Request for Additional Information No. 5771 Revision 0

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Next Generation Nuclear Plant Pre-Application Activities
Department of Energy - Idaho National Laboratory
Docket No. PROJ 0748
SRP Section: ARP MST - Mechanistic Source Term
Application Section: RAI on CCN 221271 - NGNP Mechanistic Source Terms White Paper

QUESTIONS for Advanced Reactor Branch 1 (ARB1)

ARP MST-1

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

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

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

(Section 5) {FQ-1, FQ-4 (MST-5), FQ-5 (MST-6)} [2(a), 3]

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

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

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

· Plutonium fission generally accounts for a large and variable fraction of the total burnup in high-burnup LEU fuels. For a given initial uranium enrichment and total fuel burnup, the magnitude of the plutonium fission fraction will vary with changes in the neutron energy spectrum. An HTGR spectrum may convert more uranium to plutonium than the softer spectra in water-cooled MTRs like the ATR and FRJ2 (DIDO). Furthermore, for a given content of plutonium in relation to U-235, the hotter thermal neutron spectrum in an HTGR, which typically peaks near the 0.3 eV fission resonances of Pu-239 and Pu-241, will more strongly favor plutonium fission over U-235 fission.

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

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

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

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

- Reducing the TRISO fuel's initial enrichment
- Hardening the MTR's thermal neutron spectrum
- Increasing the MTR's epithermal neutron spectrum
- Replacing some UO_2/UCO in the fuel kernels with PuO_2/PuCO.

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

"Physics of TRISO fuel irradiation in test reactors versus HTGRs:

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

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

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

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

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

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

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

ARP MST-2

RAI MST-2: Describe in detail how the temporary increase in Cs diffusivity through SiC that is expected to be present during irradiation and absent afterward will be evaluated from the results of past and planned tests and experiments. (Section 5) {FQ-2 (MST-2), FQ-16 (MST-21)} [2(a), 3]

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

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

For example, recent experiments and atomistic simulations have suggested that lattice vacancies play an important role in both the solubility and the diffusion of Cs in SiC.[1] It is well known that neutron irradiation produces not only extended defects such as dislocation loops and voids but also temporary lattice vacancies and interstitials that disappear soon after irradiation stops.[2] These non-equilibrium vacancies and interstitials would likely increase solubility of Cs in SiC and accelerate Cs diffusion during irradiation.

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

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

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

ARP MST-3

<u>RAI MST-3</u>: Describe how the NGNP prototype plant will use startup and operational tests, operational monitoring, and periodic confirmatory measurements and inspections, in conjunction with license conditions and other regulatory controls, to confirm and monitor that initial and evolving NGNP operating conditions and performance are consistent with those predicted and considered for licensing. (Sections 1-6, Appendix C) {FQ-3 (MST-3), FQ-23 (MST-21), FQ-26 (MST-31)} [2, 3]

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

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

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

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

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

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

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

(d) Temperature distributions in the core and/or at the core outlet will likely have to be periodically measured to confirm and monitor that operating temperature profiles and histories remain consistent those for which the fuel and other materials and components have been designed, qualified, and analyzed. Such temperature measurements in past HTGRs have employed a variety of sensor technologies including "NATE" sensors,[1] thermocouples, noise thermometers, and melt-wires. The following observations bear noting in this context:

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

○ Recent analyses of the AVR's melt-wire experiment have shown that the neglect of core bypass flows was likely a major contributor to the >250 °C under-prediction of peak core operating temperatures by the computational models used for AVR design and safety analysis.[2], [3]

○ In the THTR, the predicted and actual core bypass flows were reported as 7% and 18%, respectively.[4] However, because the THTR ran for only 1.2 full-power years, no data were obtained on the increases in bypass flows that generally occur with the irradiation-induced shrinkage of HTGR reflector blocks.

○ At FSV, measurements in the core outlet plenum showed radial temperature gradients significantly larger than expected. These were attributed to higher-than-predicted bypass flows caused in part by excessive pressure drops across the helium inlet orifices above each fuel block stack.[5]

(ii) The inherent long-term instability of the radial profiles of pebble-flow rate, power, and temperature (caused largely by high pebble friction in helium that decreases with increasing temperature – see related comments under RAI MST-7) gave rise in the THTR to a large inner-core hot spot[6] that tended to intensify with operating time. Despite effective monitoring of evolving outlet temperature radial profiles in the bottom reflector and the use of adaptive zonal loading strategies to counteract the pebble flow profile instability, the evolution of increasingly excessive helium outlet temperature gradients in the THTR appears to have contributed to the early failure of 35 of the ~2,600 hot-duct insulation attachment bolts in just 1.2 full-power years of operation.

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

○ have highly variable and uncertain bypass flows between blocks within the core in addition to those in the reflectors,

○ may be susceptible to various local "closed-core" under-cooling effects including undetected local blockages of fuel coolant holes,

○ may keep fuel in potential hot spots for many months at a time, and

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

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

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

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

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

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

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

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

ARP MST-4

RAI MST-4: Provide a more definitive discussion of "event-specific mechanistic source terms" that considers the comments provided below. (Sections 1, 2, 3, and 4) {MST-4, MST-84} [1]

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

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

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

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

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

ARP MST-5

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

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

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

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

(Sections 5 and C-4) {FQ-1 (MST-1), FQ-4 (MST-5), FQ-5 (MST-6)} [2(a), 3]

Comments: The requested knowledge summary should draw on the current body of relevant technical literature, including the TRISO fuel phenomena identification and ranking table (PIRT) results documented in NUREG/CR-6844 (2004) and references cited therein and related information such as [1], [2], [3] as appropriate. The discussion should address related aspects of RAIs MST-1 and MST-2 and specifically describe the technical basis for INL's following statement in TEV-1022:

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

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

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

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

ARP MST-6

RAI MST-6: In reference to RAI MST-1 and TEV-1022, provide the detailed MTR design and test information needed to perform independent modeling of (a) the completed, ongoing, and planned AGR TRISO fuel and material irradiations in the ATR MTR and (b) the past TRISO fuel irradiations in the FRJ2 (DIDO), HFR, and Siloe MTRs. (Sections 3, 5) {FQ-1 (MST-1), FQ-4 (MST-5), FQ-5 (MST-6)} [2(a), 3]

Comments: The NRC staff will use the requested information to perform selected independent calculations of the spectral burnup isotopics and fission product inventories in TRISO fuel irradiated in the ATR and other MTRs. The independent calculations will help the staff further judge the applicability of MTR fuel irradiation tests and understand potential issues with the nuclear analysis tools and approximations used in simulating the tests as well as the NGNP core. Such analysis issues would involve, among others, (i) the uses of PIE isotopic data in the test analyses, (ii) the modeling of fuel double heterogeneity, (iii) the effects of differences between the historic nuclear yield, decay, and branching data used in the ORIGEN2.2 computer code and the updated data used in the ORIGEN-S computer code, and (iv) the different treatments of the temperature- and neutron-spectrum-dependence of nuclide-specific nuclear reaction rates in the JMOCUP computer code and modeling methods used by INL and the SCALE computer code methods used by NRC (e.g., the specific nuclides and temperatures for which spectrum-specific cross sections are calculated and used by the respective ORIGEN codes).

