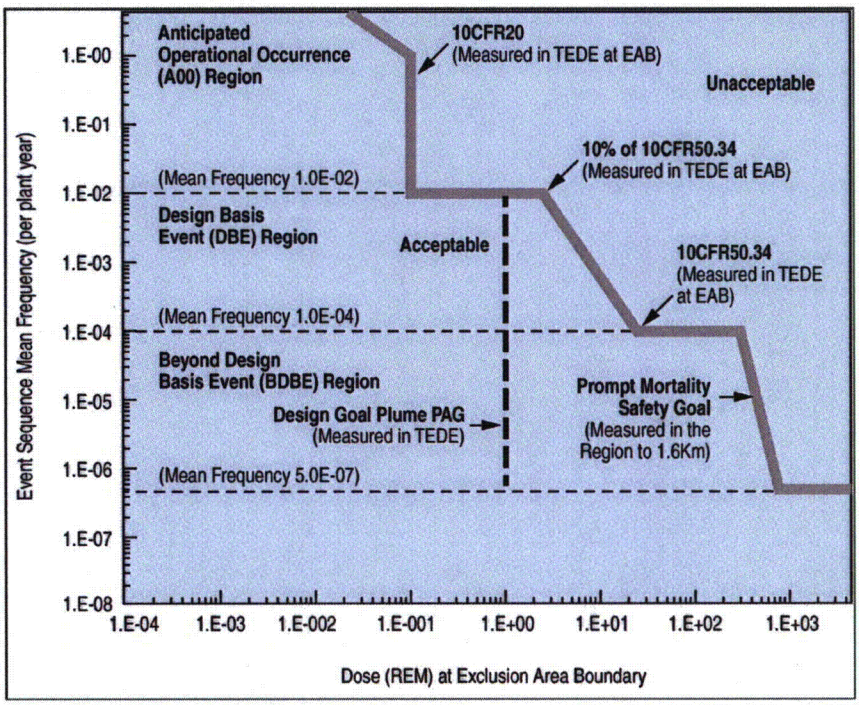


Figure 1: Frequency-Consequence Chart with Top Level Regulatory Criteria

Reference:

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

RAI FQ-B34/MST-B34: The entire fuel acceptance criterion is based on being able to provide fuel that is of equal quality to the German fuel. Rather than relying entirely on sampling of product to determine fuel quality, it may be prudent to establish on-line process monitoring standards. If on-line inspection is not practical, it may still be possible to set ranges for fluid flows, chemistry or other processing variables that can be monitored for all particles. Have on-line process monitoring or inspection techniques been considered?

Comment: It has been difficult for the US to produce fuel of equivalent quality to the German fuel in the past. Controlling the manufacturing process is crucial to obtaining high-quality fuel.

Response FQ-B34/MST-B34:
One of the major program elements of the AGR Fuel Development and Qualification Program [1] is to establish fuel fabrication capability in the U.S. that replicates in the NGNP particle design, to the greatest extent possible, the properties of the coatings on German fuel particles, which have exhibited superior irradiation and accident performance. The fuel fabrication program element also includes establishing the fuel fabrication infrastructure, process development for the kernels and compacts, coating process model development, QC methods development, fuel process scale-up analyses, and development of the process documentation needed for technology transfer to industry.

Based on more than 50 years of coated fuel particle fabrication technology development, it is known that certain product attributes need to be specified and confirmed by QC methods. It is also known that a few process parameters can affect particle attributes and performance and therefore need to be monitored on-line during fabrication as part of the fuel QC effort. Therefore, on-line process monitoring is being considered for a limited number of key process parameters.

The AGR Fuel 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, both for product attributes and for on-line process parameter monitoring, will be established once scale-up and process optimization work is complete and prior to the fabrication of the NGNP first core.

For fuel fabrication on a production scale, the fuel manufacturer will decide the type and level of process monitoring that will be employed. Process monitoring will be an important part of fuel manufacturing operations to ensure that the process is running properly. However, fuel acceptance will still be based on testing of the product for compliance with the fuel product specifications. The key coating process parameters will be specified and controlled as a supplement to the product specifications to ensure that the coatings have the properties necessary for acceptable irradiation performance.

Currently available 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 [2].

References:
1. J. Simonds, "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," PLN-3636, Rev. 0, INL, September 2010.
2. https://inlportal.inl.gov/portal/server.pt/community/ngnp_public_documents/, NRC Training Presentations.

RAI FQ-B35/MST-B35: Manufacturing quality assurance is essential. For each kernel and particle characteristic, can the numerical values, tolerances, measurements, methods and resolution of measurement techniques be included?

Comment: Eventually this information will be needed by the fuel manufacturer.

Response FQ-B35/MST-B35:
The detailed information requested in this RAI will be provided at the appropriate time in the fuel product specification. As noted in the response to RAI FQ-B34/MST-B34, final detailed fuel fabrication process control and product characterization procedures, both for product attributes and for on-line process parameter monitoring, will be established once scale-up and process optimization work is complete and prior to the fabrication of the NGNP first core.

Currently available 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-B36/MST-B36: The crystal structure of the SiC layer is given as key parameter controlling fuel performance in Section 3.3.3. Why isn't it listed as a parameter that will be monitored?

Comment: An adequate set of monitored fuel specifications is necessary to ensure the proper fuel quality.

Response FQ-B36/MST-B36:
The crystal structure and crystallite size of SiC will be dependent on the deposition conditions. The proper crystal size structure and density for SiC is achieved by controlling the partial pressure of the gases and the deposition temperature in the coater.

RAI FQ-B37/MST-B37: Are there any pre-irradiation thermal tests? Are there any tests to check the pre-irradiation bonding between the coating layers? What additional information could be obtained on small samples from large batches of fuel that could indicate fuel quality for parameters that are fundamental to performance?

Comment: Most of the fuel characteristics that will be monitored are for individual layers or properties. This question is asking if it is possible to batch test in a way that measures the aggregate particle properties necessary for fuel performance.

Response FQ-B37/MST-B37:
Based on more than 50 years of coated fuel particle fabrication technology development, it is known that certain product attributes need to be specified and confirmed by QC methods. It is also known that a few process parameters can affect particle attributes and performance and therefore need to be monitored on-line during fabrication as part of the fuel QC effort. However, no measureable aggregate particle property necessary for acceptable fuel performance, as suggested in the comment that accompanies this RAI, has ever been identified. The properties of the individual coating layers are specified to ensure that the coatings work together as a coating system.

There has been no indication over these years of coated fuel particle fabrication technology development of a need for pre-irradiation thermal testing or tests to check the pre-irradiation bonding between coating layers. The process that is used to apply the TRISO coating to the fuel kernels, and in particular the conditions for deposition of the IPyC and the SiC layers, has been shown to make lack of bonding unlikely. Bonding of the IPyC and the SiC is of primary interest to ensuring proper particle performance.

RAI FQ-B38/MST-B38: The FQ White Paper states "All irradiated fuel spheres will be subjected to heating tests simulating transient and accident temperatures, first at 1600°C for 100 hours and then at 1800°C for 100 hours." Could the material be annealed and the properties improved by holding the material at 1600°C? Will this improve the performance at 1800°C?

Comment: This question is also related to the path-dependence of material properties and the difficulty in determining which data is conservative for the design.

Response FQ-B38/MST-B38:
With regard to the pebble sphere testing, the response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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.

For prismatic compacts, the AGR program plans isothermal safety testing at a given temperature and will not test in a stair-step fashion at 1600 and then 1800°C, where annealing effects could potentially influence the results. However, some safety testing of fuel compacts using the time-temperature response of an HTGR to a depressurized cooldown is anticipated.

RAI FQ-B39/MST-B39: There are very few failed particles in the database and it is not explained how the specific failure mechanism can be determined for each particle. There are so many particles in a sphere or compact, it is unclear if the failed particles could be identified to determine the failure mechanism. How are fuel particle failures used to benchmark a computational code if the failure mechanism for each particle is not identified?

Comment: Code verification is an important step in determining the test conditions for the fuel qualification program.

Response FQ-B39/MST-B39:
For many of the known historical failure mechanisms for TRISO fuel discussed in the Fuel Qualification White Paper, the failure fractions were large (percent level) and could be identified during PIE. Changes were made in fabrication, the design service condition envelope, or the fuel design to minimize or eliminate the problems. Now, with high quality fuels such as are being produced in the NGNP/AGR Program, very few failed particles are encountered as a result of irradiation (none in the AGR-1 test). Margin testing is being conducted in the AGR-2 test (one irradiation capsule with temperatures up to 1,400°C), and more extensive margin testing is planned for AGR-7 with the expectation of failing some fuel. Safety testing will be conducted at increasing temperatures to observe failure thresholds.

Exposed kernel particle failures are detectable by fission gas releases measured during irradiation and safety testing. If those failures are few, finding the failed particles in a fuel compact or fuel sphere by optical microscopy of cross-sections is not practical. The most effective way to identify a relatively few

failed particles is to measure the fission product content of individual particles by gamma counting. This can be conveniently done on a large number of particles (such as a deconsolidated fuel compact or fuel sphere) with an automated individual microsphere gamma analysis (IMGA) apparatus. This tool, effectively used in the past, is currently in use in the AGR-1 PIE.

When a fuel particle deficient in fission product content is discovered, it can be sectioned and examined by optical microscopy, scanning electron microscopy (SEM), and transmission electron microscopy (TEM), for example, to determine the release mechanism. The NGNP/AGR program is also pursuing the possibility of performing nondestructive examination of individual irradiated particles using x-ray tomographic methods. Some failure mechanisms, such as corrosion of the SiC layer by fission products and thermal decomposition of the SiC layer at elevated temperatures, would be evident in many particles within a fuel compact or fuel sphere and would be observable by optical microscopy of cross sections, again likely to be followed by more detailed SEM and TEM examinations.

RAI FQ-B40/MST-B40: There are a large number of material properties and failure mechanisms that contribute to fuel performance. Will a statistical analysis method, such as "design of experiments" be used to determine the importance of various factors and combinations of factors on fuel performance? Is so, please describe this method.

Comment: Many variables, including material properties and operating conditions, interact to determine fuel performance. Identifying the most important combinations of factors that affect fuel performance is large undertaking that could benefit from a logical and quantifiable method to screen the factors.

Response FQ-B40/MST-B40:
A thorough review of 40 years of U.S. and international coating experience was performed in the wake of the prior performance of the U.S. TRISO fuel in the late 1980s to understand what went wrong. The conclusions of the review were used to develop the specifications for AGR-1.

Beyond that review, statistical design of experiment approach was used to understand the importance of some of the key fuel attributes, material properties, and service conditions on fuel performance [1]. Subsequently, a detailed parametric evaluation of fuel attributes was performed to support development of the AGR fuel specification [2]. As was confirmed during these evaluations, many of the specifications when evaluated over a wide range do not strongly affect the failure probability of TRISO particles. Results of a detailed analysis suggest that the three most important specifications for the thermomechanical performance of the fuel are buffer thickness, sphericity, and anisotropy of the IPyC layer, followed by SiC and IPyC thicknesses. OPyC properties, SiC density and IPyC density had little impact on predicted thermomechanical fuel performance. (See response to RAI FQ-B52/MST-B52 for more details.) Additional assessments as part of the Generation IV international forum have identified that the key material properties needed to accurately describe the thermomechanical behavior of TRISO fuel include: strength of PyC and SiC layers, shrinkage/swelling of PyC, and irradiation induced creep of PyC. Measurements are currently underway in the United States and Europe to measure these properties on modern TRISO fuel.

The excellent performance of AGR-1 shows that the relationship between fuel fabrication process parameters, fuel particle properties, and fuel performance are now well understood.

References:

1. G. K. Miller, D. A. Petti, D. J. Varacalle and J. T. Maki, "Statistical Approach and Benchmarking for Modeling of Multi-dimensional Behavior in TRISO-coated Fuel Particles," *J. Nucl. Materials,* Vol. 317, 2003, pp. 69-82.

2. D. A. Petti, "NGNP/VHTR Technology Development," presentation to DOE Nuclear Energy Advisory Committee, April 2011.

RAI FQ-B41/MST-B41: Has the process summarized in FQ Section 3.2.1.2 been demonstrated in the US at the FDL?

Comment: In Section 3.2.1.2, the German TRISO particle manufacturing process is summarized, noting perceived high quality of the particles produced by the Germans at that time.

Response FQ-B41/MST-B41:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B42/MST-B42: Will the QA program outlined for the PFP be applied to the FDL?

Comment: A summary of the quality assurance program for the PFP is given in section 5.2.1.2. Although this is a summary of the program, it nevertheless indicates the important parameters to be monitored and controlled. It is not clear from the discussion that this same program will be applied to the FDL.

Response FQ-B42/MST-B42:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B43/MST-B43: It is suggested that the transient data might be more limiting, and that the data base might be extended, either by experimental measurements or analytic means. Discuss the maturity of the analytic models that might be used to address this area.

Comment: The projection of TRISO particle behavior beyond the current experimentally measured base by means of analytic models requires a numerical algorithm that has been validated. The only code mentioned in the FQ White Paper is PARFUME, and it is not clear if this code has been validated for this extrapolation, or if another code will be used.

Response FQ-B43/MST-B43:
Reactor vendor codes are described in Appendices D and E of the Mechanistic Source Term White Paper. The goal is to extend the experimental database so that analytical extrapolation is not needed. The PARFUME code is currently under development. Algorithms used in PARFUME have been verified during its development phase. Validation of PARFUME is planned to be completed after the AGR-7 irradiation experiment that will provide the validation data. For assessing transient versus isothermal temperature profile safety tests, PARFUME results will be used for scoping and guidance purposes only in the support of safety heat up experiments. Results from both the transient and isothermal safety tests will identify differences and aid in the code validation process.

RAI FQ-B44/MST-B44: In mentioning the destructive PIE of irradiated fuel spheres and fuel compacts, no mention is made of the determination of transuranics as a function of particle position within the sphere. This data would be a very valuable benchmark for validating cross section and burnup codes, because of the double heterogeneity effect, which is unique to this fuel type. Is it possible to include this measurement in the PIE protocol? If it is included, please state it more clearly.

Comment: The PIE of fuel compacts is described in this section, and includes the determination of fission product distribution. However, it is not clear if the transuranic content of the TRISO kernels will be determined as a function of position within the fuel compact. The desire to have this data will be useful to validate reactor physics codes that have a double-heterogeneity model. These models are necessary to carry out fuel element calculations, and a variety of models are used in the various codes used for this step.

