OFFICE OF NEW REACTORS

ASSESSMENT OF WHITE PAPER SUBMITTALS
ON FUEL QUALIFICATION AND MECHANISTIC SOURCE TERMS

NEXT GENERATION NUCLEAR PLANT

PROJECT 0748

1. INTRODUCTION

The Next Generation Nuclear Plant (NGNP) Project was established by the U.S. Department of Energy (DOE) as required by Congress in Title VI, Subtitle C, of the Energy Policy Act of 2005 (EPAct). The mission of the NGNP Project (i.e., the Project) is to develop, license, build, and operate a prototype high temperature gas cooled reactor (HTGR) plant that generates high temperature process heat for use in hydrogen production and other energy-intensive industries while also generating electric power. To fulfill this mission, the Project is considering a modular HTGR with either a prismatic block or pebble bed core and safety features described as follows:[1]

> *"To achieve the safety objectives for the NGNP Project, the HTGR relies on inherent and passive safety features. Modular HTGRs use the inherent high temperature characteristics of TRISO-coated fuel particles, graphite moderator, and helium coolant, along with passive heat removal capability of a low-power-density core with a relatively large height-to-diameter ratio within an uninsulated steel reactor vessel to assure sufficient core residual heat removal under loss-of-forced cooling or loss-of-coolant-pressure conditions.*
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> *The primary radionuclide retention barrier in the HTGR consists of the three ceramic coating layers surrounding the fissionable kernel to form a fuel particle. As shown in Figure 4, these coating layers include the inner pyrocarbon (IPyC), silicon carbide (SiC), and outer pyrocarbon (OPyC), which together with the buffer layer constitute the TRISO coating. The coating system constitutes a miniature pressure vessel that has been engineered to provide containment of the radionuclides and gases generated by fission of the nuclear material in the kernel. Thousands of these TRISO-coated particles are bonded in a carbonaceous material into either a cylindrical fuel compact for the prismatic HTGR or a spherical fuel element for the pebble bed HTGR. These fuel particles can withstand extremely high temperature without losing their ability to retain radionuclides under all accident conditions.*

[1] INL/EXT-11-22708, "Modular HTGR Safety Basis and Approach," NGNP information paper submitted September 6, 2011, Project 0748, ADAMS accession number ML11251A169, excerpt page 8.

Fuel temperatures can remain at 1600 ºC for several hundred hours without loss of particle coating integrity [INL 2010a]. This high temperature radionuclide retention capability is the key element in the design and licensing of HTGRs."

As stipulated by the EPAct, the Project and the Nuclear Regulatory Commission (NRC) have been engaged in pre-licensing interactions on technical and policy issues that could affect design and licensing of the NGNP prototype. Such early interactions are encouraged by the Commission Policy Statement on the Regulation of the Advanced Nuclear Power Plants, which states in part the following:

"During the initial phase of advanced reactor development, the Commission particularly encourages design innovations that enhance safety, reliability, and security… and that generally depend on technology that is either proven or can be demonstrated by a straightforward technology development program. In the absence of a significant history of operating experience on an advanced concept reactor, plans for the innovative use of proven technology and/or new technology development programs should be presented to the NRC for review as early as possible, so that the NRC can assess how the proposed program might influence regulatory requirements."

DOE's contractor, Idaho National Laboratory (INL), is conducting research and development in support of the Project and has prepared a series of white papers on aspects of the HTGR design and safety basis in order to obtain NRC feedback on design, safety, technical, and/or licensing process issues that could affect NGNP deployment.

On July 21, 2010, the Project submitted the two interrelated white papers that are the subject of this assessment document, namely:

INL/EXT-10-17686, "NGNP Fuel Qualification White Paper" (ADAMS accession number ML102040261, referred to herein as the FQ white paper)

INL/EXT-10-17997, "NGNP Mechanistic Source Terms White Paper" (ADAMS accession number ML102040260, referred to herein as the MST white paper)

The cover letter for these submittals states, in brief, that these white papers summarize the planned approaches to fuel qualification and the development of mechanistic source terms. The approaches are stated to apply generically, to the extent possible, to both the pebble bed and prismatic block HTGR design options being considered for the NGNP prototype.

The stated primary purpose of the white papers is to obtain NRC feedback on the acceptability of the planned high-level approaches as the basis for developing related aspects of the NGNP safety analysis report to be submitted for licensing the NGNP prototype. The purpose of this assessment document is to provide the requested NRC feedback.

2. ASSESSMENT PROCESS

To develop the feedback requested by the Project, the NRC assembled an assessment working group composed of personnel from the NRC offices of New Reactors and Nuclear Regulatory Research. The working group was assisted by personnel from Brookhaven National Laboratory and Sandia National Laboratories. Appendix A of this document lists the NRC working group participants and national laboratory participants.

The NRC working group started the assessment process by discussing the white papers with Project personnel at a public meeting held September 1-2, 2010 (ADAMS accession numbers ML102590247, ML102700497). Routine bi-weekly conference calls between NRC and the Project served to facilitate continuing coordination of all interactions related to NGNP, including those for assessing the subject white papers. By letter dated May 3, 2011 (ADAMS accession number ML111250375), the Project reported that its plans pertaining to pebble fuel qualification (i.e., fuel qualification for the pebble bed HTGR design option), and to related aspects of mechanistic source terms development, were changing and were no longer based on those described in the respective white papers. In accordance with the Project's request, the working group did not assess those portions of the white paper that relate to the Project's changing plans for pebble fuel, as further noted in Section 3 and Appendix B of this document.

Over the course of the assessment process, the working group issued two sets of requests for additional information (RAIs) related to the content and stated feedback objectives of the subject white papers. The first set, consisting of 118 RAI questions, was issued on June 7, 2011 (ADAMS accession number ML111530271) and the second set, consisting of 82 RAI questions, on July 25, 2011 (ADAMS accession number ML112030135). The Project responded to the two sets of RAIs on August 10, 2011 (ADAMS accession number ML11224A060) and September 21, 2011 (ADAMS accession number ML11266A133), respectively.

On October 19, 2011, the NRC working group held a public meeting with Project personnel to clarify and discuss selected RAI responses deemed particularly important toward completing the NRC assessment feedback. At that meeting, the NRC working group commented that the word "acceptable" as used in the Project's stated outcome objectives carries regulatory/legal connotations that would not be appropriate for the white paper assessments. Therefore, in completing the assessments, the NRC working group has instead assessed the proposed approaches in terms of whether they are reasonable, thereby effectively replacing "acceptable" with "reasonable" in the Project's feedback requests.