ARP MST-7

RAI MST-7: For the pebble-bed NGNP design (or similar) described in the white papers, identify the basic design provisions and operating strategies that will be used to monitor and control the long-term inherent instability of pebble flow-rate, power, and temperature profiles. If the evolution and effects of such pebble flow profile instabilities are believed to be adequately known, small, negligible, or somehow non-existent, describe the technical basis for that belief considering the comments that follow. (Appendix B) {FQ-6} [2, 3]

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

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

The accelerating passage of pebbles through the inner core region thus gave rise to a large inner-core hot spot that tended to intensify over the THTR’s short operating lifetime. This occurred despite (i) the ability in THTR to monitor this unanticipated effect using fuel residence-time data inferred from the fuel handling system as well outlet temperature profiles measured by the “NATE” sensors in the bottom reflector, and (ii) the ability in THTR to dampen and control the effect by using the central and peripheral loading tubes to implement adaptive inner- and outer-core loading strategies (i.e., loading absorber pebbles and more highly burned fuel pebbles into the inner core region and fresher fuel pebbles into the outer core region).[1], [2]

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

One might expect such pebble flow profile instabilities to be less pronounced in modern cylindrical-core pebble bed designs (e.g., the depicted pebble-bed NGNP design and similar designs such as China's High Temperature Reactor Power Module (HTR-PM), with their relatively slender cores and lack of in-core rods, and to have considerably less bearing on the annular-core designs considered until recently. With regard to the potentially exacerbating effects of floor debris on pebble flow profile gradients, it is noted that without in-core rods the AVR reported known pebble fractures at rates ranging from 1 per ~3,000 pebble circulations in the first years (mainly with U.S.-made graphite-shell-type fuel pebbles) to 1 per ~10,000 to 40,000 circulations in the final years of its 21-year operation (almost entirely with German-made A3-pressed fuel pebbles). It would thus appear that pebble fractures cannot be precluded in the pebble-bed NGNP design and will have to be considered in safety analysis and licensing. (See related comments for RAIs MST-3 and MST-8.)

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

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

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

· From recent studies at INL,[3] one can conclude that the ability to simulate multiple cycles of friction-mapped pebble flow in HTR-PM-sized cores with pebble bed mechanics codes like INL's PEBBLES computer code remains elusive, especially with the repetitions needed for nuclear and thermal feedbacks as well as parametric and uncertainty studies. Years of supercomputing clock time would be required unless major breakthroughs are achieved in speeding up the codes.

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

· Until practical tools are developed to rigorously model pebble flow mechanics with locally variable friction and nuclear and thermal coupling, useful studies can and should be done using simplified approximations of pebble flow mechanics. Approximations based on the apparent physical similarities[4] (e.g., similar flow velocity profiles) between pebble flow and the laminar flow of viscous liquids would likely be worth pursuing, noting that both liquid viscosity and pebble friction decrease with increasing temperature. For example:

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

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

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

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

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

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

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

[4] G. Murphy, *Similitude in Engineering*, Ronald Press Co., 1950.

ARP MST-8

RAI MST-8: Describe how the potentially extended core residence times of TRISO fuel in intact pebbles, broken pebbles, and pebble fragments are factored into the fuel qualification plans and treated in mechanistic source terms calculations for the pebble bed NGNP design. (Section 5.2) {FQ-7} [2(a), 3]

Comments: Experience from the AVR as well as the THTR (e.g., see related comments under RAI MST-7) suggests the likelihood that at least a small population of TRISO fuel in intact pebbles, broken pebbles, and pebble fragments will be subjected to highly extended or indefinite core residence times. For example, it is expected that some broken pebbles and pebble fragments will find stable resting positions on the core floor before reaching the core discharge chute. These stationary fragments could then also slow or permanently block the flow of neighboring intact pebbles to the discharge chute.

The fuel population thus subjected to greatly extended or indefinite core residence times could then experience burnup and fluence levels well beyond those for which the fuel is otherwise to be qualified. Fuel performance and release calculations should account for the resulting potential for elevated failure fractions and reduced retentiveness in the affected populations of CFPs, along with their relatively high irradiation temperatures at the bottom of the core (as well as the relatively limited accident heatup temperatures at that location) and their elevated fission product inventories corresponding to high or extreme burnup. The RAI response should also consider how the size of the affected fuel population will be estimated or assumed and what measurements or inspections will be performed to ensure that the actual affected populations are within expected or assumed limits. Related comments are also provided under RAIs MST-3 and MST-7.

ARP MST-9

RAI MST-9: Provide an analytical study evaluating the proposed use of a Fick's Law bulk diffusion approximation to model the combinations of more complex transport phenomena noted in Appendix Section C-4 (page 71) of the MST white paper. (Section C-4) [2, 3]

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

ARP MST-10

RAI MST-10: Derive and explain the use of release-to-birth ratio (R/B) in terms of total mass balance (i.e., birth, decay, burnout, release) and the fission product inventory releases (including metallic fission products) resulting from transport through intact, defective, and failed CFPs. Identify the cases and situations in which it is not appropriate to use R/B in place of the release fraction. (Sections 3 and 5, Appendix A) {FQ-8 (MST-10), FQ-9 (MST-11), FQ-18} [2(a), 3]

ARP MST-11

RAI MST-11: Describe how the nuclide birth and inventory values used in evaluating past and planned MTR-based TRISO-fuel-irradiation tests are calculated. For example, will the values for AGR-1 be taken from the same JMOCUP and MCNP computer code ATR model calculations mentioned in TEV-1022? If not, explain in detail how the AGR values will be calculated and checked/calibrated against PIE data. (Sections 3 and 5, Appendix A) {FQ-8 (MST-10), FQ-9 (MST-11), FQ-18} [2(a), 3]

ARP MST-12

RAI MST-12: Clarify the statement on page 3 of the MST white paper regarding the generation and transport of each radiologically significant species of fission product. Explain whether the stated functional dependencies should also include diffusion through kernel and intact coating layers as functions of operating and accident conditions. Explain whether the list of conditions should also include neutron fluence and plutonium burnup. (page 3) [2, 3]

ARP MST-13

RAI MST-13: Provide an analytical study evaluating how fast neutron fluence > 0.1 MeV correlates to material damage (i.e., the total fluence spectrum folded with the respective material's displacements-per-target-atom (dpa) damage response function) for the specific material fluence spectra encountered in (a) NGNP service and (b) the irradiation tests conducted in MTRs for the fuel materials addressed in the white paper. The study should report how the evaluated correlations of >0.1 MeV fluence to material damage vary between the different spectral conditions encountered in material service versus testing. (Sections 3 and 5, Appendix A) {FQ-10} [2, 3]

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

ARP MST-14

RAI MST-14: Provide an expert review of the conflicting explanations for how carbonaceous dust is produced in pebble-bed and block-core HTGRs. Key reference documents should be provided and translated as necessary to facilitate the review.

Summarize the review findings, including any significant outstanding questions and recommendations for resolving them, and revise or supplement the associated information in Table 5-1 and on page 48 of the MST white paper as warranted. (Sections 4 and 5, Table 5-1, page 48) [2(b), 3]

Comments: [Note: Efforts in response to this RAI should be coordinated with the ongoing INL/NRC interactions on dust issues.] The NRC requester is aware of two conflicting explanations for how the observed amounts of dust were produced in the AVR. On the one hand, an article published in 2008 by Rainer Moormann of Forschungszentrum (Research Center) Jülich (FZJ) attributes the source of the AVR's finer dust mainly to abraded A3-3 pebble matrix binder coke, noting its especially high affinity for cesium.[1] On the other hand, a German article published in 1990 by Rudolf Nieder, the AVR's resident chemist, concludes that most of dust was chemically produced from catalyzed reactions of the relatively high CO and H_2 impurity levels in the AVR primary circuit.[2] A translated excerpt from the latter article follows:

"Carbon Transport

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

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

*can run in the reverse direction under certain conditions, i.e., carbon is deposited from mixtures of carbon monoxide and hydrogen [2]**[3]**. These conditions are*
- suitable catalyst,
- relatively high CO and H_2 *concentrations,*
- reducing gas atmosphere.