Response FQ-B44/MST-B44:
Currently, burnup measurements (including transuranic radiochemical analysis) are made using three batches of 20 particles each randomly selected from all the particles obtained by deconsolidation of a fuel compact. The particles are crushed, the kernels leached, and the solutions subjected to mass spectrometry and gamma spectrometry. It is estimated that at best, differential deconsolidation of a compact into four axial segments could be done. Most physics codes do not model this level of detail.

The NGNP/AGR program is also taking extensive gamma-ray spectrometric measurements of the fuel compacts with the intention of measuring burnup based on fission product absolute activities and fission product ratios, and benchmarking these measurements against the radiochemical burnup measurements (including transuranics) performed on selected fuel compacts. Data will be available for each fuel compact in 0.1 inch axial segments, and preliminary results indicate fairly good agreement with the calculated burnup numbers.

RAI FQ-B45/MST-B45: What range of carbon and oxygen will be included in the testing of UCO fuel? Has this range been manufactured before by other vendors?

Comment: The range of the carbon and oxygen fractions in the kernel of the TRISO particles implies a range of requirements on the manufacturing process. Thus, if some other manufacturer had fabricated fuel using the same range of carbon to oxygen, a good starting condition would be available for the fabrication process.

Response FQ-B45/MST-B45:
The specification for UCO fuel to be tested in the NGNP/AGR Fuel Program includes atom ratios in the kernel of carbon (C), uranium (U) and oxygen (O) as follows:

$$C/U = 0.4 \pm 0.1$$

$$O/U = 1.5 \pm 0.2$$

$$(C + O)/U = \leq 2.0$$

These specifications amount to an initial UC_2 fraction (UC is also usually present, but for simplicity only UC_2 is discussed here) between 15 and 25%, the balance being UO_2. For the burnup anticipated in an

NGNP design, the range of initial UC_2 fraction between 15 and 25% will ensure that the oxygen potential in the fuel kernel will be controlled by the chemical equilibrium of the reaction

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

More details regarding the thermochemistry of the UCO fuel are presented in the response to RAI FQ-50/MST-54.

TRISO-coated UCO particle fuels in this composition range were fabricated in the United States many years ago prior to the NGNP/AGR Program. The most recent fabrication campaign prior to the NGNP/AGR Program, roughly 20 years ago, supported the MHTGR and New Production Reactor (NPR) Programs. The fuel for NPR irradiations was fabricated at B&W, which is also involved in fabricating the fuel for the NGNP/AGR Program. Thus, the only currently operational commercial fuel fabrication facility with prior experience with UCO TRISO-coated particle fuel is B&W, which also participates in the NGNP/AGR Program.

RAI FQ-B46/MST-B46: What technical issues are alluded to in the AGR-1 test goals?

Comment: The description of the AGR series of tests is somewhat rudimentary, and this might be acceptable for the tests to be carried years from now. However, the first test described in AGR-1 requires more detail.

Response FQ-B46/MST-B46:
Technical issues related to the AGR-1 test objective of successfully designing, fabricating and operating a multi-capsule test train are described in Reference 1. A few of the major issues involved are:

- Neutron flux tailoring with the use of boron carbide dispersed in graphite sample holders and a hafnium shield;
- Slip fit of capsule through tubes to prevent bowing from differential thermal expansion;
- Thermocouple survivability;
- Gas flow to control capsule temperature and allow for continuous fission gas monitoring; and
- Brazing together of different metals.

There were several technical issues associated with the AGR-1 test objective to provide a fundamental understanding of the relationship between the fuel fabrication process and fuel product properties and irradiation performance. The most signification of these were related to pyrocarbon layer anisotropy, pyrocarbon density, SiC layer microstructure, and the inner pyrocarbon—SiC layer interface (bonding). These fuel fabrication, properties, and performance relationships are described in detail in Reference 2. Technical issues related to the final AGR-1 test objective of providing irradiated UCO fuel for safety testing will be documented in the AGR-1 PIE report. For this objective, the primary issue involved remote handling and disassembly of the test train and its components.

References:

1. S.B. Grover and D.A. Petti, "Initial Irradiation of the First Advanced Gas Reactor Fuel Development and Qualification Experiment in the Advanced Test Reactor", proceedings GLOBAL 2007, September 9-13, 2007, Boise, Idaho USA, paper #177457.

2. 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.A. and German TRISO-coated particle fuel, and their implications on fuel performance," Nuclear Engineering and Design, 222 (2003), pp. 281-297.

RAI FQ-B47/MST-B47: (a) Describe how the MTR fuel test conditions match the conditions expected in the NGNP with respect to the entire nuclear, physical, and chemical environment. (b) Explain how irradiation at only two testing temperatures will be limiting.

Comment: It is not clear that for the testing for "Full Burnup Irradiation Target" two temperature test conditions are sufficient. Thermal mechanical fuel cycling may actually show that shifting between temperatures can be more demanding than staying in a single fuel (temperature) zone.

Response FQ-B47/MST-B47:
The two temperature test conditions noted in the RAI were associated with PBMR plans for irradiation of pebble fuel. In the context of pebble bed fuel qualification, the response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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.

In the context of prismatic fuel qualification, the fuel compacts will experience a variety of temperatures and burnups in the planned AGR experiments. For example, in the AGR-1 experiment, the time-averaged peak temperature was between 1200 and 1250°C, depending on the capsule; the time-averaged volume average temperature was ~1100°C; and the time-averaged minimum temperature was between 850 and 900°C, depending on the capsule. The thermomechanical aspects of the fuel test conditions are discussed in the response to RAI FQ-B7/MST-B7. With regard to the chemical environment, future irradiations, beginning with AGR-3/4, will include capsules with inlet gas impurities at expected technical specification levels throughout the irradiation. The overall set of fuel test conditions are intended to bound expected NGNP service conditions.

RAI FQ-B48/MST-B48: Explain how the testing of fully burned fuel, as opposed to partially burned fuel, will give the only necessary data for the distribution of gaseous and volatile fission products throughout the fuel lifetime.

Comment: Migration and nuclear transformation of the gaseous fission products may affect tested results. Also, a note in the FQ White Paper says that partial burnup testing may not be performed in conjunction with the NGNP project.

Response FQ-B48/MST-B48:
In the context of prismatic fuel qualification, irradiated fuel compacts experience a range of burnups, given the axial neutron flux profile in the Advanced Test Reactor. Compact burnups in AGR-1 range from 11 to 19% FIMA. In AGR-3/4, where releases from failed fuel will be studied parametrically and the impact of burnup on that release is more important, projected burnups range from 6 to 16% FIMA, and temperatures for different capsules will be established at discrete temperatures between 900 and 1300°C. This large parametric variation in temperature and burnup should be adequate to understand gaseous and volatile fission product behavior throughout the fuel lifetime. Irradiation acceleration factor is only 1.6 for AGR-1 and AGR-2 and 2.5 for AGR-3/4.

In the context of pebble bed fuel qualification, the response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B49/MST-B49: Prior HTGR/HTR fuel qualification programs (US, UK, Germany, Japan, China) have made efficient use of fuel testing in real-time HTGR/HTR neutron environments available in experimental or prototype HTGR/HTR facilities, coupled with fuel testing in accelerated neutron environments, in material test reactors (MTRs). Explain the rationale for not including as major elements in the NGNP fuel qualification program:

- real-time testing of candidate NGNP fuel systems in existing HTGR/HTR irradiation facilities;
- safety testing of irradiated candidate fuel elements from these real-time environments, and
- qualification/confirmatory testing and monitoring in the NGNP prototype plant.

Comment: The combination of irradiation testing under real-time HTGR/HTR neutron environments coupled with accelerated testing in MTRs has a proved to be an effective process to evaluate candidate fuel systems under representative irradiation conditions in a reasonable period of time. Real-time qualification testing in an HTGR/HTR environment seems to be downplayed for both the pebble-bed and prismatic fuel concepts in the FQ White Paper. The only significant irradiation testing in this planned is under accelerated MTR environments.

The German fuel development program of the 1980s and 1990s had access to the AVR along with a number of European MTRs. This access to real-time and accelerated irradiation environments proved to be an invaluable asset to their development efforts. The success of this effort is evidenced in that all of the experimental fuel performance data (Chapter 3 and Appendix A) presented as evidence for the NGNP pebble bed fuel concept was generated within that program.

Currently there are two experimental HTGR/HTRs operating - one is the 30 MWth HTTR in Japan and the other is the 10 MWth HTR-10 in China. These experimental facilities provide the only available irradiation environments representative of the prismatic and pebble bed HTR concepts. The NGNP Program should initiate cooperative programs between the U.S., Japan, and China to provide irradiation testing access to these experimental facilities.

Response FQ-B49/MST-B49:
The decision on what facilities to use to test TRISO fuel was made early in the AGR program back in 2003. At that time, key drivers were irradiation volume and ability to test fuel at modest levels of acceleration like the successful German program based on recommendations made in Reference 1. Based on the aggressive NGNP deployment date of 2021, as required by the Energy Policy Act of 2005, irradiation testing in ATR and safety testing at hot cell facilities at both INL and Oak Ridge National Laboratory (ORNL) were the only viable options to meet the schedule. While testing in an HTGR offers potential benefits, it also offers some drawbacks. For example, use of HTR-10 and HTTR for fuel qualification was considered to be problematic because of the low power density and low capacity factor of the reactors, the lack of in-reactor testing infrastructure, and, most importantly, the lack of PIE and safety testing infrastructure. Beyond these considerations, the international shipping of material containing enriched uranium both in the unirradiated and irradiated state is becoming extremely difficult,

time consuming, and expensive, with extensive government-to-government agreements being required. All of these factors led to U.S. centric program. As mentioned in the response to RAI-FQ-3/MST-3, monitoring of the activity in the reactor coolant to confirm fuel performance is expected in the NGNP prototype reactor.

Reference:

1. D. A. Petti et al., "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," *Nuclear Engineering and Design*, 222 (2003) pp. 281-297.

RAI FQ-B50/MST-B50: Now that the South African (SA) Pebble Bed Modular Reactor (PBMR) Project has been terminated, how does the NGNP pebble bed concept go forward? A major portion, if not all, of the only detailed plan available in the FQ White Paper now appears to be no longer applicable. Has an alternative business interest come forward to pick up where SA left off or will the NGNP pebble bed concept be eliminated?

Response FQ-B50/MST-B50:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B51/MST-B51: Discuss the methods available to monitor the coated particle characteristics and fuel-sphere/fuel compact characteristics and show explicitly just how the monitoring will take place. Will process controls be in place to insure that the coated particle characteristics fall within the proper range? Please elaborate as to which process controls are important in controlling coated particle characteristics such as SiC microstructure, iLTI and oLTI anisotropy, buffer layer density, Weibull properties (SiC characteristic strength and modulus), iLTI permeability, and coating layer thicknesses. Which methods in the fabrication process are employed to control mechanical, thermal, elastic and Weibull (SiC characteristic strength and modulus) properties? Again, give specific examples.

Response FQ-B51/MST-B51:
See response RAI FQ-B34/MSTB34.

RAI FQ-B52/MST-B52: Which characteristics identified for coated particles contribute directly to TRISO fuel particle irradiation performance and are they explicitly accounted for in the current fuel performance model(s)? What is the range of acceptability for the characteristics listed and how are these affected by the fuel normal operating conditions (temperature, burnup and fast fluence)? Provide details of how the limits of these characteristics were determined to ensure successful TRISO fuel performance? In what manner will fuel at the respective limits be selected and tested to insure they meet fuel particle performance requirements?

Comment: This RAI item pertains to evaluating the qualification and performance of NGNP candidate fuel concepts (see discussion in draft NUREG 1338, February 26, 1996, page 7). Of particular interest are the items listed below:

- Design thicknesses of fuel particle coatings and the bases for these thicknesses given fuel particle failures from manufacturing, normal operation, and accidents.
- Quality control of the manufacturing process for particle fuel and resulting tolerances on particle coatings.
- Fuel performance of specific coated particles and coating tolerances demonstrated from irradiation and safety tests.

Response FQ-B52/MST-B52:
The approach in the AGR Fuel Development and Qualification Program was to apply historic German coatings on a UCO kernel. The AGR fuel specification was initially derived from historic U.S. and German fuel specifications. This specification was modified based on an assessment of fuel performance using the PARFUME code. Initially, a statistical design of experiment approach was used to understand the importance of some of the key fuel attributes, material properties, and service conditions on fuel performance [1]. Subsequently a detailed parametric evaluation of fuel attributes was performed to support development of the AGR fuel specification. [2]

For a specific fuel attribute in Figure 1, the specific value on the x-axis (e.g., buffer thickness) is held constant (e.g., at 85 μm) and all other attributes are allowed to vary about their mean values in accordance with their population standard deviation in the Monte Carlo analysis to establish a fuel failure probability (see response to RAI FQ-B24/MST-B24 to understand how failure rates are calculated). Another calculation is then done at a different buffer thickness and a different failure probability is calculated. This is done for a discrete number of specific values and then repeated for each fuel attribute to obtain the fuel failure fraction curves shown in Figure 1.

The results show that many of the specifications when evaluated over a wide range do not strongly affect the failure probability of TRISO particles. The three most important specifications for the thermomechanical performance of TRISO fuel are buffer thickness, asphericity (aspect ratio), and anisotropy of the IPyC layer followed by SiC and IPyC thicknesses. OPyC properties, SiC density, and IPyC density had little impact on predicted thermomechanical fuel performance. Based on this analysis, critical limits were established for the specification. Currently, the AGR-1 and AGR-2 fuel meet the specifications for these critical attributes with large margin, further reducing the potential for fuel failure.