3. ASSESSMENT RESULTS

This section of the assessment document presents the feedback developed by the NRC working group in assessing the Project's white papers on NGNP fuel qualification and mechanistic source terms. The feedback thus reflects the views of the NRC working group members. Section 3.1 presents a high-level overview of the proposed approaches and assessment and introduces a framework for presenting more detailed assessment results in Sections 3.2-3.13.

This assessment does not provide a final regulatory decision on any aspect of the NGNP technical licensing approach or NGNP design. Completion of the NGNP prototype design and safety basis in accordance with the assessment feedback provided herein will not be sufficient justification for the design. Such conclusions will be provided in the NRC staff's safety evaluation of a future combined license or design certification submittal. The safety evaluation will determine whether or not the proposed NGNP design complies with applicable NRC regulations, consistent with NRC guidance for reviewing such license applications and relevant technical policy guidance provided by the Commission.

The assessment feedback on these white papers is preliminary, since many issues identified by the working group cannot be addressed or resolved until more information about the NGNP design and the results of planned or needed supporting safety research and development are available. Even so, the working group believes identifying these issues is beneficial to the Project, so that relevant insights can be considered in further developing the NGNP design and its safety basis.

The specific technical issues for which the Project has requested NRC assessment feedback are presented in Section 6 of the respective white papers in terms of stated outcome objectives. As noted above, on May 3, 2011, the Project issued a letter retracting those stated outcome objectives that relate to the Project's changing plans for pebble fuel (ADAMS accession number ML111250375). For convenience, and to facilitate referencing in the feedback discussions that follow, Appendix B of this document consolidates and enumerates the Project's updated outcome objectives for fuel qualification and mechanistic source terms.

Throughout this section, codes shown in parentheses after subsection topical headings map that assessment feedback to the Project's stated outcome objectives as labeled in Appendix B.[2] As reflected in Appendix B, and consistent with the Project's letter of May 3, 2011, the assessment feedback provided in this document addresses neither objective FQ1 for pebble bed reactor fuel qualification nor the affected pebble-fuel-specific aspects of objectives ST2 and ST3 for mechanistic source terms. The working group nevertheless included in its two sets of RAIs all previously developed RAI questions specific to pebble bed reactor fuel. This was done in view of the full or partial relevance those questions may be found to hold to the Project's changing plans for pebble bed reactor fuel. Accordingly, the Project's RAI response submittals simply acknowledged such pebble-specific RAI questions with a brief reply citing the Project's May 3, 2011, statement that such information in the white papers should be withheld from further review.

Additional details and background information on each feedback topic and subtopic are generally available in related RAIs in the form of individual RAI questions and comments developed by the working group and associated responses provided by the Project. Each topic is thus footnoted with a list of related RAIs.[3] Some RAIs are listed under one or multiple

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[2] Assessment feedback comments concerning the outcome objectives listed in Appendix B for source term calculation (ST2) and validation (ST3), respectively, are presented for convenience under the same topical headings. For example, the objective codes "(FQ2) (ST2/3(a))" indicate that the feedback presented under that heading addresses objective FQ2 for prismatic fuel qualification and objectives ST2 and ST3(a) for the respective calculation and validation of radionuclide retention and transport within the TRISO particle fuel kernels and coating layers and surrounding materials of the prismatic fuel element.

[3] Notes on RAI numbers and references:

(1) Because fuel qualification can be viewed as largely a subtopic of mechanistic source terms, many RAI questions on the FQ white paper were repeated verbatim as RAI questions pertaining to the MST white paper. This approach sought to ensure due consideration of fuel qualification RAI questions in terms of the interrelated approaches and objectives of both white papers.

feedback subtopics and some not at all. RAIs not listed under any subtopics either asked for simple clarifications adequately provided by the RAI responses or concerned only the pebble-fuel-specific information that the Project had withdrawn from further review. Regarding the former, the working group generally understands that the Project intends to incorporate appropriate clarifications in any future versions of the white papers and in any subsequent NGNP submittals related to HTGR fuel qualification and mechanistic source terms.

3.1 Assessment Overview (FQ2) (ST1, ST2/3(a)-(d))

The NRC working group's overall assessment is that the proposed high-level approaches to NGNP fuel qualification and mechanistic source terms are generally reasonable, albeit with several potentially significant caveats. This means that, subject to further consideration and resolution of details and issues noted subsequently in this assessment document, the working group's review of these white papers has found no fundamental shortcomings that would necessarily preclude successful implementation of the presented high-level approaches towards establishing the technical bases for related NGNP prototype licensing submittals. In addressing the Project's requests for feedback, the assessment comments provided herein are intended to facilitate continuing efforts by the Project and NRC towards achieving effective resolution of technical and policy issues for HTGR licensing and regulation.

The following subsections present an overview of the assessment feedback in two parts. First, Subsection 3.1.1 provides a brief overview of the basic approaches presented in the FQ and MST white papers. Subsection 3.1.2 then broadly assesses the overall scope and structure of the presented approaches and thereby establishes a topical framework for presenting the working group's detailed assessment feedback in Sections 3.2-3.13.

3.1.1 Overview of the proposed approaches to fuel qualification and mechanistic source terms

Section 2.3.1 of the MST white paper states the following:

> *"The safety basis of the HTGR precludes core damage that could significantly affect radiological consequences and, therefore, focuses on preventing and limiting the release of relatively small amounts of radioactive material as a result of event sequences that could occur with this design. The calculation of source terms for these conditions is event-specific and requires validating the characteristics and integrity of barriers to the transport and release of radionuclides from the plant for each event."*

The proposed technical approach to establishing and validating the characteristics and integrity of the primary release barrier, the tristructural-isotropic (TRISO) fuel particle, relies extensively on results from the Project's ongoing NGNP Advanced Gas Reactor (AGR) Fuel Development

(2) RAI questions on fuel qualification and mechanistic source terms were numbered with the respective prefixes FQ and MST. For example, referring to the numbers in bold font in the respective RAI documents, the first RAI questions were numbered FQ-1 and MST-1, respectively, within the first set of RAIs, and FQ/MST-B1 within the second set of RAIs.