Deposition of carbon occurs mainly in the presence of metallic catalysts, e.g., with Fe but not metal oxides. Iron carbides are also effective catalysts. With highly alloyed steels, which have a protective oxide layer of chromium oxide or iron-chromium-spinels, a reduction to elemental iron in an HTR primary circuit atmosphere is not possible. On the other hand, with the carbon and low-alloyed steels used in the AVR steam generator, the Fe3O4 protective layer originally present is reduced to elemental iron at H_2/H_2O *ratios >10. Such relatively high hydrogen concentrations were always present in the AVR as the result of diffusion from the water-steam loop. Applying laboratory results to the AVR primary loop shows carbon deposition to have already started after a relatively short incubation time, whereupon continuing deposition led to the formation of iron carbides with a significantly stronger catalytic effectiveness [3]**[4]**.*

*Using a simplified mass balance [4]**[5]** one can estimate a dust production rate of 3 to 3.5 kg per year, meaning that after 21 years of reactor operation there should have been about 60 kg of chemically produced dust. The additional production of dust by mechanical means would likely be modest by comparison. Pebble breakage occurred only very occasionally. Pebble wear during operation was only trivial; machining marks were still largely recognizable on visually inspected discharged pebbles. Grain-*

*size analysis of dust samples showed particle diameters consistently <1 micrometer, and predominantly <0.5 micrometer [5]**[6]**, a finding likewise consistent with chemically produced dust.*

*Independent of these experimental findings, there are model calculations that can determine the carbon deposition rate for given chemical conditions such as catalysis, etc., in an HTR primary loop [6]**[7]**. Based on a rough approximation, the calculated amount of deposited carbon is found to be of the same order of magnitude as actually observed. The calculations, however, are being continued."*

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

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

[3] [2] E. Stolz, F. L. Werner, *Kohlenstofftransport in den Hochtemperaturreaktoren THTR und AVR (Carbon Transport in the THTR and AVR High Temperature Reactors)*, atw. Februar 1968, S. 99-103.

[4] [3] M. R. Everett, *Some Aspects of Carbon Transport in High Temperature Gas Cooled Reactors*, Dragon Project Report DPR 365, 1965.

[5] [4] U. Wawrzik, *Interner AVR-Bericht (AVR Internal Report)*, 1984.

[6] [5] U. Wawrzik, et al, *Staubverhalten im AVR Reaktor (Dust Behavior in the AVR Reactor)*, Jahrestagung Kerntechnik (Nuclear Technology Annual Conference), Travemünde,1988.

[7] [6] D. V. Kinsey, *The Estimation of Carbon Deposition in HTRs*, Dragon Project Technical Note DPTN/565, 1974.

ARP MST-15

RAI MST-15: Describe how the material properties of fuel pebbles can affect core operating and accident conditions and the transport of fission products, how accurately the material properties are known over anticipated service conditions (i.e., variable temperature, neutron fluence, helium impurity levels, etc.), how the properties have been measured, what additional data, if any, would be needed to adequately characterize the material properties of fuel pebbles, and what plans have been made to address such data needs. (Sections 4 and 5) {FQ-11 (MST-15), FQ-35 (MST-39)} [2, 3]

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

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

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

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

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

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

Related comments are also provided under RAIs MST-3, MST-7, MST-8, and MST-14.

ARP MST-16

RAI-MST-16: Revise the white paper to correctly refer to the A3-3 matrix material in fuel pebbles as “carbon composite,” not “graphite.” (Sections 3 and 5, Appendix A) {FQ-12} [2, 3]

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

ARP MST-17

RAI-MST-17: Where conservative estimates, approximations, assumptions, and results are noted in the white paper, describe also how best-estimate (BE) and best-estimate-and-uncertainty (BEAU) mechanistic source terms would be calculated. (Section 4.6.2, Section C-5) {MST-17, FQ-27 (MST-32), FQ-28 (MST-33), FQ-29 (MST-34), FQ-30 (MST-35, MST-68, MST-88} [1, 2, 3]

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

ARP MST-18

RAI MST-18: Identify all chemical elements (and the chemical forms thereof) that can pass through structurally intact SiC (a) during power operation at temperatures up to (i) 1250 °C and (ii) 1350 °C (i.e., as might occur in undetected hot spots) and (b) during core heatup accidents at temperatures up to (i) 1500 °C, (ii) 1600 °C, and (iii) 1700 °C. Indicate the yields of each element from the thermal-neutron-induced fission of U-235, Pu-239, and Pu-241, respectively, and estimate the total amounts that pass through the SiC layer. (Sections 4 and 5, Appendix C) {FQ-13 (MST-18), FQ-14 (MST-19), FQ-15 (MST-20)} [2(a), 3]

Comments: Include all stable and radioactive nuclides that can pass through the SiC layer, including europium, palladium,[1] and other elements, regardless of whether they have direct radiological significance. This requested information should be used to inform the response to RAI MST-19.

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

ARP MST-19

RAI MST-19: Describe the potential cumulative effects that the passage of silver and other elements could have on the retentiveness and strength of intact TRISO coating layers and the data available/needed for evaluating such effects. (Sections 4 and 5, Appendix C) {FQ-13 (MST-18), FQ-14 (MST-19), FQ-15 (MST-20)} [2(a), 3]

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

ARP MST-20

RAI MST-20: Provide a discussion of mechanisms and conditions that can increase the diffusion or permeation release of metallic fission products from intact TRISO particles. It is noted that FQ white paper describes "failure" mechanisms (in Section 3.1.2) but not mechanisms that can increase diffusive releases from that vast majority of CFPs whose SiC layers remain structurally intact. (Sections 4 and 5, Appendix C) {FQ-13 (MST-18), FQ-14 (MST-19), FQ-15 (MST-20)} [2(a), 3]

ARP MST-21

RAI MST-21: Describe in detail how the AGR test program will evaluate cesium diffusivity through intact SiC under (a) NGNP operating conditions as well as (b) accident heatup conditions. (Sections 4 and 5, Appendix C) {FQ-2 (MST-2), FQ-16 (MST-21)} [2(a), 3]

Comments: The response should build upon relevant information provided in response to RAIs MST-1, MST-2, MST-18, MST-19, and MST-20.

ARP MST-22

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

ARP MST-23

RAI MST-23: Clarify how liquid water (as opposed to water vapor) can be present in the primary circuit as mentioned in Section C-6 of the MST white paper. (Appendix C) [2, 3]

ARP MST-24

RAI MST-24: Expand on the entrainment discussion in white paper Section C-6.2 to also address the entrainment effects of non-break flow transients, break shock and vibration, and break-induced local flow reversal on dust, friable surfaces, etc. (Appendix C) [2, 3]

ARP MST-25

RAI MST-25: Indicate the fission power levels achieved by fuel reirradiation prior to heat-up testing in the AGR program and the earlier German programs in relation to those needed to match the short-lived nuclide inventories expected to be present during an NGNP heatup accident. (Section 5) {FQ-19} [2, 3]

ARP MST-26

RAI MST-26: Discuss the validation benchmark calculations that either have been or will be conducted for the German test data. (Section 5, Appendix C) {FQ-20} [2(a), 3]

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

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

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

ARP MST-27

RAI MST-27: Since the German UO_2 test data for TRISO coated particle diffusion rates are largely based on post-irradiation heating tests, discuss how the additional testing for pebble bed reactor fuel will evaluate fission product transport under high temperature irradiation. (Section 5, Appendix C) {FQ-21, FQ-37 (MST-41)} [2(a), 3]