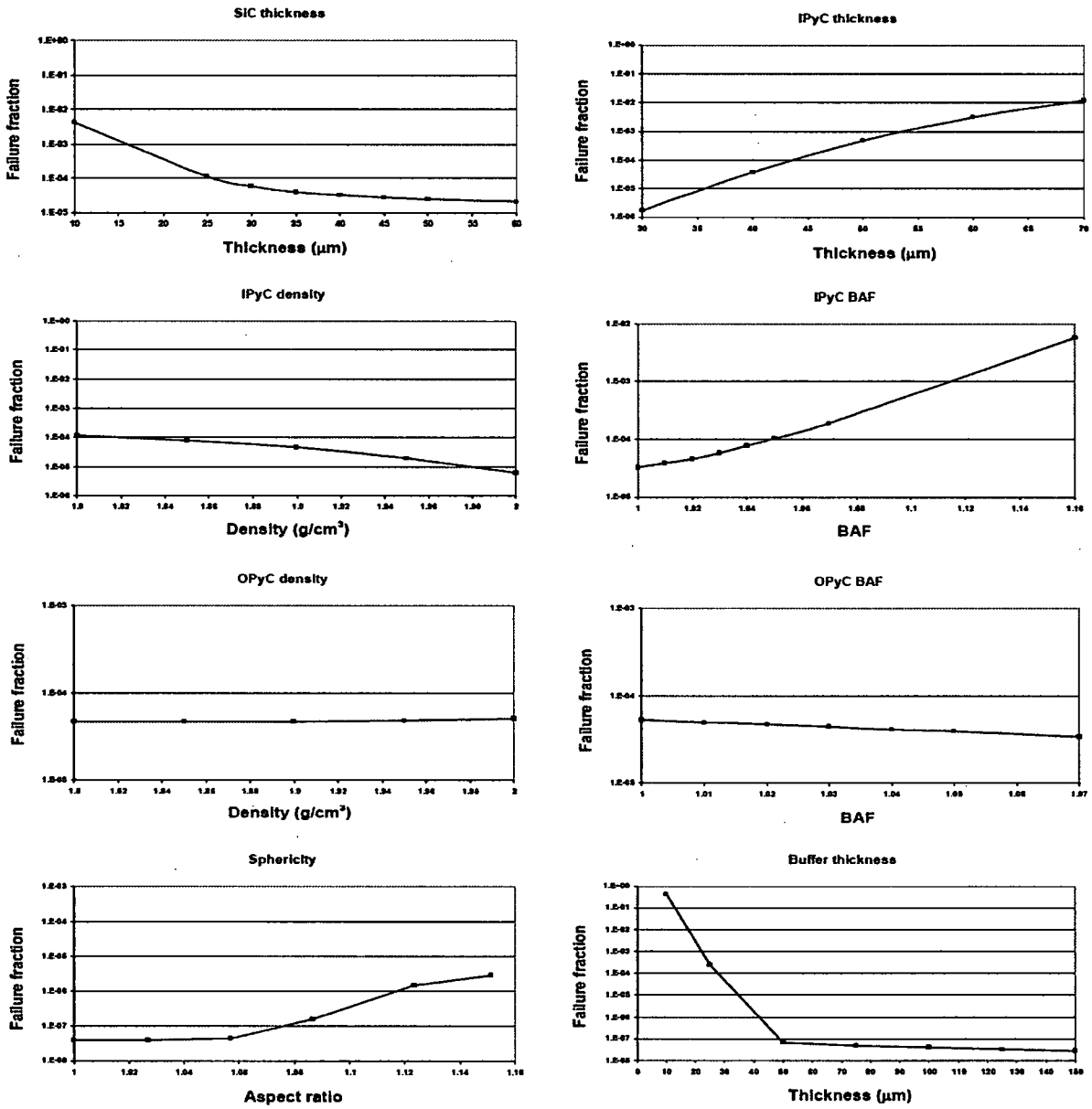


Figure 1. Effect of particle attributes on TRISO-coated particle failure fraction.

Additional assessments as part of the Generation IV international forum have identified that the key material properties needed to accurately describe the thermomechanical behavior of TRISO fuel include: strength of PyC and SiC layers, shrinkage/swelling of PyC, and irradiation induced creep of PyC. Measurements are currently underway in the United States and Europe to measure these properties on modern TRISO fuel.

Quality control on the fuel fabrication process is established using statistical sampling. All specifications are based on 95% confidence assessments of measured value against the specification value. Sample

sizes are based on estimates of historical standard deviations, binomial probabilities, and 5% false rejection rate. Some properties are treated as continuous variables (e.g., thicknesses and densities). Others are treated as discrete (e.g., SiC defects). The precision of measured mean values and the standard deviations of attribute populations measured today are much better than historical values because of improved measurement techniques and better control of coating process. These well-characterized data sets remove potential sources of uncertainties in fuel performance predictions.

Fuel will not be explicitly tested at the limits of each critical fuel attribute for practical reasons. Instead, the irradiation and safety testing will bound that expected in the NGNP using TRISO fuel fabricated to meet the specification at 95% confidence. The irradiations and accident testing will contain enough fuel to statistically meet (if not exceed by large margin) the incremental failure rate specifications under normal and accident conditions that are used in the NGNP design. This proof testing approach is considered acceptable because of (a) the highly characterized nature of TRISO fuel being produced in the AGR program; (b) the use of fuel design choices, fuel specifications, and design service limits to mitigate known fuel failure mechanisms to acceptable levels; and (c) the lack of fuel failures expected under normal and accident conditions (as demonstrated by AGR-1 and as indicated in the fuel performance analyses conducted to date).

References:

1. G. K. Miller, D. A. Petti, D. J. Varacalle and J. T. Maki, "Statistical Approach and Benchmarking for Modeling of Multi-dimensional Behavior in TRISO-coated Fuel Particles," *J. Nucl. Materials,* Vol. 317, 2003, pp. 69-82.
2. D. A. Petti, "NGNP/VHTR Technology Development," presentation to DOE Nuclear Energy Advisory Committee, April 2011.

RAI FQ-B53/MST-B53: Provide additional information on the Burn Leach procedures as to whether differences exist between the procedures used for fuel compacts vs. those used for spherical fuel elements. Identify any follow-on chemical analyses performed to determine the absolute quantity of free uranium, or other heavy metal contaminants (fissile or fertile). What assumptions are made about the source of this contamination and discuss the justification/evidence for these assumptions? Address whether specific quantitative analyses are, or will be, performed to determine the isotopic content of the measured contamination?

Comment: Exposed uranium and other fissile/fertile materials are key factors in controlling the source term for either spherical fuel elements or cylindrical compacts that contain TRISO coated particle fuel. Whether the measured contamination levels consist of natural uranium, enriched uranium, thorium, or other fissile materials appear to be important factors in source term determination.
Initial low-levels of fertile material contamination (Th-232 and natural uranium) in the fuel matrix material and the subsequent breeding of fissile material isotopes have been shown to be a significant source of in-reactor fission gas release [Ref.].

Reference: H. van der Merwe and J. Venter, "A Method to Evaluate Fission Gas Release During Irradiation Testing of Spherical Fuel", Proceedings HTR2008: 4th International Topical Meeting on High Temperature Reactor Technology, 23 Sep.-1 Oct., 2008, Washington DC, USA. Paper 2008-58184]

Response FQ-B53/MST-B53:
For spherical fuel elements, the response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in INL letter CCN 223977

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For cylindrical fuel compacts, see the responses to RAIs FQ-B3/MST-B3 and FQ-B78/MST-B78.

RAI FQ-B54/MST-B54: Statements made relative to accident testing appear to be contradicting. Please discuss this apparent contradiction and provide any additional data to illustrate your position.

Comments: The statement made referring to the transient temperature profile being more limiting does seem to be correct. In the transient tests illustrated, the temperature profiles achieve peak temperatures of 1620°C and 1700 °C in ~30 hr based on the DLOFC accident scenario, followed by a cool-down to ~1200°C over the next 270 hrs. This scenario compares to the isothermal heating tests where the temperature profile is much less severe, rising to 1250°C over a period of ~30 hr then rising to 1600°C in an additional ~8 hrs, or 1700°C and ~10 hr. These soak temperatures can last up to several hundred hours. Based on the data provided, 25 of the 26 recorded failed particles observed in the 1600°C and 1700°C heating tests failed during transient testing. Only one of the 26 failures was recorded in an isothermal test at 1700°C. Thus, the follow-on statement that "analytical models and a general understanding of the relevant phenomena indicate that the isothermal test should be more challenging to the fuel" does not appear accurate.

Response FQ-B54/MST-B54:
The basis for model projections and general expectations that post irradiation heating isothermal tests would be more limiting for fuel performance than a transient simulation test with peak temperature at or near the constant temperature of an isothermal test is discussed in the response to RAI FQ-B18/MST-B18.

As discussed in the response to RAI FQ-33, the thermal time constant of the German TRISO particle design is approximately 40 milliseconds. The differences in the heating rates for the isothermal and transient simulation heating tests are on the order of a few degrees per hour. Such rates are far too slow to impose thermal stresses on the coated particles.

The transient tests were all conducted with spheres irradiated in the AVR that were discharged after spheres from the same loading that were subjected to isothermal tests, with 19 of the observed failures coming from a single sphere. Thus the spheres subjected to transient testing were generally exposed to higher burnup and longer times at the potentially elevated temperatures (possibly exceeding 1400°C at times depending on the trajectory through the core) associated with the nominal 950°C mixed mean coolant outlet temperature operation of the AVR. The generally more severe irradiation conditions of the spheres subjected to transient simulation tests, and the unknown irradiation conditions for individual spheres represent important uncontrolled factors in the interpretation of the heating test results.

RAI FQ-B55/MST-B55: Please elaborate further on the suggestion made that "Additional confirmatory data under actual service conditions may be provided by a post irradiation test and inspection program to be conducted on fuel discharged from the pebble-bed design." Discuss any planning for real-time testing of either FDL or PFP fuel.

Comment: This suggestion would appear to be a "must," especially in view of the fact that no real-time testing of spherical elements are planned, or at least identified, in this fuel qualification plan. This point also applies to the prismatic fuel. The inclusion of a "Post Irradiation Fuel Inspection Program" for either

NGNP reactor concept should also be a "must" requirement. Such programs were invaluable for all previous experimental and prototypical HTGR plants. See also RAI FQ-B55/MST-B55.

Response FQ-B55/MST-B55:
For pebble bed fuel, the response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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 was noted in the comments that accompany this RAI that the point of the RAI also applies to prismatic fuel. Information regarding potential post irradiation examination of prismatic HTGR fuel discharged from the NGNP is provided in the response to RAI FQ-26/MST-31.

RAI FQ-B56/MST-B56: Provide the pre-irradiation as-fabricated quality data on recent UO_2 TRISO fuel particles fabricated into: a) spherical fuel elements in South Africa's FDL (described on page 80); and b) fuel compacts at ORNL for inclusion into the AGR-2 irradiation test (described on page 89) now underway at the ATR in Idaho. Discuss the as-fabricated quality data relative to the NGNP pebble bed as-fabricated requirements (Table 16) and prior German fabrication experience (Tables 2 and 3). Inclusion of this data will provide important information regarding the status pebble bed fuel manufacturing capability today.

Comment: The FQ White Paper provides no reference to as-manufactured quality data of UO_2 TRISO fuel particles fabricated in recent years in South Africa. These fuel particles have in at least two campaigns been fabricated into: a) spherical fuel elements in South Africa's FDL (described on page 80); and b) fuel compacts at ORNL for inclusion into the AGR-2 irradiation test, now underway. Both of these campaigns appear to be part of the approach outlined in FQ Sections 5.2.2.1 and 5.2.2.2 on pages 78 - 82.

At the HTR-2010 conference in Prague (October 2010), results were presented on "Recent Advances in HTR Fuel Manufacture" [Ref.]. The FQ White Paper presents as-manufactured fuel quality results on HTR pebble bed fuel element fabrication based on the German experience from the 1980's, the HTR-10 fuel element production in China, and recent results from South Africa's FDL production of spherical elements (see figure below). The South African fuel fabrication work appears to be part of the effort (Testing of Laboratory-Produced Fuel Spheres) described on page 80 of the FQ White Paper.

This important information should be included in the FQ White Paper together with detailed discussions as to how well the pre-characterization data meet NGNP as-fabricated quality requirements (page 62, Section 4.2.3.1 and Table 16, page 72) and fuel element characteristics (listed on pages 76 and 77). Comparisons to previous German production campaigns should also be provided. Failure to include these data impedes understanding of the current production capability for UO_2 TRISO fuel and spherical fuel elements for the NGNP pebble bed concept.

Reference: H. Nabielek, C. Tang, and A. Müller, "Recent Advances in HTR Fuel Manufacture", paper 094, HTR 2010 Conference, Prague, CZ (October 2010).

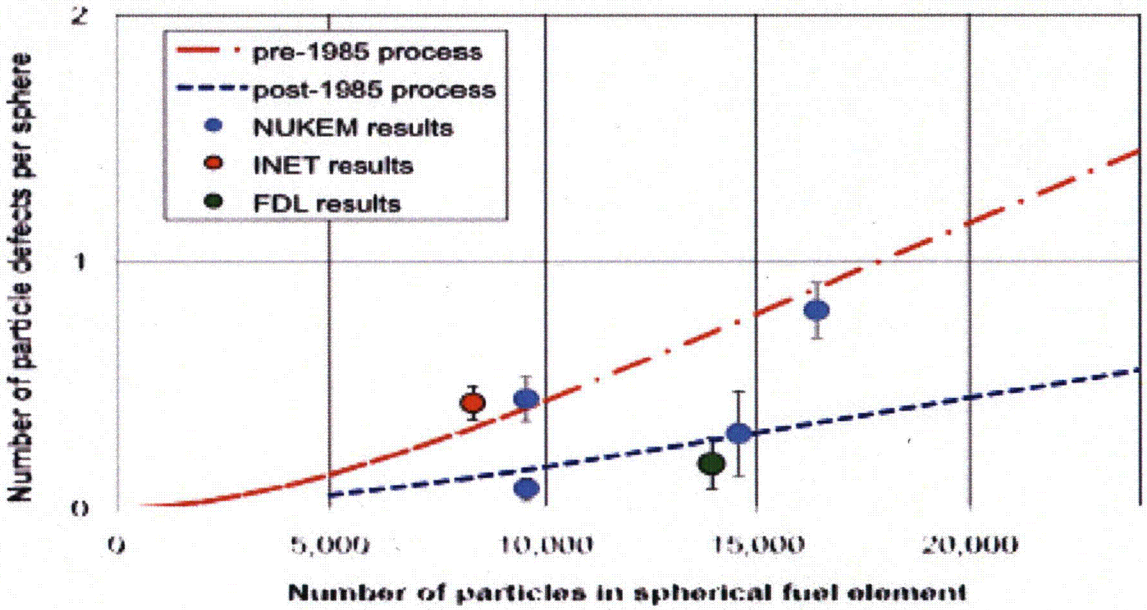


Figure Caption. Number of particle defects in manufacture as a function of the number of particles in a spherical fuel element. The red circle represents Chinese 2001 results and the green circle represents South African 2009 results. Black circles represent German results obtained by NUKEM during 1981-1988 and show the post-1985 improvement of manufacturing quality ("late NUKEM").