(3) This assessment document refers to RAIs in abbreviated form. For example, "F1" refers to RAI FQ-1, "M1" to RAI MST-1, and "B1" to RAI FQ-B1/MST-B1.

and Qualification Program[4] (henceforth called the NGNP/AGR Fuel Program). Building on decades of international experience with HTGR TRISO fuel development and testing, the scope of the NGNP/AGR Fuel Program encompasses development of the fuel design, fabrication processes, and fuel quality assurance measures as well as irradiation and safety testing of fabricated fuel samples.

Irradiation testing is performed in the Advanced Test Reactor (ATR), a water-cooled materials test reactor (MTR) located at INL. A series of irradiation and post-irradiation safety tests, designated AGR-1 through AGR-8, provides the proposed basis for fuel development and qualification by testing the integrity and performance of fabricated fuel under service conditions intended to envelope those to be encountered during NGNP normal operations and licensing basis events (LBEs). The AGR test series progresses from initial shakedown tests on fuel fabricated with developmental lab-scale equipment and controls to qualification tests on fuel fabricated with production-scale equipment, procedures, and quality controls. The formal fuel qualification tests are designated as AGR-5/6.

Also included in the planned AGR test series are special tests (i.e., AGR-3/4) involving designed-to-fail fuel (i.e., coated fuel particles with no buffer layer and a thin pyrocarbon (PyC) layer) for use in developing data needed to model radionuclide retention and transport in TRISO fuel particle kernels and the carbonaceous/graphitic fuel elements in which the TRISO fuel particles are embedded. At the time of this assessment, the AGR-1 and AGR-2 irradiation tests of preliminary fuel designs fabricated with developmental equipment, processes, and controls had been completed, and the AGR-1 post-irradiation safety tests were in progress.

Proposed approach to fuel development and qualification

The Project's technical approach to fuel development and qualification builds upon an extensive national and international experience base with HTGR TRISO coated fuel particle technology that has accrued over several decades. Included in the international experience base are developments in the design, analysis, manufacture, irradiation testing, post-irradiation examination (PIE), and post-irradiation safety testing and licensed in-reactor operation of TRISO coated particle fuels in HTGRs in Germany, Japan, and China.

The first successful demonstration of what many consider the reference standard for high performing uranium dioxide (UO_2) TRISO fuel was achieved in Germany in the 1980s. This was followed by similarly successful demonstrations reported in Japan and China. The Chinese used the same fuel fabrication equipment that had been used in Germany. In the early 1990s, DOE sponsored a fuel development program for the design, manufacture, and irradiation testing of high-enriched TRISO coated particle uranium oxycarbide (UCO) fuel for use in the New Production Reactor (NPR). The NPR was a proposed prismatic block modular HTGR designed for material production and electric power generation. However, the NPR TRISO coated particle fuel exhibited relatively poor irradiation test performance.

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[4] PLN-3636, "Technical Program Plan for the NGNP Advanced Gas Reactor Fuel Development and Qualification Program," Revision ID: 0, September 30, 2010.

A central strategy of the Project's NGNP/AGR Fuel Program has been to qualitatively and quantitatively analyze the international and national TRISO coated particle experience base to develop a more scientific understanding of the fuel fabrication processes and fuel properties that result in high performing fuels in-reactor. The Project has sought to reverse engineer the design of the fuel particle and the development of fuel fabrication equipment, fabrication processes and specifications, process controls, fuel product specifications and characterization techniques, and statistical analysis methods that will result in high performing fuel. The objective has been to manufacture fuel that consistently meets process and product specifications and satisfies NGNP fuel performance requirements for normal operations and accident conditions.

To test the in-reactor performance of manufactured TRISO fuels against requirements, the AGR test irradiations first monitor fuel performance during accelerated irradiation in the ATR by measuring fission gas releases. Irradiated fuel samples then undergo PIE and post-irradiation safety testing. Safety testing consists of heating the irradiated fuel samples (i.e., fuel compacts) to anticipated peak HTGR accident temperatures (e.g., 1600 °C) while measuring radionuclide releases to monitor and record any indications of individual particle failures. Unheated and heat-tested irradiated fuel samples undergo PIE.

An early indication of the effectiveness of the NGNP/AGR Fuel Program in implementing this strategy is reflected in the AGR-1 and AGR-2 TRISO coated fuel particle defect rate from fuel manufacture and the fuel particle failure rate during the AGR-1 and AGR-2 fuel irradiations. To date the TRISO coated fuel particle defect rates from fuel manufacture have been within the fuel particle design defect limits for manufacture and the fuel particle failure rates during the AGR-1 and AGR-2 fuel irradiations have been within the fuel particle design failure rate limits for the design normal operation service conditions projected for the NGNP design. However, at the time of this assessment, there was insufficient failure rate data from post-irradiation accident heating (i.e., safety) tests on the AGR-1 fuel to draw firm preliminary conclusions on the accident performance of the fuel being developed for the NGNP prototype.

The AGR-5/6 tests are the formal reference tests for NGNP fuel qualification. These are intended to demonstrate the irradiation performance of fuel fabricated to the established NGNP fuel manufacture specifications, using production-scale fuel fabrication equipment, processes, and quality assurance (QA) methods. The qualification test fuel will be irradiated at NGNP normal operating design conditions and then safety tested and examined post-irradiation in statistically sufficient quantities to demonstrate that the fuel performance during NGNP normal operating design conditions and NGNP accident conditions meets the established fuel performance requirements.

Proposed approach to developing NGNP event-specific mechanistic source terms

The intended principal barrier to radionuclide release for postulated accidents, including beyond design basis events (BDBEs), in modular HTGRs is the TRISO coated fuel particle. Beyond the TRISO fuel particles, three additional physical barriers to radionuclide transport and release are considered by the Project in its proposed approach to predicting the release of radionuclides to the environment during HTGR accidents. These additional barriers are the carbonaceous fuel elements in which the TRISO particles are embedded, the reactor system helium pressure boundary, and the reactor building. The white papers present a proposed approach to predicting event-specific release source terms based on the development, validation, and application of mechanistic models that calculate the transport of radionuclides across the four concentric barriers. A stated preliminary goal is to demonstrate with 95% confidence that predicted releases from the core are accurate to within a factor of four for fission gases and a factor of ten for fission metals.