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

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

ARP MST-28

RAI MST-28: (a) Discuss how the regulatory requirements for technical specifications (TS), as noted in the comments below, will be applied to the NGNP fuel design. (b) Discuss whether the NGNP TS will contain requirements for controlling the initial accident source terms to those assumed in the accident analyses by monitoring and limiting gaseous fission product releases (for controlling the fraction of failed fuel particles in the core during normal operations) and monitoring and limiting metallic fission product releases (for controlling releases from failed and intact fuel particles during normal operations). (Sections 2 and 3) {FQ-3 (MST-3), FQ-23 (MST-21), FQ-26 (MST-31)} [2, 3]

Comments: The FQ white paper discusses NUREG-0800, Standard Review Plan, Section 4.2, "Fuel System Design." Additionally, NUREG-0800, Standard Review Plan, Chapter 16 “Technical Specifications” states that at the Operating License stage, compliance with Title 10 Code of Federal Regulations (10 CFR) 50.34 requires applicants to propose TS in accordance with 10 CFR 50.36. 10 CFR 50.36 requires proposed TS to include the following:

- 10 CFR 50.36(c)(1)(i)(A) Safety Limits. Safety limits apply to important process variables necessary for an appropriate level of protection for the integrity of certain physical barriers that guard against the uncontrolled release of radioactive material.
- 10 CFR 50.36(c)(2) Limiting Conditions of Operation (LCOs). A TS LCO of a nuclear reactor must be established for each item meeting one or more of the following 10 CFR 50.36(c)(2)(ii) criteria:
  - (ii) Criterion 2. A process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.
  - (iii) Criterion 3. An SSC that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier.

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

ARP MST-29

RAI MST-29: Discuss how the “functional failure” mechanism indicated in the comments below will be addressed in the calculation of MSTs. (Appendix C) {FQ-24} [2(a), 3]

Comments: Significantly higher than predicted diffusion/release of metallic fission products from the coating layers of intact fuel particles could be considered a “functional failure” of TRISO coated fuel particles. Elevated and potentially unacceptably high fission product diffusion could result, for example, from significantly higher than expected fuel particle operating temperatures. See also related RAIs MST-1, MST-2, MST-18, MST-19, MST-20, and MST-21.

ARP MST-30

RAI MST-30: Discuss how analytical predictions of the NGNP fuel design (in connection with the NGNP fuel qualification program or the NGNP safety analysis) will be conducted to ensure that the fuel design bases are met. If analytical predictions are to be provided, discuss whether the analytical models and methods for these analytical predictions will be developed and evaluated in accordance with RG 1.203, “Transient and Accident Analysis Methods.” (Section 2.3.3.3) {FQ-25} [2(a), 3]

ARP MST-31

RAI MST-31: For the first-of-a-kind NGNP prototype plant, loaded with fuel from a first-of-a-kind prototype fuel production facility, discuss whether the initial fuel core loading will involve any special surveillance procedures and/or testing programs beyond those that would be planned for a follow-on commercial version of the NGNP and fuel production facilities. This might involve special fuel operational performance fission product release monitoring and/or selected fuel pebble or compact post-irradiation accident simulation testing. (Section 2) {FQ-3 (MST-3), FQ-23 (MST-21), FQ-26} [2, 3]

ARP MST-32

RAI MST-32: For pebble fuel, discuss whether/how particle defect/failure rate statistics (i.e., 50% confidence, 95% confidence) are related to the calculation of BE or BEAU mechanistic source terms for anticipated operational occurrences (AOOs), design basis events (DBEs) (and design basis accidents (DBAs)) and beyond design-basis events (BDBEs) (and emergency preparedness). Discuss the statistic that is used to develop these confidence values. (Sections 4 and 5) {MST-17, FQ-27 (MST-32), FQ-28 (MST-33), FQ-29 (MST-34), FQ-30 (MST-35), MST-68, MST-88} [2(a), 3]

ARP MST-33

RAI MST-33: For pebble fuel, will the 95% confidence or the design failure fraction values be used for all mechanistic source term predictions for all the event categories? If so, discuss the basis for developing the NGNP design failure fraction values from the NGNP 95% confidence failure fraction values shown in Table 13 and the NGNP design failure fraction curve shown in Figure 22 of the FQ white paper. (Sections 4 and 5) {MST-17, FQ-27 (MST-32), FQ-28 (MST-33), FQ-29 (MST-34), FQ-30 (MST-35), MST-68, MST-88} [2(a), 3]

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

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

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

> "Acceptable limits on the event sequence consequences and the analysis basis for the LBE categories are as follows:
>
> AOOs - 10 CFR Part 20: 100 mrem total effective dose equivalent (TEDE) mechanistically modeled and *realistically calculated* at the exclusion area boundary (EAB). For the NGNP facility, the EAB is expected to be the same area as the controlled area boundary.
>
> DBEs - 10 CFR §50.34: 25 rem TEDE mechanistically modeled and *conservatively calculated* at the EAB.

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

ARP MST-34

RAI MST-34: For prismatic fuel, discuss whether/how the particle defect/failure rate statistics (i.e., 50% probability, 95% probability) are related to the mechanistic source term calculation bases and LBE consequence limits for AOOs, DBEs (and DBAs) and BDBEs (and emergency preparedness) cited in Section 3.2 of the MST white paper and Section 1.4 of the LBE white paper? Discuss the statistic used to develop these probabilities. (Section 3) {MST-17, FQ-27 (MST-32), FQ-28 (MST-33), FQ-29 (MST-34), FQ-30 (MST-35), MST-68, MST-88} [2, 3]

ARP MST-35

RAI MST-35: For prismatic fuel, discuss how the 95% probability failure fraction values will be used for all mechanistic source term predictions for the event categories? (Section 3) {MST-17, FQ-27 (MST-32), FQ-28 (MST-33), FQ-29 (MST-34), FQ-30 (MST-35), MST-68, MST-88} [2, 3]

ARP MST-36

RAI MST-36: In FQ Section 4.3, Prismatic Fuel, no figure similar to Figure 22 (for pebble fuel) is provided. For the prismatic approach, describe the source of the data to be used for developing the codes for conservatively predicting particle failure fractions versus fuel operating conditions (e.g., temperature, burn-up, or time at temperature) to be used in the NGNP safety analysis. Describe the source of the data to be used for validating the codes. Discuss the statistical probability levels (e.g., 50%, 95%) applicable to the failure fractions predicted by the codes used in the prismatic analytical approach. (Sections 4 and 5) {FQ-31} [2(a), 3]

ARP MST-37

RAI MST-37: Compare the conservatism of the pebble-bed empirical approach to the conservatism of prismatic-block analytical approach with respect to the predicted design failure fractions. Compare the statistical statements associated with the predicted particle failure fractions for each fuel type. (Section 4 and 5) {FQ-32} [2, 3]

ARP MST-38

RAI MST-38: Describe the air ingress testing plans for pebble fuel and block fuel. (Section 5) {FQ-34, MST-81} [2, 3]

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

ARP MST-39

RAI MST-39: Discuss whether the matrix-graphite abrasion test, or any other test, will be performed to verify the assumed in-reactor fuel pebble matrix dust generation rate (e.g., g/core pass) over the fuel lifetime. (Section 5) {FQ-11 (MST-15), FQ-35 (MST-39)} [2, 3]

ARP MST-40

RAI MST-40: Discuss whether and how the irradiation, PIE, and safety testing will provide any separate effects data needed to demonstrate and confirm that the rates of fission product transport through the individual fuel particle coatings and through the matrix material are consistent with those of the reference German fuel particles and matrix material. If confirmatory data will not be collected, discuss the technical basis for assuming that the fission product transport (e.g., diffusion) rates associated with the NGNP fuel will be the same as the German fuel. (Section 5) {FQ-36} [2, 3]