Response FQ-B56/MST-B56:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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 noted that quality data on the properties of UO_2 fuel fabricated in South Africa, beyond that contained in the reference cited in the comments that accompany this RAI, is proprietary to PBMR (Pty) Ltd.

RAI FQ-B57/MST-B57: Provide detailed information on the differences in the neutron spectra experienced in MTR irradiation facilities (such as the HFR in Petten, the IVV-2M in Russia and the ATR in Idaho) as compared to that anticipated in the NGNP pebble bed or the NGNP prismatic concepts.

Comment: This is of particular importance relative to accelerated neutron environment concerns and the fact that no real-time irradiation testing is discussed or planned in the FQ White Paper. Generally, MTR irradiation facilities have the capability to tailor the neutron spectra to meet their customer's irradiation needs. This possibility should be explored with the respective MTR facilities and adjustments made to insure a more representative HTGR neutron energy spectrum. This factor is important in insuring that the

proper fission and activation product distributions are achieved in the accelerated irradiation tests compared to the real-time HTGR neutron environment. Significant differences can lead to unrealistic fission/activation product inventories within the TRISO coated fuel particles.

Response FQ-B57/MST-B57:
Detailed information on the differences in neutron spectra between the ATR and those expected for pebble bed and prismatic designs of the NGNP are presented in TEV-1022 [1]. Reference 1 also discusses differences in inventories of the fission products silver and palladium. The response to RAI FQ-4/MST-5 concludes that the somewhat larger inventories of silver and palladium expected in the prismatic design of the NGNP, compared with irradiations in the AGR, pose no significant concerns for the performance of the SiC layer, the principal fission product barrier in TRISO-coated particle fuel. Additional information regarding the inventories of rare earth elements is provided in the response to RAI FQ-1/MST-1.

Regarding the concern expressed in the comment provided with this RAI that no real-time irradiation testing is planned in the NGNP/AGR Program, it should be noted that the AGR-1 irradiation was conducted at an acceleration of only about 1.6 relative to the irradiation time expected for the prismatic design of the NGNP. The AGR-2 irradiation is being conducted at a similarly modest acceleration factor. Subsequent irradiations are planned at acceleration factors of about 2.5. It is generally thought that accelerated irradiations are conservative in that they produce higher fuel particle powers and temperature gradients through the particles. Real-time irradiations slow down the rate of information produced from irradiation, PIE, and safety testing, and would significantly extend the overall schedule. As discussed in the response to RAI FQ-3/MST-3, a surveillance program will be conducted during startup of the NGNP during which data on fission gas release during real-time irradiation will be available, as well as data on plateout. Furthermore, as discussed in the response to RAI FQ-26/MST-31, it is also anticipated that PIE will be performed on fuel removed from the reactor as part of the surveillance program.

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

RAI FQ-B58/MST-B58: Provide the rationale for irradiation testing only nine FDL fuel spheres to establish an early indication of the effectiveness of the NGNP pebble bed fuel manufacturing process.

Comment: Based on the published South African FDL fuel element quality data [Ref.], the as-fabricated burn leach data fall between the AVR 21-2 and HFR-K5/-K6 as- fabricated quality data. Translating this into failure fraction numbers, as was done on page 29, the expected defect fraction with 50% confidence failure fraction is ≤1.3 x10-5, and with a 95% confidence, the failure fraction is ≤2.2 x10-5.
Nine elements, each with ~14,400 particles per element, represent a total population of ~129,600 particles. A successful irradiation with no observed failed fuel in-reactor would yield a failure fraction with 50% confidence of ≤5.35 x10-6, and with 95% confidence ≤2.31 x10-5. These results indicate that nine irradiated fuel elements is an insufficient population to demonstrate NGNP pebble bed as-fabricated fuel quality requirement at the upper 95% confidence level. However, increasing the number of elements to ten would provide a sufficient population to meet the requirements, provided no in-reactor failures are observed.

Reference: - H. Nabielek, C. Tang, and A. Müller, "Recent Advances in HTR Fuel Manufacture", paper 094, HTR 2010 Conference, Prague, CZ (October 2010).

Response FQ-B58/MST-B58:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B59/MST-B59: Provide details of the planning and the methodology that will be employed for the Heating Tests identified in Section 5.2.2.2.

Comment: No mention is made of how many irradiated fuel elements will be heated. The only indication that more than one test is to be performed is the term "Fuel spheres shall undergo ..."
Details of the methodology are also insufficient based upon the large testing database available from the German accident testing program from the 1980's and 1990's. All isothermal heating tests proposed should consider following the same heating sequence (or an equivalent sequence) as developed and used in the German FDP (Ref. – Schenk, Pitzer, and Nabielek, Jül-2234, September 1988, [Fig. 8, page 13]). Such multi-step heating sequences allows for:

- Fission gas release measurements at room temperature;
- Cleaning/moisture removal at ~300°C;
- Equilibration of internal gas pressure in TRISO fuel particles at temperatures of 1050°C and 1250°C; and
- Heatup to selected accident condition temperature and isothermal soak at this temperature for the required test period.

The heating test facility should be continuously purged with He during the entire accident test simulation, and this purge gas shall be monitored continuously for 85Kr release from the test fuel. This means from the start of the test at room temperature, through the heat-up sequence, the actual isothermal soak at the correct temperature/duration, and finally the cool-down to room temperature.

Response FQ-B59/MST-B59:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B60/MST-B60: Describe a detailed sampling plan and an accompanying statistical analysis that validates the selection of 15 spherical fuel elements are sufficient to establish the as-manufactured fuel quality of test elements from the NGNP pebble bed fuel production facility.

Comment: Typically fuel elements would be selected from each of the production lots for evaluation. NUKEM, GmbH, employed a sampling plan that generally called for the selection of five random elements from each production lot. Thus, whether 15 elements are sufficient does not make much sense without knowledge of the number and size of the typical production lots for this fuel manufacturing facility. In some of the AVR production campaigns, like AVR 19 where 24,600 elements were produced, as many as 70 elements (five each from 14 production lots) were destructively analyzed. For the smaller

campaigns like for the HTR Modul Proof test where ~100 elements were manufactured, only 10 were destructively analyzed.

The actual number of elements to be destructively examined in order to validate as-fabricated fuel quality will largely be a function of the Ufree/Utotal value and the heavy metal per fuel element (assuming reference UO_2 TRISO particle design). If the equilibrium NGNP pebble bed fuel element has ~6.2 g of uranium and ~11,200 particles based on the FQ White Paper (adjusted fuel design, page 84), then estimates for the number of elements to be examined can be made based on anticipated Ufree/Utotal values. For Ufree/Utotal values of 6 x10-5, 3 x 10-5, and 1 x10-5, the corresponding number of failed particles per element, on average, are ~2/3, ~1/3 and ~1/9, respectively. Using this information, sampling tables can be generated, as shown below, that provide an estimate of the failure fraction and confidence level that can be established. If for example, the failure fraction probabilities identified on page 29 [expected failure fraction ≤1.3 x10-5 (50% confidence) and ≤2.2 x10-5 (95% confidence)] are associated with an Ufree/Utotal value of ~1 x10-5, are to be established, then ~45 equilibrium fuel elements from the production facility would have to be destructively examined (via Burn Leach) to validate the as-fabricated fuel quality level.

No. of Fuel Elements Examined*	Ufree/Utotal Limit	Expected number of Failed Particles (on average)	Failure Fraction (95% Confidence)	Failure Fraction (50% Confidence)
15	60 x10-6	10	≤1.0 x10-4	≤6.3 x10-5
15	30 x10-6	5	≤6.3 x10-5	≤3.4 x10-5
15	10 x10-6	~2	≤3.7 x10-5	≤1.6 x10-5
25	60 x10-6	~17	≤9.1 x10-5	≤6.3 x10-5
25	30 x10-6	~8	≤6.3 x10-5	≤3.1 x10-5
25	10 x10-6	~3	≤2.8 x10-5	≤1.3 x10-5
45	60 x10-6	30	≤8.1 x10-5	≤6.1 x10-5
45	30 x10-6	15	≤4.6 x10-5	≤3.1 x10-5
45	10 x10-6	5	≤2.1 x10-5	≤1.1 x10-5

* Equilibrium Production-Line Fuel Sphere with 7 g UO2 (6.17 g HM) and containing ~11,200 particles (adjusted fuel element design, page 84).

Response FQ-B60/MST-B60:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B61/MST-B61: Compared to previous German campaigns, the irradiation of only 12 NGNP pebble bed Production Line fuel elements appear to be a small sample size to qualify a full-scale fuel element production facility. Discuss the rational for choosing this sample size and any backup plans in the event that the proof test performance goals are not achieved in these accelerated MTR irradiation tests. Provide the rationale and justification for not requiring, at a minimum, the irradiation testing of a similar number of Production Line fuel elements under real-time HTGR/HTR irradiation conditions. Does the lack of such a plan to include irradiation testing in real-time HTR environment cause any concern?

Comment: Because of the lack of an experimental or prototype HTGR/THE irradiation facility in the US, establishing a cooperative program between the US NGNP Program and Chinese Government, where NGNP pebble bed Production Line fuel elements could be routinely tested in the HTR-10 experimental reactor, would appear to be of prime interest. This same point can be made for a similar cooperative program between the US NGNP Program and the Japanese Government to irradiated NGNP prismatic fuel compacts under real-time conditions in the HTTR experimental reactor.

The irradiation of only 12 spheres, each containing 11,200 UO_2 TRISO coated particles, may not be a statistically significant sample. To irradiate the same particle population as for elements containing 14,400 particles, will require a minimum of 16 fuel elements be irradiated.

Response FQ-B61/MST-B61:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B62/MST-B62: The white paper states (page 98) that "Pretest predictions and after-test calculations will be performed for each irradiation test and some of the safety tests." Please provide these pretest predictions for the completed AGR-1 irradiation test and the AGR-2 irradiation currently underway. Discuss the results of pretest predictions in relation to the observed in-reactor fuel performance data based on monitored short-lived fission gas species. Indicate whether the formal ATR pretest predictions are derived from the same suite of fuel performance models and codes designed to predict NGNP prismatic fuel performance under normal operating conditions.

Comment: With the exception of the IAEA sponsored CRP-6, no references to any documents containing ATR pretest predictions were made in the FQ White Paper. Discuss the rationale for not including these performance predictions here. In most cases, these computer codes and the performance models have existed for many years and in some cases over decades.

Response FQ-B62/MST-B62:
Pretest and posttest performance predictions are clearly important for deriving the maximum benefit from the NGNP/AGR irradiation tests. In general, pretest performance predictions have two primary purposes: optimize the test service conditions, and provide an unbiased base case for code validation. In the case of the early AGR tests, the first purpose is more important, and for the final tests, the second purpose is more important. Posttest performance predictions follow the same pattern. For the early AGR tests, the emphasis is on supporting the derivation of improved component models and material property correlations. For the final validation tests, the posttest and pretest predictions would ideally be nearly

identical. Practically, complex irradiation tests do not typically run precisely as planned, and the primary reason for the posttest predictions is to reflect actual as-run test conditions.

The NGNP/AGR Fuel Development and Qualification Program is described in Section 5.3.1 and summarized in Table 21 of the Fuel Qualification White Paper. AGR-1 was primarily a shakedown test to qualify a new, untested irradiation test train design (six independently purged capsules, etc.) and a new post irradiation heating facility. However, the test train performed so well that AGR-1 can be used for gathering performance data for high-quality UCO fuel. AGR-2, which employs the same now-qualified test train used for AGR-1, is a data gathering test for both UCO and UO_2 fuel particles. AGR-3/4 and AGR-5/6 are also data gathering tests. AGR-7 and AGR-8 are validation tests for fuel performance and fission product release, respectively.

Extensive pretest nuclear and thermal analyses were performed by INL to support the design of the AGR-1 and AGR-2 irradiations. However, the inference of this RAI is that the primary interest is fuel performance and fission product release predictions. A limited amount of fuel performance predictions were made by INL with the PARFUME code. No pretest predictions were made by the reactor suppliers for either AGR-1 or AGR-2 using their design codes. Extensive pretest performance predictions have been made by both GA and INL for the AGR-3/4 irradiation, but the primary emphasis has been on optimizing the service conditions and developing a comprehensive test matrix for the new 12-capsule test rig. Comparable pretest analysis is anticipated for the AGR-5/6 tests, and pretest predictions using the reference reactor design methods will be made for the AGR-7/8 validation tests for both the irradiation and post irradiation heating tests.

Posttest performance predictions will be made for all of the AGR irradiations reflecting the actual as-run irradiation histories. GA has made posttest fuel performance and fission product release predictions for the AGR-1 irradiation, but the analysis report has not been issued at this writing because INL is still evaluating the thermal performance of the six capsules. (A number of the capsule thermocouples failed during irradiation, complicating the estimation of the actual operating temperatures). Furthermore, upon request from INL, GA would intend to make posttest predictions for all of the remaining AGR irradiation tests (with the possible exception of the three UO_2 capsules in AGR-2).

As described in Appendix D of the Mechanistic Source Terms White Paper, GA uses the CAPPER and TRAMP codes for irradiation capsule analysis. CAPPER and TRAMP are the local-point analogs of the full-core analysis codes SURVEY and TRAFIC-FD, respectively. Both sets of codes contain the same component models and have been benchmarked against each other. The post irradiation heating tests ("safety tests") would be analyzed with SORS/NP1, which contains the Goodin-Nabielek accident condition fuel performance model.

Pre-test fuel performance predictions have been completed by INL for AGR-1 and AGR-2 [1, 2] using PARFUME. No particle failures were calculated to occur during the AGR-1 irradiation. Given the lack of fuel failures in AGR-1 the gas release is associated with contamination. Since the contamination levels are an input into PARFUME, the code agrees well with the data. This detailed information was not included in the white paper because it was judged not to be needed to meet the objectives of the white paper review.