Radionuclide transport in fuel particles and fuel elements

An international database of HTGR fuel-related radionuclide transport data was compiled in the 1990s and published in summary form in 1997 by the International Atomic Energy Agency (IAEA) in TECDOC-987. The Project plans to reference (i.e., use) these data in modeling the fission product transport in the NGNP fuel in connection with the mechanistic source term calculation. Additionally, data from the NGNP/AGR Fuel Program will be used to confirm (or modify as needed) the applicability of the reference TECDOC-978 data to the NGNP fuel and to establish data needed to model fission product transport data for the NGNP UCO fuel kernels and NGNP fuel matrix material.

For AGR-2 and AGR-7, the release of radionuclides under irradiation conditions will be measured via PIE and analyzed to derive effective diffusion coefficients under irradiation. The resulting diffusion coefficients derived from AGR-2 and AGR-7 test data will be reported and compared to the international database values documented in IAEA-TECDOC-978. AGR3/4 will be used to develop data needed to model fission product transport in the NGNP UCO fuel kernels and NGNP fuel matrix material.

The supplemental AGR test data is intended to confirm that these aspects of NGNP fuel radionuclide transport analysis can reference, or adapt as needed, the international data in TECDOC-978 for use in modeling fuel radionuclide retention and transport for the prediction of NGNP event-specific mechanistic source terms. The Project plans to conduct additional experiments to develop data that will be needed to model fission product transport in the fuel under chemical attack conditions due to air ingress and moisture ingress.

Radionuclide transport in the primary system and reactor building

Models for radionuclide transport in the primary circuit and reactor building include those for plateout and liftoff of radionuclides from surfaces in the primary circuit; generation, accumulation, and re-entrainment of carbonaceous dust contaminated with radionuclides; and distribution, condensation, plateout, and settling of radionuclides in the reactor cavity and the other interconnected volumes within the reactor building. Effects of moisture and air ingress on radionuclide transport, the role of helium purification system, and reactor building venting are other aspects of modeling radionuclide transport in the primary circuit and reactor building.

The white papers indicate that the NGNP/AGR Fuel Program plans to perform single effects tests in an out-of-pile helium loop to characterize radionuclide deposition on and re-entrainment from primary system surfaces (i.e., plateout and liftoff) under normal and off-normal HTGR conditions.

The Project conducted an assessment, which it characterized as a conceptual PIRT (phenomena identification and ranking tabulation), of the effects of moisture ingress on the HTGR performance in February 2011. The major phenomena and issues of high importance and requiring more attention, as noted by the Project, are:

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

The Project, in collaboration with NRC, also conducted an HTGR dust workshop in March 2011. A document that describes potential HTGR dust safety issues as well as research and development needs was prepared, based upon the discussions at the workshop.

3.1.2 Assessment of the overall scope and structure of the proposed technical approaches

The technical approaches presented in both white papers are based to a great extent on activities further described in the NGNP/AGR Fuel Program Plan. Based on that plan, Section 5.1 of the FQ white paper identifies the following five common elements of the proposed NGNP fuel qualification program:

- *Establishment of a fuel-product specification*
- *Implementation of a fuel-fabrication process capable of meeting the specification*
- *Implementation of statistical quality assurance procedures to demonstrate that the specification has been met*
- *Irradiation of statistically sufficient quantities of fuel with monitoring of in-pile performance and PIE to demonstrate that normal operation performance requirements are met*
- *Safety testing of statistically sufficient quantities of irradiated fuel to demonstrate that accident condition performance requirements are met.*

Both white papers note that, in demonstrating fuel performance capability, the NGNP/AGR Fuel Program also provides data for use in developing and validating predictive models of NGNP fuel performance and fuel radionuclide transport. The resulting predictive models play a prominent role in the source term analysis approach described in the MST white paper.

It is the working group's preliminary view that the elements identified by the Project are necessary but not sufficient as the bases for a comprehensive fuel qualification program and that additional elements should be added before and after the five elements listed by the Project. The following element should be identified as a necessary first step:

- Establishment of fuel design service conditions and performance requirements for normal operations and accidents

While Section 4 of the FQ white paper does address fuel service conditions and performance requirements, the working group's view is that additional service condition and performance parameters should be specified beyond those presented in the FQ white paper. Adequate specification of fuel service conditions and performance requirements should thus be highlighted as a key element of the fuel qualification program. The basis for this view is discussed in Section 3.2.[5]

The final set of fuel qualification irradiation and safety tests described in the FQ white paper is to be performed on fuel fabricated with production-scale equipment, but not explicitly on fuel fabricated on the production lines of the NGNP fuel fabrication facility. The working group's view is that a comprehensive fuel qualification program should further include "proof testing" of fuel fabricated on the NGNP fuel facility production lines to demonstrate irradiation and safety performance equivalent to that of fuel fabricated with developmental production-scale equipment, procedures, quality controls, etc. A comprehensive fuel qualification program should thus include the following penultimate element:[6]

- Irradiation and accident proof testing of NGNP fuel fabricated on the production lines of the NGNP fuel fabrication facility

The working group also believes that significant programs of pre-operational and operational testing, monitoring, inspection, and surveillance will likely be needed in the NGNP prototype to confirm safety-related design predictions and thereby verify and supplement the developmental technical bases for NGNP fuel qualification and mechanistic source terms. Such prototype testing, inspection, and surveillance programs may necessitate the incorporation of special prototype instrumentation and related design features in the NGNP prototype that, subject to successful confirmation of conformance, would not be required in subsequent NGNP plant designs. Given that the NGNP/AGR Fuel Program relies exclusively on accelerated fuel irradiation testing in a non-HTGR test reactor environment, it is the working group's view that the following element should be identified as a necessary final step for NGNP fuel qualification as well as mechanistic source term development:[7]

- Establishment and implementation of NGNP prototype pre-operational and operational programs to verify and supplement the developmental technical bases for fuel qualification and mechanistic source terms

The working group's preliminary assessment is that the five qualification program elements identified by the Project, if supplemented by the three additional program elements stated

[5] Related RAIs: F1/M1, F3/M3, F4/M5, F5/M6, M12, F13/M18, F14/M19, F15/M20, F16/M21, F22, F23/M28, F24/M29, F33, F34/M38, F41/M45, F48/M52, F49/M53, M73, M85, M86, M115, B6, B11, B13, B22, B27, B30, B31, B32, B33, B49, B66, B77, B78

[6] Related RAIs: F26/M31, B55

[7] Related RAIs: F1/M1, F3/M3, F4/M5, F5/M6, F7/M8, F10/M13, M12, F13/M18, F14/M19, F15/M20, F16/M21, F22, F23/M28, B5, B29, B47, B49, B76, B80

above, may constitute a reasonable structure for NGNP fuel qualification and related aspects of mechanistic source terms development. Finally, the working group observes that it may be both possible and desirable to conduct the irradiation proof testing step noted above in the NGNP prototype, thereby effectively combining the final two program elements into one.