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

ARP MST-41

RAI MST-41: Since the German UO_2 test data for TRISO coated particle diffusion rates are largely based on post-irradiation heating tests, discuss how the additional testing for pebble bed reactor fuel will evaluate fission product transport under high temperature irradiation. (Section 5, Appendix C) {FQ-34 (MST-38), FQ-37, MST-81} [2, 3]

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

ARP MST-42

RAI MST-42: For pebble fuel, discuss whether irradiation tests will involve fuel temperature transient conditions which simulate the cyclical fuel temperature changes (shown in FQ Figure 16) that will occur over time during power operation due to the re-circulation passage of fuel pebbles through the core. (Section 5) {FQ-38} [2, 3]

ARP MST-43

RAI MST-43: Discuss the technical basis that was used for developing the “design failure fraction” values in Table 13 and Figure 22 for the HTR-Modul. Discuss the technical basis that will be used for developing the design failure fraction values in Table 13 and Figure 22 for the NGNP PBMR fuel. Given the significant reliance of the predicted pebble-bed NGNP fuel failure probabilities on the historical German data, discuss the basis for projecting that the pebble-bed NGNP design failure fractions will be less than the HTR-Modul fuel design failure fractions as shown in the table and figure. (Table 13, Figure 22; Appendix A) {FQ-39} [2(a), 3]

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

ARP MST-44

RAI MST-44: The German LEU UO_2 TRISO fuel irradiation data was developed using quality assurance procedures that may not be consistent with the NRC regulations in 10 CFR 50 Appendix B. Discuss the compensatory measures, if any, will applied in using the German data to offset any material deficiencies in the conduct of the German tests and the analysis of the German test data. (Section 5) {FQ-40} [2, 3]

ARP MST-45

RAI MST-45: In the FQ discussion concerning particle effects, the effect of temperature change on the particles is not listed. Discuss the effects of temperature changes from fabrication, room temperature, normal operation, and abnormal operation. (Appendix C) {FQ-41} [2(a), 3]

ARP MST-46

RAI MST-46: Neither swelling nor creep (either from high temperatures or from irradiation) is specifically mentioned on FQ page 13. Discuss how these phenomena will be considered. (Appendix C) {FQ-42} [2(a), 3]

ARP MST-47

RAI MST-47: Given the FQ white paper statements noting “the greater emphasis on fuel retention of radionuclides rather than reactor building retention following an event,” and given that meeting the specifications needed to meet required performance capability “requires precise process control,” provide a detailed discussion of the process control and characterization procedures that will be applied to fuel fabrication. (Section 5) {FQ-43} [2(a), 3]

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

ARP MST-48

RAI MST-48: Indicate the ranges of particle design parameters (e.g., buffer layer thickness) and service phenomena (e.g., kernel swelling) that would lead to significant coupling between the kernel and dense coating layers. (Appendix C) {FQ-44 (MST-48), FQ-45 (MST-49)} [2(a), 3]

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

ARP MST-49

RAI MST-49: A disadvantage of UC is increased radiation-induced swelling. What is the swelling rate of UCO, and what service conditions could further increase the gas evolution of UCO? (Appendix C) {FQ-44, FQ-45} [2(a)]

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

ARP MST-50

RAI MST-50: Expand the discussion in FQ Section 3.1.2 to address other conceivable failure mechanisms beyond those listed on FQ page 19. For instance, can differential thermal expansions and large temperature swings that create stresses lead to particle failure? (Appendix C) {FQ-46} [2(a)]

ARP MST-51

RAI MST-51: Please elaborate on how the FQ list of failure mechanisms is constructed and refined. Are these what the authors consider are the "credible" failure mechanisms, or are these failure mechanisms that have been that have been observed in the past? Describe the basis and reasoning to conclude that this list is comprehensive. (Appendix C) {FQ-47} [2(a)]

ARP MST-52

RAI MST-52: The two lists of failure mechanisms on FQ pages 19 and 23 do not appear to completely match. (For instance, IPyC cracking, OPyC cracking, and IPyC debonding seem to be categorized differently between the lists. Also issues of “Diffusive release through intact layers,” “SiC permeability/SiC degradation,” and “SiC failure via heavy metals in buffer/IPyC” are found on one list and not the other.) Discuss the differences between the two lists. (Appendix C) {FQ-48} [2(a)]

ARP MST-53

RAI MST-53: Discuss “SiC permeability/SiC degradation” in more detail than presented in FQ pages 19 and 23 and Table 1, and how it is distinguished from “diffusive release through intact layers” and “SiC thermal decomposition.” (Appendix C) {FQ-49} [2(a)]

ARP MST-54

RAI MST-54: The white paper identifies two main types of chemical attack on SiC: fission products and carbon monoxide (CO). High oxygen inventory may lead to excessive amounts of CO, whereas low oxygen inventory may lead to less fission product oxidation and ultimately more fission product diffusion and chemical attack. However, the impact of low oxygen inventory is not discussed in the white paper. Address the impact of low oxygen inventory. (Appendix C) {FQ-50} [2(a)]

ARP MST-55

RAI MST-55-Comment: (Note: The review objectives of the white paper can be addressed in the near term without a response to this comment. The comment may nevertheless be suitable for discussion in another forum.) Provide a discussion that summarizes the advantages and disadvantages of UCO versus UO_2 kernels for the NGNP project. {FQ-51} [2(a)]

ARP MST-56

RAI MST-56: Describe any recent or current efforts by DOE or potential vendors to develop new sensor systems for providing on-line measurements of (i) temperatures and (ii) fluxes or power densities in the cores of pebble-bed and prismatic-block HTGRs. If additional technology development is deemed necessary in this area, how is this reflected in the NGNP research and development plans? (Appendix C) {FQ-3, MST-3, FQ-52} [2(a), 3]

Comments: FQ Section 1.3 asks the NRC to “identify any additional information or testing needed to demonstrate adequate NGNP fuel performance.” One potential regulatory and licensing concern regarding safety significant parameters is on-line in-core temperature measurement. On-line measurements of in-core temperatures are inherently difficult in HTGRs and have never been performed in any HTGR operated to date. Additionally, historical operational experience has shown that core temperatures have often been higher than predicted/intended, and consequentially, fuel particle integrity was jeopardized. This issue may be a regulatory and licensing concern and could adversely affect the NGNP development timeline if not resolved promptly during the pre-application period.

ARP MST-57

RAI MST-57: For a prismatic-block NGNP design, discuss whether the radial gap between the outside diameter of the fuel compacts and the inside diameter of the fuel hole in the graphite fuel element is credited and modeled as a holdup mechanism or barrier in calculating the mechanistic source terms. (pages 3 and 8) [2(a), 3]

ARP MST-58

RAI MST-58: It is understood by the NRC Staff that the fuel irradiation and accident condition testing programs will focus on obtaining data for specific species of specific gaseous and metallic fission products. Discuss how these limited data will be used (i.e., extended) to model the transport of all of the radiologically significant species of fission products. (page 3) [2(a), 3]

ARP MST-59

RAI MST-59: Discuss the extent to which the helium purification system which is provided to remove gaseous and solid impurities (including, radionuclides) circulating in the primary system during normal operation is credited and modeled as part of the transport and release of radionuclides in the mechanistic source term calculations. (page 12) [2]

ARP MST-60

RAI MST-60: Discuss the extent to which filtration systems located in the low pressure reactor building are credited and modeled as part of the transport and release of radionuclides in the mechanistic source term calculations for normal operations, transients and accidents. (page 12) [2]