PARFUME does not predict particle failure in the UCO capsules in AGR-2 but it does predict particle failures in the UO_2 capsule because of the combination of gas pressure and asphericity. However, the CO gas production model in PARFUME is known to be overly conservative and may contribute to the calculated failures. The gas release from exposed kernels is in reasonable agreement with the initial R/B data (see RAI FQ-B63/MST-B63 for more information).

References:

1. D. Knudson," Advanced Gas Reactor-1Pre-Test Prediction Analyses using the PARFUME Code," EDF-5741, Rev. 1, April 2007.
2. K. Hammond, "AGR-2 Pretest Prediction Analyses Using the PARFUME Code for US Fuel Particles," ECAR-1020, May 2010.

RAI FQ-B63/MST-B63: The lack of pre-irradiation characterization data on the NGNP prismatic fuel compacts for the AGR-1 and the AGR-2 tests in the FQ White Paper is surprising. Pre-irradiation characterization data and as-fabricated fuel quality data for the fuel compact lots of AGR-1 and AGR-2 have been published elsewhere [Refs. 1 & 2].

- Provide the fuel pre-irradiation characterization data and the measured as-fabricated quality data for the AGR-1 and AGR-2 fuel compact lots. Discuss the comparison of the AGR-1 and AGR-2 specification as-fabricated quality specifications (Table 2, Reference 1) to the NGNP prismatic fuel performance requirements listed in Table 16 (page 72). Discuss how these data relate to the in-reactor performance of the AGR-1 fuel.
- Compare the as-fabricated fuel quality data for those compacts containing South African UO2 TRISO fuel particles to the NGNP pebble-bed fuel requirements and prior German manufacturing data.

These data would provide a more thorough understanding of the status of manufacturing UCO TRISO fuels and NGNP prismatic fuel compacts.

References:

1. J. A. Phillips, C.M. Barnes and J.D. Hunn, "Fabrication and Comparison of Fuels for Advanced Gas Reactors," Proceedings of HTR-2010, Prague, Czech Republic, Oct. 18-20, 2010, paper 236.
2. 2. IAEA TECDOC-CD-1645, Section 5.3, pages Table 7).

Comment: The pre-irradiation characterization data and as-fabricated quality data for the compacts containing UCO TRISO fuel particle in irradiation tests AGR-1 and AGR-2 have been available for a number of years. These results have been published in at least two international sources. Failure to include this data and discuss with respect to as-manufactured fuel quality requirements is a significant oversight. The two tables presented below were obtained from Reference 1. They present the as fabricated quality of the prismatic fuel compacts containing UCO TRISO fuel variants irradiated in theAGR-1 test, and compacts with either UCO TRISO fuel particles or UO2 TRISO fuel particles now under irradiation in the AGR-2 test.

Table 3. Comparison of burn leach defects (in) particles before compacting (Reference 1).

Fuel Type	Burn-leach Defects	Sample Size	Ratio	95% CL
AGR-1 Baseline	0	120,688	0	≤2.5E-5
AGR-1 Variant 1	1	121,117	8.3E-06	≤4.0E-5
AGR-1 Variant 2	1	50,265	2.0E-05	≤9.5E-5
AGR-1 Variant 3	1	120,660	8.3E-06	≤4.0E-5
AGR-2 UCO	5	217,159	2.3E-05	≤4.8E-5
AGR-2 UO2	1	120,000	8.3E-06	≤3.9E-5
German particles [9]	102	3,300,000	3.1E-05	≤3.6E-5

Table 4. Comparison of defects found in deconsolidated particles by LBL (Leach-Burn-Leach) [Reference 1].

Fuel Type	Defects Found	Sample Size	Ratio (U_{free}/U_{total})	95% CL
Uranium Contamination (pre-burn leach)				
AGR-1 Baseline	0	99,470	0	≤3.1E-5
AGR-1 Variant 1	0	74,699	0	≤4.1E-5
AGR-1 Variant 2	0	99,100	0	≤3.1E-5
AGR-1 Variant 3	0	99,032	0	≤3.1E-5
AGR-2 UCO	3	317,625	9.4E-06	≤2.5E-5
AGR-2 UO2	3	246,840	1.2E-05	≤3.2E-5
SiC Defects (post-burn leach)				
AGR-1 Baseline	2	49,735	4.0E-05	≤1.3E-4
AGR-1 Variant 1	0	49,799	0	≤6.1E-5
AGR-1 Variant 2	1	49,555	2.0E-05	≤9.6E-5
AGR-1 Variant 3	0	49,516	0	≤6.1E-5
AGR-2 UCO	0	254,100	0	≤1.2E-5
AGR-2 UO2	0	123,420	0	≤2.5E-5

Response FQ-B63/MST-B63:
The fabrication data for AGR-1, while technically interesting, were not considered relevant, since it was a laboratory-scale produced fuel. The fabrication data for AGR-2, published in October 2010 [1], were not available when the white paper was written (March–May 2010) and published in June 2010, and were not considered important to meet the objectives of the white paper review.

The as-manufactured quality (e.g., heavy metal contamination) for AGR-1 was purposely set at 10E-4 in recognition that the fuel was to be fabricated at laboratory scale and the United States had not done such fabrication in over a decade. The values for the mean heavy metal contamination for the baseline and three variants of fuel in AGR-1 are: 3.64E-07 (baseline), 2.75E-07 (Variant 1), 2.61E-07 (Variant 2), and 1.26E-07 (Variant 3). These values are consistent with the fission gas R/B measured on-line of between 1E-08 and 1E-07. The SiC defect specification was 10E-4, and the measured values were 4E-05 (baseline), 0 (Variant 1), 2E-05 (Variant 2) and 0 (Variant 3). As of this writing, one particle with defective SiC was found in PIE of AGR-1 baseline fuel in Capsule 6, consistent with this QC data.

For AGR-2 the as-manufactured heavy metal contamination fraction was set at 2E-05, which is representative of values used by prismatic vendors. UO_2 and UCO AGR-2 fuel met this specification at 50% confidence but failed at 95% confidence. The 95% confidence heavy metal contamination for AGR-2 UCO was 2.5E-05 and for UO_2 was 3.2E-05. These levels were considered acceptable for the

irradiation. The higher than desired levels of contamination were associated with through coating cracking of particles associated with unloading from the CVD coater. The unloading system was redesigned to minimize potential impact of particles with hard metal surfaces that could cause particle damage. Recent coating runs in the AGR program have resulted in contamination levels below the specification value of 2E-05 indicating that the handling concerns have been addressed. The SiC defect specification for AGR-2 was 10^{-4}, which is consistent with the reactor specification. Both the UCO and UO_2 met that specification by large margin.

The initial AGR-2 R/B values (~5-6E-07 for the 1250°C UCO capsules, 1.4E-06 for the 1400°C UCO capsule, and ~8E-08 for 1150°C UO_2 capsule) are consistent with these levels of contamination indicative of one exposed kernel in the UCO capsules and between 0 and 1 exposed kernel in the UO_2 capsule. A preliminary comparison of the R/B from the AGR-2 capsules and model predictions (PARFUME) for release from exposed kernels and contamination are shown in Figure 4.

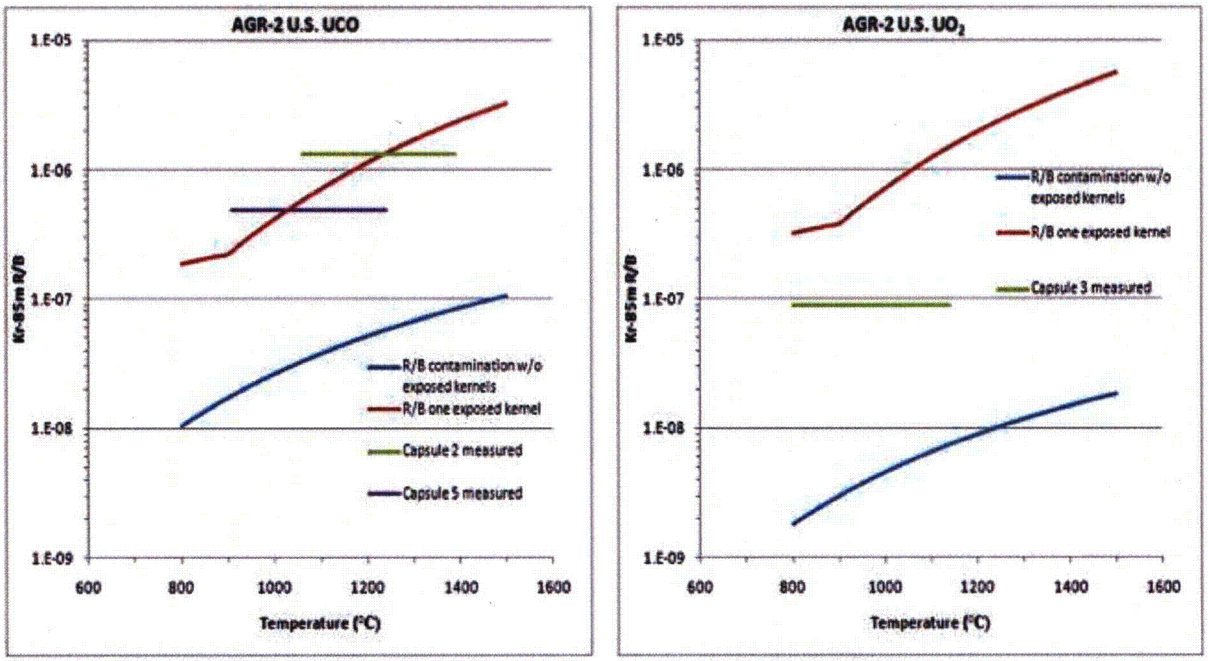


Figure 4. Results of preliminary comparison of the R/B from the AGR-2 capsules and model predictions (PARFUME) for release from exposed kernels and contamination.

It is noted that quality data on the properties of South African UO_2 fuel irradiated in the AGR-2 irradiation test, beyond that contained in the references cited in this set of RAIs is proprietary to PBMR (Pty) Ltd.

RAI FQ-B64/MST-B64: An overpower transient will heat fuel particles from the inside. The fuel testing described heats the particles uniformly from the outside. This will affect the stress distribution in the coating layers and could impact performance. Can it be shown that the heating test performance is representative of fuel particle performance during an overpower transient?

Comment: A difference in the temperature distribution in the particle (hot kernel with cooler matrix in an overpower transient versus uniform temperature increase in the heating tests) will cause two different thermal stress distributions. The thermal stress could drive failure mechanisms such as fuel particle coating cracking or fuel particle layer debonding.

Response FQ-B64/MST-B64:
As discussed in the response to RAI FQ-33, the thermal time constant of the coated particles is on the order of 40 ms, much shorter than the time frame of credible overpower transients. In addition, the high heat capacity of the fuel matrix (and graphite for the prismatic core) slows the thermal response of the matrix surrounding the particles during overpower transients. For example, as discussed in the response to RAI FQ-B27/MST-B27, the thermal time constant of a fuel pebble is on the order of tens of seconds. Thus the particle internal temperature distributions can be addressed by quasi-static analysis, and the difference in particle temperature distributions between heating tests and overpower transients is associated with the power generation in the kernels during an overpower transient (the fuel particles are essentially isothermal during the long slow heating tests). The thermal characteristics of the graphite, matrix, and coated particles are such that the maximum temperature rise from the coolant to the fuel at rated power is relatively low (<250°C) compared to the maximum temperature rise in LWR fuel pins at rated power, which can approach 2000°C. Thus the fuel can withstand overpower transients reaching several hundred percent of rated power over periods of tens of seconds to minutes. During the course of the licensing application review, when safety analysis results are submitted for the NGNP, stress levels in the particles will be analyzed by fuel performance models for limiting overpower events. Based on the results discussed below, the analyses are expected to demonstrate substantial margins to particle failure at the peak power locations in the core.

Using the results of PARFUME calculations for a 425 micron UCO TRISO particle, estimates of the effects of overpower on the peak kernel temperature, gas pressure in the particle, and SiC layer stress can be made. In the PARFUME calculations, a quasi-static thermal calculation is used to calculate the temperature in the coating layers, buffer, and kernel during its exposure in the reactor. Results typically show that the temperature rise across the TRISO coating layers (OPyC/SiC/IPyC) is very low, on the order of 2°C, and most of the temperature rise is in the buffer and kernel. As burnup accumulates, the power decreases due to depletion of the U-235, fission gas is generated and released (modeled using a diffusion and knockout model), and the kernel swells (1% $\Delta V/V$ per % FIMA). As fast neutron radiation damage occurs, the PyC and buffer layers begin to shrink. The buffer shrinks much more than the PyC because of its low density. The buffer shrinkage thus allows a gap to be formed, which is an important thermal resistance to be modeled as the neutron exposure accrues. The shrinkage of the PyC layers add significant compressive stress to the SiC layer, offsetting the tensile stress generated by gas pressure. All of these effects are calculated by PARFUME in an integral self-consistent manner. Many of the effects are non-linear, so results from a PARFUME calculation are used in the table below to provide key parameters from which simple scaling estimates can be made.

Table 1: Results of overpower on particle centerline temperature, gas pressure and SiC layer total stress

				Matrix at 1250 C					Matrix at 1400 C				
	Nominal Power (mW/particle)	Gap (microns)	Delta T to Kernel Center	Kernel CL Temperature	Gas Pressure (Mpa)	Gas stress (Mpa)	Shrinkage and Thermal Stress (Mpa)	Total Stress	Kernel CL Temperature	Gas Pressure (Mpa)	Gas stress (Mpa)	Shrinkage and Thermal Stress (Mpa)	Total Stress
1X Power													
~ BOL	100	2	20	1270	0	0	-100	-100	1420	0	0	0	0
MOL	25	13	70	1320	5	25	-325	-300	1470	10.0	50	-200	-150
EOL	6	16	45	1295	10	50	-125	-75	1445	20.0	100	-150	-50
2X Power													
~ BOL	100	2	40	1290	0	0	-100	-100	1440	0	0	0	0
MOL	25	13	140	1390	5.2	26	-325	-299	1540	10.4	52	-200	-148
EOL	6	16	90	1340	10.3	51	-125	-74	1490	20.5	103	-150	-47
5x Power													
~ BOL	100	2	100	1350	0	0	-100	-100	1500	0	0	0	0
MOL	25	13	350	1600	5.9	29	-325	-296	1750	11.6	58	-200	-142
EOL	6	16	225	1475	11.1	56	-125	-69	1625	22.1	110	-150	-40

The table shows a set of results for nominal full power operation including two particles, one in the matrix at 1250°C (considered to be a nominal maximum temperature particle) and one in the matrix at 1400°C (a maximum temperature particle considering uncertainties). The first set of columns includes the following:

- Representative values of the nominal maximum particle power at beginning of life, middle of life, and end of life are shown in the first column, reflecting the exponential decrease in power expected with increasing burnup.
- The PARFUME-calculated gap that grows with exposure as the buffer shrinks with fast fluence and the kernel swells with burnup. This gap fills with fission gas, and the thermal resistance of that gas is used to calculate the temperature drop across the gap.
- The PARFUME-calculated increase in temperature from the matrix to the kernel center, including the effect of the gap. It is small initially (~ 20°C), increases to ~70°C at middle of life, but then decreases toward end of life because the power has decreased.