The remaining sections present the working group's detailed assessment results under a logical sequence of topical headings that incorporate the structural elements noted above and apply them toward considering the interrelated contents and objectives of both white papers. The resulting topical headings are listed below, with the corresponding FQ and MST outcome objective codes shown in parentheses:

i. Establishment of NGNP fuel service conditions and performance requirements for normal operations and accidents (FQ2) (ST2/3(a))

ii. Establishment of NGNP fuel design with product specifications and process specifications for NGNP fuel fabrication (FQ2) (ST2/3(a))

iii. Establishment of a fuel fabrication process that will meet the NGNP fuel product specifications (FQ2) (ST2/3(a))

iv. Establishment and implementation of a fuel fabrication statistical quality control procedure that demonstrates the fuel product specifications are met (FQ2) (ST2/3(a))

v. Demonstration that fuel performance requirements for normal operations are met by irradiating a statistically significant quantity of fuel at NGNP fuel design conditions, monitoring fuel irradiation performance, and conducting post-irradiation examinations (FQ2) (ST2/3(a))

vi. Demonstration that fuel performance requirements for accident conditions are met by safety testing a statistically significant quantity of irradiated fuel at NGNP accident conditions and monitoring fuel accident performance (FQ2) (ST2/3(a))

vii. Irradiation and accident proof testing of NGNP fuel fabricated on the production lines of the NGNP fuel fabrication facility (FQ2) (ST2/3(a))

viii. Definition of event-specific mechanistic source terms for NGNP (ST1)

ix. Establishment and validation of models for fuel performance and radionuclide transport in fuel particles and fuel elements (FQ2) (ST2/3(a))

x. Establishment and validation of models for radionuclide transport in the primary circuit and reactor building (ST2/3(b)-(d))

xi. Application of mechanistic source term models in best estimate and conservative analyses of transients and accidents (ST2/3(a)-(d))

xii. Establishment and implementation of NGNP prototype pre-operational and operational programs to verify and supplement the developmental technical bases for fuel qualification and mechanistic source terms (FQ2) (ST2/3(a)-(d))

Assessment feedback under each of the above topical headings is presented as a series of observations on the respective topical area and its relevant subtopics. Many feedback topics or issues are identified as items for follow up in future interactions between the Project or HTGR applicant and the NRC. Certain feedback topics or issues are tagged as “major” follow-up items, meaning that the working group presently views these issues as warranting high priority for further discussion, assessment, and resolution.

3.2 Establishment of NGNP fuel service conditions and performance requirements for normal operations and accidents (FQ2) (ST2/3(a))

As noted in Section 3.1 above, the working group’s view is that the NGNP/AGR Fuel Program should more explicitly and more completely establish and document the NGNP fuel service conditions and performance requirements for normal operations and accidents. This entails, among other things, the effective interfacing of LBE selection and associated accident analysis predictions with fuel qualification and mechanistic source terms development. These and other topical interfaces are identified and briefly discussed in the respective NGNP white papers (e.g., Section 1.3.2 and Figure 1-1 in the MST white paper) and will merit continuing follow up, commensurate with their importance.

3.2.1 Fuel service conditions for NGNP normal operations

For the NGNP prismatic block core design, it is expected that the fuel design service conditions for normal operations will be significantly more demanding than those associated with past and current HTGR test reactors, such as the AVR, HTR-10, and HTTR, and past and current HTGR power reactor designs, such as Fort Saint Vrain, MHTGR, PBMR, and HTR-PM.

Currently, like the NGNP design itself, the fuel design service conditions for NGNP normal operations are not yet finalized and are therefore considered items for follow up. The normal operating fuel service conditions addressed in the NGNP/AGR Fuel Program’s normal operation irradiation tests are presently based on what the Project states to be a conservative assessment of the best available code predictions of fuel operating conditions in preliminary designs of an NGNP prismatic block core. When NGNP normal fuel service conditions are finalized, it will be necessary to show how well they are addressed by those tested in the NGNP/AGR Fuel Program.

The FQ white paper describes the targeted fuel design service conditions for NGNP normal operations in terms of maximum values of what the Project characterizes as the three dominant parameters of operating temperature, burnup, and fluence, namely:

- Maximum fuel particle operating temperature (1,400 °C)
- Maximum time-average fuel particle operating temperature (1,250 °C)
- Maximum fuel burnup (17 percent fissions per initial metal atom (FIMA))
- Maximum fuel particle fast neutron fluence (5×10^{25} n/m^2,E>0.18 MeV)

Additional fuel operating condition parameters[8]

[8] Related RAIs: F1/M1, F3/M3, F4/M5, F5/M6, M12, F13/M18, F14/M19, F15/M20, F16/M21, F22, F23/M28, F24/M29, F33, F34/M38, F41/M45, F48/M52, F49/M53, M73, M86, M115, B33, B49, B66

The working group's view is that the above set of normal operating service condition parameters should be supplemented with the following significant parameters:

- Maximum fuel plutonium burnup (i.e., burnup from fissions of bred plutonium)
- Maximum times at fuel particle operating temperatures (i.e., maximum time-at-temperature)

Plutonium burnup is viewed as significant because plutonium fission is the main source of important fission product elements (e.g., palladium, silver) that are either known (palladium) or hypothesized (silver) to potentially degrade TRISO fuel particle performance under operating and accident conditions. Multiplying this plutonium burnup parameter by fuel particle time-at-temperature yields an integral parameter addressing the potentially degrading effects of palladium and silver time-at-temperature on TRISO fuel performance. The working group's views on the importance of these additional fuel operating service condition parameters are further discussed in the context of fuel testing in Sections 3.6 and 3.7. This issue is an item for follow up.

Parameter path dependence[9]

In addition, the working group believes that additional information on how these fuel operating parameters vary with location and operating time in the NGNP core may be necessary for further evaluating questions of "path dependence." Such questions concern whether more varied combinations of fuel operating parameter values, such as maximum fluence with moderate burnup, moderate fluence with maximum burnup, low operating temperature with maximum fluence, etc., might be found to merit additional consideration in terms of how they could affect fuel operating and accident performance. These considerations would then be factored into further assessing the adequacy of the Project's proposed reliance on accelerated irradiations in the ATR to address limited combinations of high fuel operating temperature with high burnup and high fluence. This issue is an item for follow up.