ARP MST-61

RAI MST-61: Discuss the inherent or passive safety characteristics of HTGRs that are intended to reduce or limit the extent of chemical attack (e.g., oxidation) of the fuel with its associated increase of fuel particle failures, as well as the safety characteristics of HTGRs that are intended to reduce or limit the rate of transport of radionuclides (i.e., via natural circulation of air though the core). Discuss the basis for the basis for the statement that "…steps have been taken to prevent ingress of contaminants…" given that NGNP has not yet progressed beyond the conceptual design stage. (pages 13 and 14) [2]

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

ARP MST-62

RAI MST-62: Discuss the technical basis as well as the safety basis for applying a factor of four as the design margin for the release of fission gases from the core while applying a factor of 10 as the design margin for the release of fission metals from the core to account for uncertainties in the design methods. (page 15) {MST-63, MST-64} [2]

ARP MST-63

RAI MST-63: Given the health effects importance of the chemical forms of radioactive iodine gas release and the uncertainties in the design methods, discuss why a factor of four would be considered appropriate rather than a factor of ten. (page 15) {MST-62, MST-64} [2]

ARP MST-64

RAI MST-64: Discuss whether radionuclide-specific design margins may be proposed for the NGNP that differ from a factor of four for fission gases and a factor of 10 for fission metals. (page 15) {MST-62, MST-63} [2]

ARP MST-65

RAI MST-65: Provide a few illustrative examples of the kinds of NGNP licensing basis events that would be viewed as a "major accident" to be considered under 10 CFR 52.17. Discuss whether such events would be expected to be found in the DBE or BDBE frequency range. (page 16, Table 3-1) [2]

ARP MST-66

RAI MST-66: Discuss why BDBEs would not also be evaluated for purposes of containment design. (Table 3-1, page 18) [2]

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

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

ARP MST-67

RAI MST-67: Discuss how the statement from the July 30, 1993, Staff Requirements Memorandum (SRM)," that allows the use of scenario specific source terms, provided there is sufficient understanding and assurance of plant and fuel performance and deterministic engineering judgment is used to bound uncertainties". will be interpreted and applied in the NGNP deterministic safety analyses (i.e., dose consequence analyses) to mean that a bounding approach - applying deterministic engineering judgment - (rather than a best estimate approach) will be used to calculate the scenario specific mechanistic source term for AOOs and BDBEs, as well as DBAs.? (page 19) [2]

ARP MST-68

RAI MST-68: The white paper does not discuss the proposed licensing basis for calculating the scenario-specific mechanistic source terms for NGNP LBEs in the AOO frequency range, the DBE frequency range, or the BDBE frequency range. Nor does it discuss the proposed basis for calculating scenario-specific source terms for DBAs or emergency planning licensing basis events. For each of these event categories, discuss whether a best-estimate approach or a conservative approach will be used for the calculating the licensing basis mechanistic source terms. Where a best estimate approach is to be used, discuss the technical approach for the calculation. Where a conservative approach is to be used, discuss the technical approach for the calculation. The requested discussions should address whether a best-estimate or conservative approach will be used for predicting the long-term accumulation in the primary circuit of fission products released from the fuel during normal operation as needed for predicting the initial mechanistic source terms (i.e., radionuclide releases for each of the above event categories). (page 20) {MST-17, FQ-27 (MST-32), FQ-28 (MST-33), FQ-29 (MST-34), FQ-30 (MST-35, MST-68, MST-88} [2, 3]

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

ARP MST-69

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

ARP MST-70

RAI MST-70: Describe any plans for peer review of the NGNP mechanistic source terms by appropriately qualified subject matter experts. (page 21) [2, 3]

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

ARP MST-71

RAI MST-71: For the NGNP, discuss whether the major accident selected for the NGNP siting assessment will involve an examination of events from the BDBE region as well as events in the DBE region (i.e., DBAs). (page 16) [2]

ARP MST-72

RAI MST-72: What process is/was used to identify the underlying phenomena that are considered important to fission product transport and release (e.g., diffusion through intact particle coatings) in the mechanistic source term calculation for the NGNP plant design. (page 25) [2, 3]

ARP MST-73

RAI MST-73: The values shown in Table 4-3 for heavy metal contamination are very similar for pebble fuel and block fuel. However, the SiC defect fraction for block fuel is shown to be about an order of magnitude higher than heavy metal contamination. This would suggest that the PBR fuel particles are manufactured to a higher quality standard than the block fuel. If true, discuss why this is acceptable. If not true, discuss why the pebble fuel does not require a separate SiC defect fraction specification, especially with regard to accurately predicting alkali metal and alkaline earth releases from the fuel vs. fission gas release from the fuel. (page 16) [2(a), 3]

ARP MST-74

RAI MST-74: Discuss whether a natural circulation driving force could develop during the core heat-up phase if air infiltrates the primary system due to a break in the helium pressure boundary. (page 32) [2, 3]

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

ARP MST-75

RAI MST-75: For events involving breaks in the helium pressure boundary where core cooling is maintained by active cooling systems rather than passive cooling, discuss why significant circulation through the core is not expected during core heat-up. (page 32) [2, 3]

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

ARP MST-76

RAI MST-76: Discuss the extent to which fission product transport delay and holdup mechanisms in the matrix and fuel element graphitic materials are modeled for accident heat-up conditions. (page 32) [2(a), 3]

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

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

ARP MST-77

RAI MST-77: Discuss the relevance of "pressure vessel" particle coating failure probabilities to the NGNP UCO fuel design. Discuss the source of data for developing the models for the different NGNP UCO fuel particle failure mechanism predictions. (page 35) [2(a), 3]

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

ARP MST-78

RAI MST-78: Since the German UO_2 test data used for developing TRISO coated particle diffusion rates are largely based on post-irradiation heating tests as will be the data used to develop the diffusion rates for AGR UCO fuel, discuss why additional irradiation testing for UO_2 particle fuel and UCO particle fuel are not considered mechanistic source term knowledge gaps. (page 43) [2(a)]

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

ARP MST-79

<u>RAI MST-79</u>: Figure 4-1 identifies the Cleanup in the Helium Purification System and the Reactor Building HVAC System as important decontamination elements in the mechanistic source term calculations. However, Table 5-1 does not include these decontamination elements as source term knowledge gaps. Discuss the sources and the applicability of the data that is to be used to model the performance of these systems in the mechanistic source term analysis. (pages 29 and 44) [2, 3]

ARP MST-80

<u>RAI MST-80</u>: Discuss the plans, if any, to conduct phenomena identification and ranking tabulations (PIRTs) for the water-ingress and dust elements of the NGNP mechanistic source term calculation. If PIRTs are not planned, describe how the phenomena important to these conditions will be identified and assessed. (pages 29 and 44) [2, 3]

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

ARP MST-81

<u>RAI MST-81</u>: Discuss the plans, if any, to conduct fuel testing to model the effects of potential air ingress on the NGNP mechanistic source term calculation. (pages 22, 44 and 46) {FQ-24 (MST-38), MST-81} [2, 3]

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

ARP MST-82

<u>RAI MST-82</u>: Discuss whether and/or how the mechanistic source term definition (and associated phenomena) would change should the Commission decide that: (1) a traditional low leakage containment building will be required or, (2) a reactor building structure with an as yet to be defined fission product retention functional capability will be required (e.g., high efficiency filters in the vent path). (Table 3-1, pages 31 and 32) [1, 2, 3]

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

ARP MST-83

RAI MST-83: Discuss the approach that will be taken to ensure that NGNP "severe accidents" and the resulting severe accident source terms are bounded. (page 18) [2, 3]

Comments: The NGNP licensing strategy states: "Given the current state of VHTR technology design, development and experience, and the quality and completeness of the associated NGNP design-specific PRA, Option 2 is the preferred option for licensing the NGNP prototype, which makes primary use of deterministic judgment and analysis complemented by NGNP-specific PRA to establish the licensing basis and requirements." The white paper does not explicitly describe how deterministic judgment complemented by the NGNP PRA will be used to ensure that the events considered result in a set of source terms that bound the NGNP "severe accidents."