The two sets of columns for matrix temperatures of 1250 and 1400°C include the following:

- Kernel centerline temperature obtained by adding the particle delta T to the matrix temperature.
- The PARFUME-calculated gas pressure and the associated tensile stress in the SiC layer.
- The SiC layer stress associated with thermal stresses and shrinkage related to the PyC shrinkage and irradiation induced creep that changes with both temperature and time.
- The SiC layer total stress in the base case from the PARFUME calculations. The total stress is the sum of the negative (compressive) shrinkage and thermal stresses and the positive tensile stress due to gas pressure.

The next sets of results in the table represent the effect of instantaneous 2x and 5x increases in power for a particle in a 1250°C and 1400°C matrix at beginning of life (BOL), middle of life (MOL), and end of life (EOL) respectively. The impact of the increased power is an increase in the kernel temperature. That increase in temperature will increase the pressure in the particle due to the ideal gas law and hence the tensile component of the SiC stress. It is assumed that these overpower events are very short in duration (on the order of minutes given the strong negative temperature coefficient in the reactor) so that additional gas release from the kernel is small because of the limited time at temperature. Thus, in all cases, even when the kernel temperature is as high as 1600-1700°C, the resulting effect on tensile stress is small. The small increase in tensile stress has a very small impact because the SiC layer remains in compression over its entire design lifetime. Tensile stresses in the SiC layer would have to be in the range of +200 to +300

MPa to fail the SiC layer based on its strength of ~300-400 MPa. Therefore modest duration overpower transients do not pose a threat to TRISO fuel integrity.

RAI FQ-B65/MST-B65: In performing heatup tests on irradiated fuel to simulate heatup accident conditions, the heat is supplied from the outside the fuel compact. In the real accident situation, the heat is generated by nuclear decay inside the fuel kernels. Describe how these differences between actual conditions and test conditions affect kernel, coating, and matrix temperatures through a fuel compact and how these differences are considered in applying the test results to accident analysis

Response FQ-B65/MST-B65:
The heatup tests address bounding fuel temperature conditions for DLOFC accidents. In these events, the maximum fuel temperature initially decreases and subsequently slowly increases to peak temperatures over the course of several days. Thus, decay heat levels during the time of elevated fuel temperatures are less than 1% of rated power. Under these conditions, the total temperature gradient through the matrix and fuel particles is less than 2°C. The heatup tests involve either slow ramps to isothermal test conditions or transients simulating the time dependence of a DLOFC, with temperature drops through the matrix and fuel particles a fraction of a degree cesium. These small temperature differences are not significant in terms of fuel performance or fission product transport.

RAI FQ-B66/MST-B66: In discussing NRC requirements for fuel and fuel qualification, it would be useful to describe the basis for each of the fuel-related General Design Criteria and address the applicability of those GDCs to the NGNP. If a particular GDC is not directly applicable to the NGNP design, then identify analogous criteria for the NGNP and state how those criteria will be satisfied.

Response FQ-B66/MST-B66:
An assessment of the applicability of the fuel-related and other General Design Criteria (GDCs) to the HTGR is part of a larger NGNP Project effort that was recently completed, the Regulatory Gap Analysis (RGA). The RGA was conducted to evaluate existing NRC regulatory requirements and guidance against the design characteristics specific to a generic modular HTGR. A summary report on the RGA has been provided to the NRC for information [1].

Applicability of the GDCs will also be addressed in future license applications.

Reference:

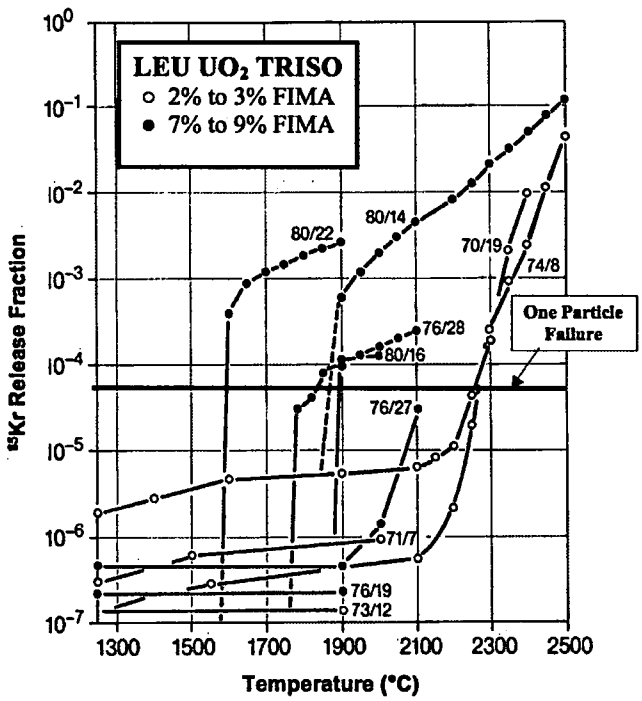
1. "NGNP Project Regulatory Gap Analysis for Modular HTGRs," INL/EXT-11-23216, September 2011

RAI FQ-B67/MST-B67: The "heating test krypton release results" presented in Table 6 for 1700°C and 1800°C have different exposed kernel fractions (at 50% confidence interval (CI) and 95% CI) than similar data presented in Table 8 and Tables A-4 and A-5. This is the result of including additional heating test data in Table 6 from non-reference fuel test specimens irradiated in MTR irradiation tests (SL-P1 and FRJ2-P27). This is why the numbers of particles listed in Table 6 are different for the 1700°C and 1800°C heating tests. The resulting confusion makes it difficult to track the exposed kernel fraction data in Chapter 3 with that of Chapter 4 and Appendix A. It is not clear that the value of the heating test data conducted on non-reference fuel specimens is sufficient to warrant their inclusion. Please clarify how the NGNP Project interprets and uses these data.

Response FQ-B67/MST-B67:
The RAI statement is correct. The results in Table 6 for 1700°C and 1800°C incorrectly included heating test data on nonreference fuel specimens, while the data in Table 8 are limited to full sized spheres from either the GLE-3 or low-enriched uranium Phase I fuel campaigns. If the material in Section 3.2 is retained in the final version of the Fuel Qualification White Paper, this error will be corrected. However, this section may be deleted for reasons stated in the supplemental information regarding pebble bed fuel qualification provided in 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. As stated in CCN 223977, the NGNP project would include irradiation and testing of sufficient quantities of fuel to demonstrate that statistical fuel performance requirements (particle failure fractions) are met without relying on the use of historical German data.

RAI FQ-B68/MST-B68: Please clarify the following statement in the FQ White Paper: "The degradation of fuel performance at elevated temperatures is regular and gradual. No sudden changes in behavior (cliff-edge effects) as a function of irradiation temperature, burnup, or accident temperature were seen."

Comment: Sudden bursts or changes in release rates were observed in ramp tests of irradiated AVR spheres to high temperatures. See for example the Kr-85 release fraction data for AVR elements 76/28, 80/22, 80/14 and 80/14 in the figure copied below [W. Schenk and H. Nabielek, Jül-Spez 487, January 1988, Fig. 57 pg. 195.]. The sudden bursts were a result of complete particle failure and occurred without prior warning. The requested clarification should (a) clearly specify the ranges of relevant parameters (e.g., irradiation temperatures, burnup levels, fluence levels, accident temperatures, etc.) over which no sudden changes in fuel behavior (cliff-edge effects) have been seen, (b) indicate the types of fuel behavior considered in this context (e.g., coating layer failures, diffusive release through intact coating layers, etc.), and (c) identify supporting technical references as appropriate.



Fractional release of 85Kr measured during accident simulation testing of GLE 3 spherical fuel elements irradiated to various burnups in the AVR

Response FQ-B68/MST-B68:
Fuel performance is defined and used in the fuel qualification paper as the particle failure fraction under normal operation or accident conditions. The ^{85}Kr release fraction during heating tests, plotted on a logarithmic scale, exhibits a sudden large increase upon failure of a single particle. This is the result of the very high retention of krypton in intact particles and the absence of failed particles prior to the heating tests because of high as-manufactured fuel quality and irradiation performance. The sudden increase in ^{85}Kr release associated with failure of one or a few particles in a sphere containing over 14,000 particles does not constitute a major sudden change (cliff-edge effect) in terms of fuel performance (particle failure fraction). The data shown in the comment indicate that the failure fraction for fuel at or near full design burnup increases gradually even at temperatures well beyond conservative maximum temperature predictions for accident conditions.

The gradual onset of particle failures observed in heating tests for a population of particles with a similar irradiation history and accident conditions can be associated with outlier particles in the statistical variations in particle properties, such as kernel diameters and coating thicknesses, within the population. In addition to these variations in particle properties, an HTGR core would also consist of large variations in irradiation history and accident condition temperature; both first order variables in terms of fuel performance. These variations in particle properties and service conditions preclude monolithic fuel performance that could produce a sudden large increase in the core average fuel particle failure fraction (a cliff-edge effect).

RAI FQ-B69/MST-B69: Please correct or clarify the statement on FQ page 37 that "no exposed-kernel failures were observed in any of the MTR irradiations." While it is true that no exposed-kernel failures were observed in any of the MTR irradiations conducted on full-size 60 mm-diameter spherical elements, nine failed LEU UO2 TRISO particle failures were in fact recorded in-reactor from fuel particles irradiated in non-reference fuel specimens (20 mm-diameter spheres, cylindrical compacts and coupons) in MTR irradiation tests SL-P1 and FRJ2-P27.

Response FQ-B69/MST-B69:
As discussed in further detail below, German analysis of the irradiation results from the non-reference test articles (small fueled spheres contained in cylindrical matrix "compacts") irradiated in SL-P1 and FRJ2-P27 concluded that there were no failures during irradiation. It was also concluded that some of the compacts contained as-manufactured defective particles at the beginning of the irradiation. The compacts were produced by forming a small fueled zone (2 cm diameter versus 5 cm diameter for prototypical spheres) surrounded by a thick unfueled matrix layer (2 cm versus 0.5 cm for prototypical spheres). The unfueled layer of the spheres was then machined to form a cylinder for placement in the irradiation rig. These non-prototypical fuel forms received much less fabrication process development than prototypical spheres and had a higher particle packing fraction in the fueled zone (~24% for FRJ-P27 and ~16% for SL-P1 versus ~10% for the GLE-3 spheres). Thus, it is not unexpected that they would exhibit a higher as-manufactured defective particle fraction than prototypical spheres.

The German interpretation of the SL-P1 irradiation data, as reported in Section 4.3.2 of the KFA report [1], concludes that there was an as-manufactured defect, but there were no particle failures during irradiation of SL-P1.

The evaluation reported in Section 5.1 of HTA-IB-7/92 [2] supports the conclusion that there were no particle failures in SL-P1 during irradiation. The report also discusses the results of FRJ-P27, concluding that Capsules 1 and 2 each contained about 5 particles with as-manufactured defects. The summary evaluation of irradiations FRJ-P27, HFR-P4, and SL-P1concludes that relatively higher initial R/B values suggested as-manufactured particle defects, with no indication of irradiation induced failures.

References:
1. W. Schenk, R. Gontard, and H. Nabielek, "Performance of HTR Fuel Samples under High-Irradiation and Accident Simulation Conditions, with Emphasis on Test Capsules HFR-P4 and SL-P1," October 1994.
2. Hans-Jürgen Hantke, "Performance of High Quality HTR-LEU Fuel Elements with TRISO Coated Particles," December 1992.

RAI FQ-B70/MST-B70: Please clarify the intended meaning of FQ Figure 11 and the data it represents in terms of possible NGNP accident conditions. Any data point not plotted as "zero" on the linear ordinate would seem to have no relevance to modular HTGR designs like NGNP. Data in the NGNP-relevant range of 10-9 to 10-6 would appear here as "zero" on the linear ordinate. More meaningful would be a figure like that provided here in RAI FQ-B68/MST-B68 (see above) [Ref.], which shows the release fraction for 85Kr measured continuously during ramp heating tests of German irradiated AVR fuel elements.

<u>Reference</u>: W. Schenk and H. Nabielek, Jül-Spez 487, January 1988, Fig. 57 pg. 195.

Response FQ-B70/MST-B70:
As correctly stated in this RAI, any data point not plotted as "zero" in Figure 11 of the Fuel Qualification White Paper has no quantitative relevance to accident performance of modular HTGR designs like NGNP. The intended meaning of Figure 11 is to show that fuel performance under extremely high temperature accident conditions depends on the thermal stability of the SiC coating layer, not on variations in burnup, fast-neutron fluence or kernel composition. As noted in the FQ White Paper, the extremely high temperatures in Figure 11 are associated with accident conditions in large HTGRs >600MW as opposed to the lower temperatures encountered in modular HTGRs. It should also be noted that ramp tests disguise time-at-temperature effects on fuel particle failure.

RAI FQ-B71/MST-B71: The following statement on page 41 of the FQ White Paper should be clarified or corrected: "TRISO-particle fuels have been ... used as the fuel in seven power and experimental reactors." TRISO-particle fuels have in fact been used in only five power and experimental reactors [AVR, Dragon, FSV, HTTR and HTR-10]. While Peach Bottom did contain some test fuel with TRISO coatings in its FTE Program, the THTR employed only fuel with BISO-HTI coatings.