Operating condition uncertainties and anomalies[10]

The working group notes the importance of considering HTGR fuel normal operating service conditions in terms of the apparent potential for large uncertainties and undetected anomalies involving such key parameters as maximum fuel normal operating temperature. It appears that such issues of HTGR core analysis and core monitoring can be addressed only in small part by analytical means and separate-effects validation testing. It is thus the working group's view that adequate resolution of these issues will likely necessitate verification of initial and evolving NGNP normal fuel operating conditions and performance through special operational monitoring, testing, surveillance, and inspection programs for the NGNP prototype. Related working group observations on the potential existence of large uncertainties and the technical challenges that limit the ability to measure conditions and detect potential anomalies in HTGR cores during normal operations are provided in several specific contexts below and more broadly with respect to the NGNP prototype in Section 3.13. This issue is a follow up item.

[9] Related RAIs: F3/M3, B5, B47, B48

[10] Related RAIs: F3/M3, F6/M7, F7/M8, F10/M13, F11/M15, F22, F23/M28, B29

3.2.2 Fuel service conditions and performance requirements for NGNP accidents

Fuel service conditions for NGNP accidents involve aspects such as fuel particle maximum accident temperature and time-at-temperature (i.e., core heatup accidents, reactivity excursions), maximum fuel particle oxidation (depressurization accidents with air ingress), and maximum fuel kernel chemical attack (moisture ingress accidents).

The accident service conditions assumed as the basis for AGR accident testing of the NGNP fuel are derived from what the Project states to be a conservative assessment of the best available information on the nuclear, thermal, and chemical environments that predicted to arise during all presently anticipated LBEs in a preliminary NGNP design with a prismatic block core.

Accordingly, the fuel design service conditions for NGNP accidents, like those for NGNP normal operations, have not yet been finalized and are considered items for follow up. When finalized, it will thus be necessary to show that they have been adequately addressed by the fuel accident conditions tested in the NGNP/AGR Fuel Program.

Additional considerations on fuel accident conditions and performance requirements[11]

It is the working group's view that, going forward, the Project's approach to establishing fuel service conditions and performance requirements for NGNP accidents should further address the following considerations:

- The analyses used in deriving fuel service conditions and performance requirements should employ validated reactor and system analysis tools with appropriate quantification and treatment of uncertainties.
- Fuel service conditions and performance requirements for NGNP accidents should be explicitly linked to a bounding set of design-specific events derived from a suitably broad spectrum of potential LBEs, including BDBEs. Reactivity excursion events are of particular concern in this regard, as are moisture and air ingress events.
- For analyzing reactivity excursions, as well as the potential for spatial xenon oscillations, the working group sees a likely need for 3D spatial reactor kinetics models with thermal-fluidic feedback to support or replace any models based on point or 1D reactor kinetics.
- For air ingress events, any necessary requirements for irradiated fuel element graphite and matrix materials to perform in ways that quantifiably prevent, delay, or limit the oxidation of fuel particle coatings should be clearly established and addressed by testing, as should the performance requirements for fuel particles ultimately exposed to chemical attack under such accident conditions (e.g., oxygen partial pressure, temperature, duration).
- Stated accident performance requirements should also be linked to stated criteria for allowing the continued use of fuel after accidents.

With regard to reactivity excursions, the NGNP/AGR Fuel Program Plan states the following key assumption:

[11] Related RAIs: F22, F23/28, F33, F34/M38, F41/M45, B6, B22, B27, B30, B31, B32, B33, B64, B66

"Radiologically significant reactivity transients are precluded by inherent characteristics of the design. Thus, no reactivity insertion accident testing is planned."

The working group notes that the potential for rod ejection accidents will require thorough evaluation in this context, as will potential reactivity insertions from enhanced neutron moderation in moisture ingress events. It is presently not clear to the working group what the potential severity would be of such power excursions or how they are being considered by the Project in relation to this key assumption.

Analyses of rod ejection accidents are presently required for current light water reactor (LWR) designs. Such analyses may thus be required for the NGNP prototype absent a compelling case to the contrary. If HTGRs have certain advantages in this regard over LWRs because of design-specific characteristics such as rod-ejection engineered safety features, rod-ejection mechanics, shorter fuel thermal time constants, or particular reactor kinetics parameters (e.g., longer neutron migration length, longer prompt neutron lifetime), then this will become clear in the course of the analysis. Understanding the resulting dynamic fuel service conditions and fuel behavior will be important in any case. Any predictive models of TRISO fuel performance (e.g., the PARFUME code) used to evaluate fuel performance and potential testing needs under pulsed power conditions should be qualified and assessed for that purpose. Should the results of any required reactivity excursion analysis reveal a need for pulsed power fuel tests, this would add significant scope to the Project's currently envisioned accident testing program for NGNP fuel. Reactivity excursions are further discussed in Section 3.10.5 below.

Fuel accident conditions and performance requirements for (a) reactivity excursions, (b) air ingress events, and (c) moisture ingress events are considered major items for follow up.

<u>3.2.3 Clarity and adequacy of fuel performance terminology</u>[12]

The Project's current definition of TRISO fuel particle failure is based on fission gas release as a result of mechanical/structural failure of all the particle coatings. This definition of TRISO fuel particle failure gives rise to potential concerns in view of the following observations:

- NGNP fuel performance requirements for as-fabricated fuel quality and in-service fuel failure are specified in terms of quality requirements for all coating layers (as indicated in FQ white paper Table 16), and
- TRISO fuel particle failure mechanisms are classified as either mechanical or thermochemical in nature and apply in some cases to specific coating layers (as indicated in FQ white paper Section 3.1.2).

In the prior history of U.S. and international efforts to develop and qualify HTGR TRISO fuel, fuel particle failure has generally been defined in terms of excessive releases of radionuclides from the particle. This includes releases of metallic (solid) fission products, such as Sr or Cs isotopes, as well as gaseous fission products, such as Kr and Xe isotopes, or a combination of both. The functional status of the SiC layer is of particular importance. Irradiated fuel particles with a failed (or defective) SiC layer will release Cs at HTGR operating temperatures but will not

[12] Related RAIs: F15/M20, F24/M29, F48/M52, F49/M53, M86, M73, M115, B11, B13, B77, B78

release gaseous fission products. The release of gaseous fission products requires the functional failure of both PyC layers as well as the SiC layer.