ARP MST-84

RAI MST-84: Please provide a clear and concise definition of event-specific mechanistic HTGR source terms. (pages 1 to 3) {MST-4} [1]

Comments: The white paper solicits NRC agreement that the definition of event-specific mechanistic source terms for HTGRs is acceptable. A concise definition of "event-specific mechanistic source terms" is not provided. Source term is defined in general (footnote (b) on page 1) as release of radionuclides from the *reactor building to the environment* – a departure from the traditional LWR source term. Event specificity is implied with regard to calculating the source terms for selected licensing basis events (LBEs). Mechanistic is interpreted in the earlier section of the text (page 2) as an approach that takes into account the multi-barrier DiD concept and distinctly different from the LWR "deterministic" approach. It is not clear if the implication is that the mechanistic approach is not deterministic which, of course, is not true. Later in the text (page 3 and also page 11, section 2.3.1), a better discussion of mechanistic approach is provided in terms of different fission product release and transport phenomena which are modeled presumably in a mechanistic manner supported by data. Also, elsewhere in the text (page 21, section 3.3.2), some elaboration is provided on what is meant by "deterministic source term." Perhaps, a better characterization is "prescriptive source term" as in TID-14484 and NUREG-1465.

The requested definition of event-specific mechanistic source terms should clearly state whether credit is taken for radionuclide retention in the reactor building and whether the MSTs are to be calculated using deterministic phenomenological models, or applying other (e.g., probabilistic) models. The supporting discussion should clearly state why the source terms should be linked to "release outside the reactor building."

ARP MST-85

RAI MST-85: Indicate where the HTGR supplier-proposed design conditions specified and clarify how they are used in relation to the NGNP design conditions in determining the mechanistic source terms? (page 6) [1, 2]

Comments: In Section 2.1, the third paragraph makes reference to HTGR supplier-proposed design conditions. It is not clear whether these design conditions are less demanding (e.g., with regard to reactor outlet temperature, operating conditions, etc.) than the NGNP design conditions.

ARP MST-86

<u>RAI MST-86</u>: What is considered core damage for NGNP? For example, is it kernel migration, breach of iPyC and/or oPyC layers, structural failure of SiC layer, failure of matrix retention capability, or some other definition? (page 11) [1, 2]

<u>Comments</u>: The white paper indicates on page 11 that there will be no postulated condition of the plant that results in significant fuel particle degradation or any other significant core damage. The use of the term "core damage" is noted. For HTGRs, the term begs a definition. This discussion should clarify that even without "core damage" radionuclides released from HTGR fuel during normal operation will accumulate in the primary system over the reactor's operating lifetime and contribute significantly to release source terms in accidents.

ARP MST-87

<u>RAI MST-87</u>: Please explain Figure 2-5 more clearly in terms of examples of how it would relate to a proposed mechanistic source term calculation. (page 15) [1, 2]

ARP MST-88

<u>RAI MST-88</u>: How will the uncertainties be treated in the NGNP approach for mechanistic source term calculation? (page 25) {MST-17, FQ-27 (MST-32), FQ-28 (MST-33), FQ-29 (MST-34), FQ-30 (MST-35), MST-60, MST-88} [1, 2]

<u>Comments</u>: The NGNP approach for mechanistic source terms relies on calculation of source terms and dose rates based on the current understanding of generation and transport phenomena. While the approach, in theory, is reasonable, it is not evident how uncertainties will be addressed.

ARP MST-89

<u>RAI MST-89</u>: Should the onset of prompt fatality be a consideration in the construct of frequency and consequence criteria for licensing basis events? If prompt fatalities were to be included, how would the NGNP approach for mechanistic source terms be affected? What is the basis for meeting the protective action guide (PAG) at a 425-meter exclusion area boundary (EAB)? (pages 25 and 26) [1, 2]

<u>Comments</u>: White paper Section 4.2 discusses NGNP top-level radionuclide control requirements. It is noted the top level requirements do not include onset of prompt fatality (300 – 500 rem dose). Also, it is noted that the requirements are discussed in relation to meeting the PAG at a 425-meter EAB.

ARP MST-90

RAI MST-90: The white paper states that Table 4-3 is based on several earlier analyses. Please provide references to these earlier analyses. (page 30) [2(a)]

ARP MST-91

RAI MST-91: SORS is an accident analysis code in the line of the NRC code MELCOR and TAC2D is a two-dimensional heat transport code. Clarify how the accident analysis will be done and how the NGNP mechanistic source term be calculated from these codes and methods. (page 35) [2]

ARP MST-92

RAI MST-92: It appears that GAUGE, BURP, and few other codes listed in Section 4.5.1 are legacy codes from the 1960s for which the validation data base must be very limited. Does the NGNP project have a plan for verification and validation of these legacy codes? If so, what is the schedule for such activities? What data will be required and generated for verification and validation? If verification and validation is not planned, provide a justification. (page 35) [2]

ARP MST-93

RAI MST-93: In Section 4.6.1, it is stated that significant reactor surveillance data will become available from the Japanese HTTR. Please describe (a) the status of any agreements under which such data will be provided, and (b) details on the surveillance data that will be acquired from the HTTR and when it would be available. (page 37) {MST-97} [2, 3]

ARP MST-94

RAI MST-94: Fort Saint Vrain (FSV) is noted as another source of data. FSV used lower-quality TRISO-coated fissile/fertile particles with kernels of HEU-carbide/thorium-carbide as opposed to NGNP's higher-quality TRISO-coated fuel particles with kernels of LEU-oxide and LEU-oxycarbide as described in the FQ white paper. In view of the differences associated with these and any other key attributes of FSV and NGNP fuel, please characterize the applicability of FSV data on fuel behavior and fission product transport to the validation/assessment of codes for predicting NGNP fuel behavior and fission product transport. (page 37) [2, 3]

ARP MST-95

RAI MST-95: Air ingress effects are not specifically mentioned in Table 5-1 under "Gap Description" in the fission product release category. Are air ingress effects going to be addressed related to the calculation of mechanistic source terms? (page 43, Table 5-1) [2, 3]

ARP MST-96

<u>RAI MST-96</u>: Describe in detail the experimental data used to evaluate effective diffusion coefficients under NGNP operating and accident conditions of temperature, neutron flux, and neutron fluence. (page 43, Table 5-1) [2, 3]

<u>Comments</u>: Table 5-1 states the following: “The suitability of using an "effective" diffusion coefficient in a model based on Fick's Law of diffusion needs to be demonstrated to ensure accurate calculations of fission product release to the primary circuit for use in shielding analyses and source term calculations.”

ARP MST-97

<u>RAI MST-97</u>: Table 5-1, it is stated that significant reactor surveillance data will become available from the Japanese HTTR. Please describe (a) the status of any agreements under which such data will be provided, and (b) details on the surveillance data that will be acquired from the HTTR and when it would be available. (pages 37 and 43, Table 5-1) {MST-93} [2, 3]

ARP MST-98

<u>RAI MST-98</u>: The white paper mentions that calculations are underway to determine the impact of dust on the behavior of fission products in the system. Please describe the calculations and the details of how they address the impacts of dust on release source terms. (page 48) [2, 3]

<u>Comments</u>: Important factors to be addressed by such analytical studies would include:

a) how the dust is produced and in what quantities, chemical compositions, and grain sizes, as suggested by related comments under RAI MST-14 and MST-16,

b) the effects of various postulated break sizes, including large (e.g., cross-duct) breaks, on how much dust exits the primary boundary,

c) the effects of variable peak fuel operating temperatures, including temperatures in postulated core hot spots, and fission product diffusivity uncertainties on the dust activity acquired from metallic fission products (e.g., cesium) from exposed fuel kernels and intact CFPs, and

d) the effects of water-ingress on dust activity and release.