Response FQ-B71/MST-B71:
The observation contained in this RAI is correct. Upon resolution of the other RAIs associated with the Fuel Qualification White Paper, the paper will be revised to state that TRISO particle fuel has been used as the driver fuel in five power and experimental reactors and as a test fuel in one other power reactor.

RAI FQ-B72/MST-B72: The following statement on page 46 of the Fuel Qualification White Paper, seems premature and weakly supported at this stage of UCO fuel development: "The heating test data for the German UO_2 fuel are considered to be generally applicable to the UCO fuel being developed by the NGNP/AGR Fuel Program." While the AGR-1 fuel fabrication campaign appears to have succeeded in producing high-quality HTGR fuel with good operating performance under AGR-1 irradiation testing conditions, more data on operating and accident condition performance would be needed to justify any application of heating test data on UO_2 fuel to the UCO fuel of interest. At present, there appears to be no test data on U.S. UCO TRISO fuel that suggest the applicability of past German accident condition test data. Please describe the existing test data, if any, that the NGNP Project interprets as supporting the applicability of UO_2 TRISO fuel heatup test data to U.S. UCO TRISO fuel.

Response FQ-B72/MST-B72:
As discussed in Section 3.3.3 of the Fuel Qualification White Paper, results from limited heating tests of batches of TRISO-coated fuel particles in which particles with UCO kernels and particles with UO_2 kernels were tested indicate that kernel composition does not appear to have a major influence on the performance of the fuel particles under accident conditions, but the structure of the SiC layer may be important. The SiC layer in UCO fuel fabricated for irradiation and safety testing in the NGNP/AGR Program has a structure that, by design, is intentionally similar to that of German UO_2 fuel particles. The safety testing planned by the NGNP/AGR Program is extensive and is intended to produce a database sufficient to qualify NGNP fuel without reliance on results from historical German safety testing; however, comparison of results from the NGNP/AGR Program with those from the German tests will be of interest.

RAI FQ-B73/MST-B73: The basis for the failure fraction values in Table 11 under normal operation is unclear. All the reference 60 mm-diameter spherical fuel elements containing LEU UO2 TRISO fuel

irradiated in MTRs (HFR-K3, FRJ2-K13, FRJ2-K15, HFR-K5 and HFR-K6) represent a total population of ~286,240 particles. [This number is larger than the MTR data provided in Table 4 because the test FRJ2-K15 (which contained AVR 21-2 fuel elements) is included.] Based on this total population with "0" defects, the 50% and 95% upper confidence intervals for a binominal distribution are 2.42 x10-6 and 1.05 x10-5, respectively. The failure fraction numbers quoted in Table 11 require "0" failures in a population of ~79,672 particles. There is no indication of what set of test elements make up this population. Possibly, it represents a specific subset of the elements irradiated in the MTR tests. Please explain how these data were obtained.

Response FQ-B73/MST-B73:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B74/MST-B74: Failure fraction data for heating tests presented in FQ Table 12 is again different from that presented in FQ Tables 6, 8, A-3, A-4, or A-5. It is not clear which are the appropriate numbers to be used for normal operating conditions and accident conditions. The failure fraction data should be correct and provide consistent indications of fuel performance for both normal operating and accident conditions. Please clarify.

Response FQ-B74/MST-B74:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B75/MST-B75: Based on the performance data presented in FQ Chapter 3, Chapter 4, and Appendix A, there is little confidence in the NGNP pebble-bed failure fraction values presented in Table 13 and plotted in Fig. 22. Provide a consistent basis for the correct set of failure fraction numbers along with details on how they relate to prior German UO2 fuel as-fabricated quality, irradiation performance, and accident testing.

Response FQ-B75/MST-B75:
The response to this RAI will be deferred to a future date consistent with the supplemental information regarding pebble bed fuel qualification provided in 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-B76/MST-B76: Please provide more detail on the normal operating conditions and accident conditions for the prismatic NGNP concept. The closely related MHTGR 350 MW design has been around since the mid-1980s and the GT-MHR 650 MW design since the mid-1990s. From the "anticipated" maximum service conditions provided in the FQ White Paper, it appears that the prismatic

UCO fuel will experience a much more severe environment than the pebble bed UO_2 fuel. This is summarized in the following table.

Parameter	Prismatic NGNP (Table 14)	Pebble-Bed NGNP (Table 10)
Max Fuel Temperature – Normal Operations (°C)	1400	1048
Maximum Time-Averaged Fuel Temperature (°C)	1250	644
Fuel Temperature – Accident Conditions (°C)	1600	1483
Fuel Burnup (% FIMA)	17	8.75
Fast Fluence (x1025 n/m2, E>0.18MeV)	5	2.39

Response FQ-B76/MST-B76:
The normal operating and accident conditions for the prismatic NGNP concept listed in the above table are design goals and are more challenging than the best estimate values for the pebble bed NGNP concept. (The prismatic concept utilizes a higher enrichment and produces a higher average discharge burnup.) The UCO fuel kernel is engineered to remain stable under these more challenging conditions, as will be demonstrated by the NGNP/AGR Program.

RAI FQ-B77/MST-B77: In FQ Table 16, the fuel performance requirements for the prismatic UCO TRISO fuel list the heavy metal (HM) contamination requirement as ≤1.0 x10-5 (Expected – 50% CL) and ≤2 x10-5 (Design – 95% CL). Does this number represent the Ufree/Utotal fraction as explained for the German spherical element? The NGNP design value for the German spherical element is 6 x10-5 and compares well with the total NUKEM, GmbH, fabrication data which was demonstrated in the production of ~60,000 spherical elements in production-scale facilities. The HM contamination value for the prismatic NGNP fuel is 3 to 6 times smaller than the German design value. What is the basis for this requirement and are there any modern (post NPR) US fuel compact production data that substantiate the feasibility for this level of production quality?

Response FQ-B77/MST-B77:
The heavy metal contamination specification for prismatic fuel ($\leq 1 \times 10^{-5}$ maximum expected and $\leq 2 \times 10^{-5}$ design value) represents heavy metal contamination outside of the fuel particles in the compact matrix and in particles with exposed kernels. It is measured by leaching a statistically sufficient sample of compacts to demonstrate that the 95% confidence heavy metal contamination is less than the design value specification. If any exposed kernels are present in the as-manufactured compacts, this enriched uranium would show up in the measurement. The fraction of particles with defective SiC in prismatic fuel compacts is measured by oxidizing the compacts that had previously been leached, and leaching the residue. This second leach identifies heavy metal in particles with defective SiC layers that had been previously protected by an intact PyC layer. The specification for this quantity (SiC defects) is $\leq 5 \times 10^{-5}$ maximum expected and $\leq 1 \times 10^{-4}$ design value.

U_{free}/U_{total}, in the specification for heavy metal contamination in the German spherical pebble bed fuel element ($\leq 6 \times 10^{-5}$ design value), is measured by first oxidizing a sample of spheres and then leaching the residue. Thus, this quantity captures heavy metal contamination in the matrix and particles with defective SiC. Since the German characterization procedure does not distinguish between heavy metal contamination and SiC defects, analysis of gaseous fission product release must assume the measured value is all from heavy metal contamination (particles with an SiC defect but an intact PyC layer would not release gaseous fission products).

Thus, a self-consistent comparison with regard to metallic fission product release is the value of 1.2×10^{-4} (prismatic compacts) versus 6×10^{-5} (pebble bed spheres) for design values of heavy metal contamination plus fraction of particles with defective SiC. A comparison with regard to gaseous fission product release is 2×10^{-5} (prismatic compacts) versus 6×10^{-5} (pebble bed spheres) for design values of heavy metal contamination to be used for safety analysis. These are two different historic approaches for establishing fuel quality that still exist for the two technologies.

The manufacture of fuel with quality levels close to the specifications for the prismatic design has been demonstrated with prototypic scale equipment for the AGR-2 irradiation (see RAI FQ-B63/MST-B63).

RAI FQ-B78/MST-B78: Does the heavy metal (HM) contamination requirement in FQ Table 16 also include Th-232 HM contamination in the fuel components, namely the fuel compact or fuel element matrix materials? Studies by South African researchers have shown that the primary source of fission gas release, in the absence of defective fuel particles, in MTR irradiation tests (HFR-K5/K6) is the breeding of U-233 from thorium contamination in the matrix material of spherical elements [Ref.] that contain high quality TRISO fuel particles (Ufree/Utotal values of ~1.35 x10-5). Will there be a requirement on thorium contamination for NGNP fuel? What were the HM contamination values on the high-quality fuel compacts prepared for the AGR-1 and AGR-2 irradiation experiments, including uranium (natural and enriched), thorium, and any other fissile or fertile materials?

Reference: van der Merwe, J.J., Verification and Validation of the PBMR models and Codes Used to Predict Gaseous Fission Product Releases from Spherical Fuel Elements. MSc. Thesis, Rand Afrikaans University, South Africa, 2003.

Response FQ-B78/MST-B78:
The heavy metal contamination specification is total uranium metal. See the response to RAI-FQ-B63/MST-B63 for values for AGR-1 and AGR-2 fuel. However, isotopic analysis of the contamination found in AGR-1 compacts indicates that the contamination is material of the same enrichment as the kernels and not related to impurities in materials of fabrication. Thus, the thorium contamination of the graphite holders used in the German irradiation experiments and the associated release of fission gas from bred U-233 is not a concern for the current U.S. AGR irradiations. In addition, the levels of other impurities in AGR TRISO fuel are much lower values than historical data indicating the higher quality of starting materials and cleaner fabrication processes. Thus, the historical contamination issues raised in this RAI are not expected to be a significant issue with modern TRISO fuels given the higher quality of the starting materials used in fabrication. Hence, no specification on thorium is anticipated in the fuel compact specification.

RAI FQ-B79/MST-B79: Please clarify how the FQ Table 16 requirement for 110mAg release for NGNP fuels applies to accident conditions. Based on prior German isothermal heating test performed on an irradiated HFR-K3 sphere containing high-quality LEU UO2 TRISO particles, the 110mAg release was >10-2 upon reaching the soak temperature of 1600°C. [Ref.] This element was irradiation at operating temperatures in the range of 1000°C to 1200°C to a burnup of 7.5% FIMA and a fluence of ~4 x1025 n/m2 (E> 16fJ). These conditions are significantly below those listed in Table 14 for the prismatic NGNP concept.

Reference: W. Schenk, G. Pott and H. Nabielek, J. Nucl. Mat. 171 (1990) pp. 19-30.

Response B79/MST-B79:
The core release limits for Ag-110m and Cs-137 given in Table 16 of the Fuel Qualification (FQ) White Paper are for normal operation and anticipated operational occurrences (AOOs) only; they do not apply to accident conditions, such as depressurized core conduction cooldown accidents. Since Ag-110m is not expected to be a dominant off-site dose contributor for such accidents, no quantitative limits on Ag-110m release have been established for such events at this writing. There were also no quantitative limits on Ag-110m release during these events in the Preliminary Safety Information Document (PSID) for the steam-cycle MHTGR. Such accident release limits for volatile fission metals will be established for the NGNP, as needed based on contribution to off-site or worker dose, later in the design process.

As indicated in the RAI, it is likely that the core-average fractional release of Ag-110m during depressurized core conduction cooldown accidents would exceed the normal operating limit of 2×10^{-4} given in Table 16 of the FQ White Paper. However, as illustrated in Section 4.2.2.2 for a pebble-bed core example, only ~3% of the core would experience temperatures in the range of 1400 to 1500°C, while no spheres are expected to exceed 1600°C. Typical results for prismatic cores are comparable. Consequently, the core-average fractional release of Ag-110m during core heatup accidents can be expected to be much smaller than the $>10^{-2}$ release fraction quoted in the RAI. Moreover, a considerable fraction of the Ag-110m (and other volatile metals) released from the core can be expected to deposit along leakage pathways in the primary circuit and reactor building; consequently, only a small fraction of the metals released from the core would be expected to be ultimately released to the environment.

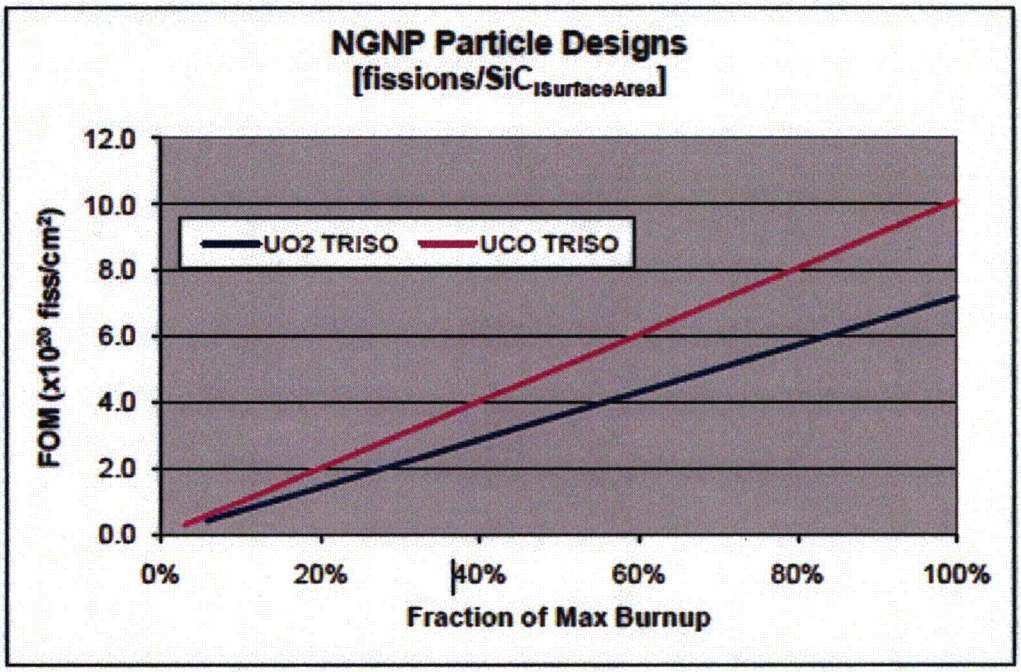
Computational tools and material property data are available to quantify the aforementioned phenomena. See Appendix D and Appendix E in the Mechanistic Source Terms White Paper for more information on the codes used in these analyses for prismatic and pebble bed modular HTGRs, respectively.

RAI FQ-B80/MST-B80: Please comment on the validity and significance of the observations presented in the comments that follow.