Defining particle failure to include only failures that release gaseous fission products appears to discount the potential importance of metallic fission product releases and seems excessively tied to the fuel performance model being developed by the Project, which is primarily mechanical in nature. Past work on modeling TRISO fuel performance under accident conditions has taken high Cs release as indicating failure of the SiC layer. It further bears noting in this regard that Cs has been observed to migrate through structurally intact SiC layers at elevated accident temperatures and that SiC decomposition becomes a dominant failure mechanism at the extreme accident temperatures considered possible in past HTGR designs like Fort Saint Vrain.

Defining fuel quality (as-fabricated) based on explicit limits on the fraction of defective particle layers suggests that one might also judge the irradiation and accident performance of TRISO fuel particles by similar criteria, noting that degradation and failure mechanisms associated with irradiation and accident conditions are generally attributed to specific particle layers. A particle with one or more defective (as-fabricated) or service-degraded coating layers generally has a higher probability of failure during continued operation and in accidents. Thus, any partial coating layer degradation or failure that occurs under accident conditions will have to be considered in determining whether the reactor can be restarted with the same fuel that experienced the accident event.

To clarify this issue, the Project should establish explicit definitions with descriptive terms like defective, failed, and functionally-failed relative to fuel particles and individual coating layers and explain how fuel performance and radionuclide transport and release are considered and modeled in each case. This issue is a follow up item.

3.3 Establishment of NGNP fuel designs with product specifications and process specifications for NGNP fuel fabrication[13] (FQ2) (ST2/3(a))

The objective of fuel design, fuel fabrication product specifications and fuel fabrication process specifications is to produce fuel which has the requisite high level fuel performance and has the requisite low level fuel radionuclide releases during NGNP normal operation and NGNP LBEs (i.e., transients and accidents). Achieving the requisite high level fuel performance and low level fuel fission product releases is critical to enabling the safety analysis to show that the NGNP satisfies the top level NRC requirements in terms of dose consequences for siting, occupational exposures and the Commission safety goals, and the NGNP operator's objective of having doses that are below the Environmental Protection Agency (EPA) protective action guidelines at the NGNP exclusion area boundary (EAB) for all LBEs.

To allow for uncertainties in the mechanistic source term analysis models and methods, the fuel design and associated specifications include an assumed factor-of-4 conservatism in fission gas release from the core and a factor-of-10 conservatism in metallic fission product release from the core. The Project states that these design margins (i.e., uncertainty factors) are largely

[13] Related RAIs: F43/M47, M62, M63, M64, M73, M115, B12, B15, B16, B33, B45, B52, B78, B81

based on engineering judgment. The Project indicates that as fuel performance and fission product transport models are developed and validated with experimental data, it may be possible to reduce the factors of conservatism in the future.

From the “conservative” allowable core releases, the corresponding in-service fuel performance requirements (e.g., fuel failure fractions, etc.) and, in turn, as-manufactured fuel quality requirements (e.g., heavy-metal contamination fraction, SiC defect fraction, etc.) are back-calculated. However, the product specifications do not yet explicitly include a back-calculation for fuel particle failure rates and fission product transport for LBEs involving chemical attack of the core and fuel.

The Project indicates that the largest sources of fission gas releases (including iodine and tellurium isotopes and noble gas isotopes) from the NGNP core are expected to be (1) as-manufactured heavy-metal contamination and (2) exposed fuel kernels. The fuel product specifications control the allowable fraction of heavy metal contamination (defective particles from manufacture and free uranium outside the particles) as well as the exposed kernel fraction (i.e., fraction of particles that experience failure of all coating layers). The latter specification involves the use of fuel performance models to predict particle failures. The Project states that, subsequently, the fractional releases of fission gases from heavy metal contamination and exposed kernels are predicted on a core-wide basis using experimentally determined release correlations.

For fission metal release, the Project states that in addition to releases from heavy metal contamination and exposed kernels, volatile metals (Ag, Cs, Sr) can also be released from fuel particles with defective or failed SiC coatings but with at least one PyC coating intact.

Volatile metals released from fuel particles are free to migrate through the fuel compact matrix, across the gap between the fuel compact and the graphite block, through the graphite web, and finally to be released into the circulating helium coolant. These additional barriers make the prediction of metallic fission product releases more complex and uncertain resulting in a larger (i.e., factor of 10) conservatism for metallic fission product releases and a factor-of-10 conservatism in developing the product specifications that affect metallic fission product releases from the fuel.

The Project states that for Fort St. Vrain the factors-of-conservatism goals for predicted- versus-measured gaseous and metallic fission products were met. For the NGNP design the Project believes that these conservative uncertainty allowance factors are both reasonable and attainable goals. Fuel performance and fuel fission product release predictive models and methods will be evaluated as part of the AGR-7 and AGR-8 code validation irradiation and accident condition tests.

The working group views the technical approach to the development of NGNP fuel design and product specifications as both rational and reasonable. However, the ultimate adequacy of these specifications will depend on the outcome of the AGR-3/4 fuel fission product transport data development tests, the AGR-5/6 fuel qualification tests and the AGR-7/8 fuel fission product transport code validation tests. The outcome of these tests will indicate the NGNP safety analysis codes and methods uncertainties and/or biases that must be accommodated in the NGNP safety analysis. These tests and their effects on the fuel product specifications are follow-up items.

At the time of the review, the Project had finalized many but not all aspects of the NGNP fuel design (e.g., particle packing fraction in fuel compacts). The Project states in the FQ white paper that it will need to finalize all aspects of fuel design and fuel manufacture for the fuel qualification irradiation testing and fuel safety testing in AGR-5/6.

3.4 Establishment of a fuel fabrication process that will meet the NGNP fuel product specifications[14] (FQ2) (ST2/3(a))

As is the case for all HTGR TRISO fuel forms, the fabrication processes that are used to manufacture both the fuel particles (i.e., fuel kernel, coating layers, overcoat layer) and the cylindrical fuel compact (prismatic block fuel) or spherical fuel element (pebble bed fuel), determine the fuel product properties, which in turn are critical to determining the performance of the fuel in terms of fuel particle failure rates and fuel radionuclide transport characteristics during normal operation and accident conditions.