ARP MST-99

<u>RAI MST-99</u>: When will the NGNP project provide its recommendations on the extent of additional testing to validate fission product liftoff and plateout models? Would there be any merit to soliciting external input (e.g., from NRC input under the existing DOE/NRC interagency agreement on NGNP) during the formulation of recommendations? (page 49) [2, 3]

<u>Comments</u>: The white paper states that the extent of additional testing needed for fission product liftoff and plateout models is still under discussion within the NGNP project team.

ARP MST-100

RAI MST-100: Please provide technical basis for the white paper's stated conclusion that PHEBUS and DEMONA are generally not applicable to HTGR fission product transport in reactor building. (page 49) [2, 3]

ARP MST-101

RAI MST-101: Please provide clarifying information on the NGNP approach for calculating a event-specific mechanistic source term in the following specific areas:

1. Treatment of uncertainties in source term calculations;
2. Efficacy of the legacy codes for source terms calculations; and
3. Validation/assessment data bases for codes.

(pages 50-51) [2, 3]

Comments: This summary-level RAI seeks integration of the responses to related RAIs above.

ARP MST-102

RAI MST-102: The approach for planned fission product tests focuses primarily on release and transport phenomena within fuel (i.e., coated particles, pebbles, and compacts). Provide more information on tests for other transport phenomena, including within the helium loop, reactor building, etc., to judge the adequacy and completeness of the proposed approach. (pages 50-51) [2, 3]

ARP MST-103

RAI MST-103: How much dust is generated in the reactor during operation? How is it generated, and what is it's size distribution? (Section 2.3.3) [2b]
Comments: This and following questions about dust are mostly concerned about one thing: although potential dust issues are identified pretty thoroughly in the paper, what is not addressed are quantitative questions like "How much?" and "What is the impact?" etc. Also not addressed is how INL plans to answer those questions. I get the impression that INL is going to rely on historical operation data and HTTR data, when it becomes available. I see no experiments planned to get further data. Regarding dust generation, the impression I got during the Dust Workshop was that most dust is formed by graphite-metal abrasion, i.e., the fuel handling system in the HTR-10 and AVR, or the circulator pistion rings in the HTTR. In the PMR design, I get the impression that the designers think that there will be little dust, so the discussion about it is moot. I heard a lot about radiation-induced spalling – I have no idea how they would even measure that. Another possibility is from refueling while handling the graphite blocks. Presumably any dust from this mechanism would be between blocks and not available to the coolant channels.

ARP MST-104

RAI MST-104: Does this dust deposit in the primary or core, and where?(Section 4.4.1)

ARP MST-105

RAI MST-105: How much fission products accumulate on the dust via adsorption or plateout, versus deposition on metal surfaces? (Section 4.4.1)

ARP MST-106

RAI MST-106: How much dust is available for liftoff in the event of an accident such as a DLOFC? Is all or part of the deposited dust immobilized by forming a crust? How much is in so-called dead zones? (Section 4.4.2, C-6.2)

Comments: There has been a lot of discussion attributing the formation of "crusts" to the introduction of organics, i.e. oil, into the primary. There is also mention about dust accumulating in dead zones. In general, dead zones will likely remain "dead" during an accident, so there will be no re-entrainment from these areas. So I think it is useful to identify these dead zones, and how much dust might be in them.

ARP MST-107

RAI MST-107: What is the effect of depressurization during an accident, for instance a DLOFC, on iodine adsorbed on steel and graphite in respect to desorption? (C-6.2)

Comments: In C-6.2, there is discussion of re-entrainment of condensable fission products, I'm assuming including iodine. This appears to concentrate on metal surfaces in the primary. I see no discussion in the paper of the effect of depressurization on iodine adsorbed on steel and graphite, namely, it will desorb.

ARP MST-108

RAI MST-108: Please clarify the statement about ">95% retention of Kr and Iodine by the kernel". (C-3)

Comments: On p.69, para.2, it talks about >95% retention of Kr and Iodine by the kernel under normal conditions. Hossein Esmaili has calculated about 15% release for Kr-85 at 1200 K under "normal conditions", and I got similar results (17% at 1250 K). This doesn't seem very "retentive", and is not ">95%". Possibly they are referring only to Kr-88 and I-131, which have short halflives, but the writing gives the impression that 95% applies to all Iodine, Xe, and Kr isotopes.
p.72, C-5, describes matrix and graphite as barriers. Diffusion rates in TECDOC-978 for Kr and Iodine are so high that they indicate that the resistance of matrix/graphite is ignored for these gases, so is a barrier only for metals. This is actually stated in Section 4.4.3, bottom of p.33. A further comment: both C-4 and C-5 have titles referring to "Radionuclide Transport in …" whereas they only refer to metals.
In C-6, reference is made to "condensable fission products" as plating out on the primary surfaces. There is some danger of confusion here, as plateout can be driven by several mechanisms, including condensation, aerosol deposition, and adsorption.

ARP MST-109

RAI MST-109: There is currently no database to validate the use of the usual LWR systems codes for radionuclide transport in the reactor building. How is INL going to address this? (Section 5.5.2, C-7)

Comments: Regarding the reactor building, there is discussion (C-7) that the mechanisms for calculating FP transport are the same as for LWRs (true as far as I know), but there is no database to support the calculations for HTGRs, so this is a problem. The source term is now being defined as “release from the reactor building”, so this is a critical issue.

ARP MST-110

RAI MST-110: How many experiments are envisioned to satisfy the objective to fill in the gaps in the fission product transport models -- for all nuclides, all base materials, all temperatures and effect of neutron fluence? For the total amount of experiments totaled up, how well estimated will the error bands be? (p. 25, #8)

ARP MST-111

RAI MST-111: How have the nuclides been selected for transport analysis? Is it assured that the selected nuclides adequately cover the various dose estimate requirements? Should C-14 and tritium be included in your “fission product” listing as special cases? Why are Cs134 and Cs136 omitted on p. 38? (Table 4-1)

ARP MST-112

RAI MST-112: Why is the direct cycle turbine not listed along with the steam generator as a possible special source? (p.27, paragraph 1)

ARP MST-113

RAI MST-113: How do you define “barrier”? How many of the “barriers” listed are truly barriers? (p. 27, paragraph 5)

ARP MST-114

RAI MST-114: Is “plateout” used to signify aerosol deposition, both dust and molecular deposition e.g., sorption on surfaces? Are there separate model for each mechanism? Similarly for liftoff, does that refer only aerodynamic removal of aerosols, or also for desorption? Are there separate models for aerosol deposition and for chemisorptive (or condensation) deposition? (Fig. 4-1)

ARP MST-115

RAI MST-115: Clarify the meanings of “fuel quality”, “HM contamination”, and “defective particles”. Are these terms used consistently? Are HM contamination and fuel particles failures actually reported separately in PIEs, or is this a future intention? (p. 30, Fig. 4-1 and Table 4-3)

ARP MST-116

RAI MST-116: Since dust removal is cited as a release mechanism, fission product association with dust must be specified. Where in the White Paper is radioactivity association with dust considered? (p.31, #4)

ARP MST-117

RAI MST-117: What are the inputs, principal modeling assumptions, and outputs of each code? Particularly, what is the role of each program? Why does there appear to be duplication? (Fig 4-3 and Appendix D)

ARP MST-118

RAI MST-118: Why are Xe135m and Kr88 releases rising with FIMA? What is it in NOBLEG that predicts this? (p. 40, Figs 4-7, 4-8)