Comments: The UCO TRISO particle design envisioned for the NGNP concept is quite different and will undergo significantly different operating conditions than the UO_2 TRISO particle design in the NGNP pebble-bed concept. The UCO fuel kernel, nominally 425 μm-diameters, will have an enrichment of 14% U-235. The UO_2 fuel kernel is nominally 500 μm-diameters with an enrichment of 4.2 to 4.5% in the startup core and 7.8% subsequently. Although the TRISO coating design was not presented for the UCO particle design, the assumption is that it will be similar to the TRISO coating applied to the UO_2 fuel kernel. From a cursory evaluation of the operation conditions each of these TRISO fuel particle designs (Tables 10 and 14) will experience, it appears that the UCO particle in the NGNP prismatic concept will encounter a much more severe operating environment than the UO_2 particle in the pebble bed concept.

The fission density in the UCO fuel kernel will be significantly higher than that in the UO2 fuel kernel. At the peak burnups of each, the fission density in the UCO kernel (at 17% FIMA) will be ~11.3 times that of the UO_2 kernel (at 8.75% FIMA). Assuming that each concept adopts a similar TRISO coating design (95 μm-thick Buffer, 40 μm-thick iLTI, 35 μm-thick SiC and 40 μm-thick oLTI), then the SiC layer in the UCO fuel particle will have a smaller inner surface area than that of the UO_2 particle (~23% smaller). Under normal operating conditions at peak burnup, the number of fissions/cm2 of inner SiC surface in the UCO particle is ~45% higher than for the UO_2 TRISO particle. Thus, the flux of fission products to the SiC surface will also be ~45% higher for the UCO particle. The figure below illustrates how this difference in fission product flux will vary with accumulated burnup for each particle design. Coupling the higher fission density and the fission product flux with the differences in operating conditions in the prismatic [higher operating temperatures (up to 400°C) and ~2X in fluence] versus the

pebble-bed concepts means that the UCO fuel particle design will experience a significantly more severe operating environment than the UO_2 fuel particle design.



Response FQ-B80/MST-B80:
The UCO TRISO particle design does encounter more severe operating conditions in a prismatic core than the UO_2 TRISO in a pebble bed core. However, the degree of severity is much less than implied in the RAI. In the table below are a number of TRISO particle designs including:

a. Historic German UO_2 TRISO 10.6% enriched and a burnup of 10% FIMA,

b. Japanese UO_2 TRISO in HTTR 6% enriched and a burnup of 3.6% FIMA,

c. Japanese advanced UO_2TRISO 10% enriched and a burnup of 10% FIMA,

d. A hypothetical 425 μm UO_2 particle 15% enriched and a burnup of15% FIMA,

e. A hypothetical 350 μm UO_2 particle 19.9% enriched and a burnup of 20% FIMA,

f. A 200 μm HEU TRISO used in the US NPR program, 93% enriched and a burnup of 80% FIMA,

g. The 350 μm fissile TRISO from the GT-MHR design, 19.9% enriched and a burnup of 26% FIMA,

h. The AGR 425 μm fissile particle 15% enriched and a burnup of 15% FIMA, and

i. The 500 μm fertile particle from the GT-MHR design 0.7% enriched and a burnup of 6% FIMA.

The behavior of these particles was compared to three different metrics:

- Gas pressure (dependent on fission gas and CO (if applicable) and available buffer volume)

- Stress due to gas pressure ($\frac{P_{gas} d_{SiC}}{\delta_{SiC}}$) where P_{gas} is the gas pressure, d_{SiC} is the diameter of the particle at the SiC layer, and δ_{SiC} is the thickness of the SiC layer
- Fission product corrosion ($\frac{V_{ker} Bu}{A_{SiC}}$), where V_{ker} is the volume of the kernel, Bu is the burnup and A_{SiC} is the inner surface area of the SiC layer which collectively is a measure of flux of fission products at SiC surface

These metrics were normalized to 1.0 for the German particle. The results in the table below suggest that the 425 μm particle experiences much less gas pressure and gas stress than a German particle (metrics of 0.47 and 0.43) and a corrosion metric of 1.1, only slightly above the German TRISO particle. The lack of CO pressure in the UCO and the smaller surface to volume ratio of the AGR UCO TRISO particle compensate for the effects of greater burnup (more fission products) and smaller kernel size. (The metric in the RAI considers only the number of fissions not the number per unit volume of kernel material. The surface to volume ratio in the metric in the table below takes that into consideration.)

Comparison of Different Particle Designs

Parameter	German	JAERI HTTR	JAERI Advanced	425 micron UO2	350 micron UO2		US NPR	US LEU Fissile	AGR 425 micron	US Fertile
Kernel Composition	UO_2	UO_2	UO_2	UO_2	UO_2		UCO	UCO	UCO	UCO
Kernel Diameter, microns	500	600	550	425	350		200	350	425	500
Buffer Thickness, microns	95	60	100	100	100		100	100	100	65
IPyC Thickness, microns	40	30	35	40	40		50	40	40	40
SiC Thickness, microns	35	30	35	35	35		35	35	35	35
IPyC Thickness, microns	40	45	40	40	40		40	40	40	40
Enrichment, %	10.6	6	10	15	19.9		93	19.9	15	0.7
Max. Burnup, % FIMA	10	3.6	10	15	20		80	26	15	6
*Gas pressure figure of merit (kernel volume*burnup*(FG yield + CO yield if applicable)/buffer volume)*	*4.24*	*3.41*	*4.49*	*4.75*	*4.79*		*3.31*	*2.62*	*2.00*	*1.74*
Normalize gas pressure FOM	*1*	*0.81*	*1.06*	*1.12*	*1.13*		*0.78*	*0.62*	*0.47*	*0.41*
*Stress figure of merit = gas pressure FOM*d-sic/thickness SiC*	*93.2*	*88.7*	*105.3*	*95.6*	*86.2*		*47.3*	*47.1*	*40.2*	*35.3*
Normalize gas stress FOM	*1*	*0.95*	*1.13*	*1.03*	*0.92*		*0.51*	*0.51*	*0.43*	*0.38*
*Solid Fission Product Corrosion FOM ((Volume of kernel*burnup)/surface area of SiC)*	*0.35*	*0.21*	*0.41*	*0.39*	*0.36*		*0.43*	*0.47*	*0.39*	*0.25*
Normalized solid fission product corrosion FOM	*1*	*0.61*	*1.17*	*1.10*	*1.02*		*1.21*	*1.33*	*1.10*	*0.71*

Using these metrics, this comparison demonstrates that the 425 μm UCO particle is overall less challenged than a 500 μm German UO_2 particle. Furthermore, the potential for CO corrosion of the SiC layer in UO_2 TRISO is not accounted for in this analysis which would further demonstrate that the 425 μm UCO TRISO particle has performance advantages over the 500 μm UO_2 TRISO.

RAI FQ-B81/MST-B81: Does the fuel design for the NGNP prismatic concept employ fertile particles? Does the prismatic fuel design employ fissile particles of a single enrichment or multiple enrichments? Previous U.S. prismatic HTGR designs, up thorough the mid-1990s, employed a two particle system consisting of both a fissile particle and a fertile particle, the latter containing either thorium or natural uranium. Recent GA pre-conceptual Modular Helium Reactor concepts (GA-A25401 and GA-A5402, April 2006) also employed a two particle system with an enriched UCO TRISO fissile particle and a

natural UCO TRISO particle as the fertile particle. If a fertile particle is no longer employed, how are flux profiles flattened during normal operation? Are multiple enrichments, multiple compact packing fractions, and/or burnable poisons employed to achieve desired power shapes?

Response FQ-B81/MST-B81:
At this time, a reactor concept has not been selected for the NGNP demonstration plant. However, GA has conducted core performance analyses for various prismatic modular HTGR core designs that use a single fuel particle of the design currently being qualified by the NGNP/AGR Fuel Development and Qualification Program with one or more U-235 enrichments. The results of these analyses indicate that an acceptable core design can likely be obtained with a single fuel particle design and single U-235 enrichment, but that the use of multiple U-235 enrichments would improve the design, particularly with respect to the power peaking in the approach-to-equilibrium cycles. Multiple U-235 enrichments (with a fissile and fertile particle being the extreme case), fuel loading zoning, and burnable poison zoning can all be used by the reactor designer as means of achieving acceptable axial and radial power shaping within the core.

RAI FQ-B82/MST-B82: Quantify how thermocouple (TC) failures in the AGR tests affect uncertainties in the irradiation temperatures of the fuel and fuel element materials.

Comment: The response should quantify the added uncertainties in the temperatures of fuel and fuel element materials (i.e., local capsule temperature histories, capsule time-average peak temperatures, and capsule time-average volume-average temperatures) that result when multiple TCs fail in AGR tests. Of particular interest are the fuel temperature uncertainties associated with the specific TC failures identified in the completed AGR-1 test and the ongoing AGR-2 test. The response should include or reference a description of test thermal models and how they are used to support the evaluation of temperatures and temperature uncertainties in the AGR irradiation tests.

Response FQ-B82/MST-B82:
Temperatures in both the fuel (AGR) and advanced graphite capsules (AGC) are measured using thermocouples embedded in the graphite sample holders. Temperatures are calculated using a detailed ABAQUS finite element model for the test train. The detailed finite element model (about one million elements for an AGR capsule) considers

- Conduction in all of the test train materials and the gas mixture in the control gap
- Radiation heat transfer across gaps, especially the thermal control gap of the experiment,
- Degradation in thermal conductivity with fast fluence in graphitic materials, and
- Shrinkage/swelling of graphitic components with fluence and temperature.

Heat rates in the ABAQUS thermal model come from a detailed Monte Carlo Neutral Particle (MCNP) code analysis of the position and energy dependent neutron flux in the test article during each reactor cycle in the ATR. For the graphite irradiations, heat rate from the metallic components are most critical. For the fuel irradiations, the MCNP calculations take into account changes in fission heat with burnup of the fuel, burnout of the boron in the graphite, and heat rates of thru tubes and other metallic components in the test train. In addition, the effect of the hafnium shroud (and its depletion) in the AGR-1 and AGR-2 experiments is explicitly modeled.

The thermal calculations for the graphite capsules indicate that the finite element model is within ±40°C of the thermocouple measurements. Thermocouples have not failed in the AGC experiments and do not drift significantly at the lower temperatures in those experiments, so a detailed comparison between model and data can be made. This good agreement validates the overall methodology and suggests that a detailed thermal model can accurately calculate the temperature of graphite specimens.

For the fuel capsules higher risk of thermocouple drift and failure complicates the execution of the experiments because as the fuel burns out more insulating gas must be added to the control annulus to compensate for decreases in fuel compact fission power (neon flow adjusted to maintain thermocouple temperature set points). Two approaches are being used in the AGR program to control the experiment in the event where all thermocouples in a capsule have failed. As-run physics and thermal calculations are performed prior to each cycle to determine if gas mixes need to be adjusted to maintain target fuel temperature. The model is benchmarked against operating thermocouples early in the irradiation before drift and irradiation-induced changes in the graphitic components occur to ensure that the model is a good predictor of both thermocouple and fuel temperature. In subsequent cycles, differences in measured versus calculated temperature response are indicative of thermocouple drift. If the comparison between the measured and calculated temperatures at the thermocouple location is good (< 50°C difference), the uncertainty associated with the calculated fuel temperature should not be that much larger given the short distance from the thermocouple to the fuel and the high conductivity of the graphite components. This was the approach used for the AGR-1 irradiation.

The daily peak and daily average temperatures for AGR-1 are also being compared to the R/B measured from each capsule. Because there was no fuel particle failure in AGR-1, the R/B from the heavy metal contamination is very sensitive to temperature and thus is a measure of the temperature in each capsule. The R/B always increased in AGR-1 indicating that temperatures remained high and increased with time during the irradiation. There is a strong correlation between the temporal temperature predictions and the temporal R/B, indicating that the thermal model is capturing all of the important physical phenomena.

For AGR-2, in parallel with the use of the thermal model to calculate fuel and thermocouple temperatures, a detailed statistical analysis called control charting is used as part of the NGNP data management system to track changes of each thermocouple pair in the capsule. If the thermocouple pair differences fall outside of the control band, it is an indication of drift of at least one thermocouple in the pair. That thermocouple would be watched more closely, and the thermocouple temperature set point would be adjusted to include the bias value calculated from as-run simulation. If the thermocouple fails totally during irradiation, the statistical model has "learned" what its behavior should be based on prior operating cycles and thus is used to "predict" what it should be were it to still be operational. Combining these two approaches reduces overall uncertainties related to gas mix settings for an upcoming reactor cycle to maintain temperature.

Final as-run calculations for AGR-1 are currently underway. Physics and thermal models are being run for each day of the irradiation to correspond to the detailed R/B data available from the experiment. The PIE of AGR-1 indicated that the graphite fuel holder swelled (due to boron addition to the graphite) and the control gap shrank; however, the gap between the graphite body and the fuel compact increased, offsetting somewhat the reduction in control gap. Thus, the AGR-1 temperatures are being recalculated to account for this behavior. The thermal model is being compared to thermocouple data between 50 and 150 days, when the experiment was at temperature and the greatest number of thermocouples was operational.

Differences between measured and calculated temperatures are generally ±35 to 100°C depending on the capsule.

Sensitivity analyses using the thermal model for the AGR experiments have been completed indicating that the two most important parameters are control gap size and fuel heat rate. Of less importance are the conductivities of the fuel compact and the graphite holder and the emissivity of the metallic components. In general the uncertainty in control gap size should be very small (~5%) because it should not change and fuel heat rates calculated by MCNP should be accurate to statistical limitations of the code given the large amount of geometric detail of the experiment and ATR in the MCNP model. However, for AGR-1, the control gap shrinkage was non-uniform axially within a capsule and varied between capsules because of the different levels of boronated graphite used in the different capsules. This will complicate the assessment of temperature uncertainties for AGR-1. Preliminary as-run calculations for AGR-2 are showing better comparisons between measured thermocouple responses and calculated temperatures because the level of boron in the AGR-2 graphite is below that which showed swelling in AGR-1.

Detailed uncertainty calculations for AGR-1 were underway at the time of this writing that will also include uncertainty contributions from covariance effects of the key modeling parameters and any modeling biases. These should be available in the next 3 to 6 months along with detailed reports of the final AGR-1 physics and thermal analyses and the sensitivity analysis.

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