In this regard, the Project has made significant efforts to develop a more scientific understanding of the relationship between fuel fabrication process and fuel product properties and the relationship between fuel product properties and fuel performance during normal operation as well as fuel performance during accident heat-up conditions. The Project has also devoted significant efforts to developing fuel fabrication equipment, fabrication processes, and fabrication process controls to apply this knowledge to the manufacture of fuel with fuel properties that meet the required level of fuel performance during normal operation and accident conditions.

The goal for fuel particle manufacture technology development is to achieve a fuel fabrication process that is capable of producing fuel at least as good as the fuel produced by German fuel fabrication technology in terms of heavy metal contamination, as-manufactured fuel particle defect rate, and in-reactor fuel performance. To develop fuel manufacture technology that meets the fuel performance requirements for the NGNP, the Project selected a TRISO coated particle design with a UCO fuel kernel, a kernel manufacturing process built on the German UO_2 kernel manufacturing process, and a particle coating process that replicates to the greatest extent possible the properties of the coatings of German TRISO coated fuel particles.

Fuel for the AGR-1 fuel irradiation tests was manufactured with production-scale fuel kernel fabricating equipment and processes and laboratory-scale equipment and processes for fabricating the coating layers and the compacts. Fuel for the AGR-2 irradiation tests was manufactured with production-scale equipment and processes for fuel kernel and coating layer fabrication and laboratory scale equipment and process for the compacts.

The AGR-1 and AGR-2 fuel has been irradiated in the ATR at design conditions representative of the NGNP core. To date, the performance of the AGR-1 and AGR-2 fuel indicates that the Project has achieved considerable technical knowledge and know-how in the manufacture of

[14] Related RAIs: F43/M47, B9, B16, B34, B36, B37, B40

fuel that can meet the NGNP fuel particle failure rate specifications for NGNP normal operation and heat-up accident conditions. The working group's preliminary view is that the TRISO fuel production-scale fabrication equipment and processes and controls have the potential to meet the fuel product specifications and the potential to meet the fuel performance requirements for the fuel design service conditions for NGNP normal operations and accidents.

The fabrication process and product specifications for the NGNP fuel qualification tests (i.e., AGR 5/6) have not yet been finalized. When finalized, the fuel manufacturing specifications, including the manufacturing process parameters and related acceptance criteria, and the fuel product parameters and related acceptance criteria for the fuel used for fuel qualification should be identical to those used for the manufacture of the production fuel for the NGNP reactor. The working group believes that the fuel to be used for NGNP fuel qualification tests (i.e., AGR-5/6) should be fabricated entirely with production-scale equipment and processes.

It is expected that the fuel for the NGNP core will be fabricated in a large fuel fabrication facility with a number of production lines for fabricating fuel kernels, production lines with coaters for coating the fuel kernels, production lines for over-coating particles and production lines for making fuel compacts. Each production line is expected to produce fuel product in lots and batches. The variability (e.g., mean and standard deviation) of attributes of the finished fuel will depend on the variability across the lines and the way the lots and batches are mixed to feed into the next step in the fuel fabrication process. On the other hand, the fuel for fuel qualification (i.e., AGR-5/6) will likely be fabricated from a single line involving a single piece of fabrication equipment for each step in the fabrication process (i.e., kernel, coating, over coating and compacting). The variability on fuel attributes for a production facility may be different from the fuel attribute variability for fuel made with a single line. Simulating the large fuel fabrication facility variability might be achieved by mixing several runs (i.e., batches, lots) from the single line. Whether and how the Project plans to address differences in the variability of product attributes between fuel fabricated for AGR-5/6 with a single line of "production-scale" equipment and fuel fabricated for the NGNP prototype with multiple lines of equipment in a large fuel fabrication facility is a follow-up item.

3.5 Establishment and implementation of a fuel fabrication statistical quality control procedure that demonstrates the fuel product specifications are met[15] (FQ2) (ST2/3(a))

NGNP fuel fabrication quality assurance program procedures, within the context of both the fuel fabricated for the NGNP/AGR Fuel Program as well as the fuel fabricated in the NGNP fuel fabrication facility for loading into the NGNP reactor core, must ensure a very low probability of accepting fuel whose attributes and properties do not meet the fuel product specifications. To demonstrate that the fuel attribute and property specifications have been met for the population with a sufficiently high confidence, reliable and accurate characterization methods (i.e., measurement techniques) must be established and standardized, acceptable and consistent sampling methods must be established, and standardized and acceptable statistical analysis methods must be established and consistently implemented.

[15] Related RAIs: F43/M47, B12, B15, B20, B21, B33, B34, B35, B36, B37, B40, B51, B53, B56, B60, B63

The FQ white paper provides selected limited information on the characterization methods for the kernel, coated particle, and compact product specifications as well as limited information on the sampling methods. For example, the FQ white paper states that characterization methods used for some product parameters are destructive (e.g., content of uranium in the fuel matrix) while the characterization methods for other product parameters are non-destructive (e.g. fuel kernel diameter, particle coating thicknesses, fuel compact length).

For IAEA Coordinated Research Project 6 (CRP-6), round-robin TRISO coated particle fuel characterization benchmark studies showed that different standardized characterization methods can result in systematic and significant differences in variable property measurements (e.g., coated particle diameter). Accordingly, the working group's view is that the NGNP fuel characterization methods used in the fuel qualification program should be used by the NGNP production fuel fabrication facility. Any significant changes proposed with regard to characterization methods should be fully assessed prior to being implemented.

The white paper includes a limited overview on the statistical analysis methods and acceptance criteria. Fundamentally, the statistical methods involve a statistical analysis of a product attribute property or product variable property to determine whether the population from which the sample was taken should be accepted as meeting the product acceptance criterion or rejected as not meeting the product acceptance criterion. If the acceptance test has a 95% confidence level, there is no more than a 5% chance of accepting a product attribute that should be rejected. This means that there is a 5% chance that selected fuel product attribute (e.g., fuel kernel diameter, SiC thickness) might be accepted as meeting the specification but should have been rejected.

However, the Project observes that as the true value of a property in a population that is within the specification approaches the specification limit, the minimum sample size that will be needed in order to accept the population at the 95% confidence level (and to avoid false rejection of the population) becomes large. As such, for economic reasons, it will be important for the fuel manufacturer for the fuel qualification program as well as for the NGNP prototype to seek to achieve a quality level that is significantly better than specification requirements to avoid excessive rejection of good product with reasonable sample sizes.

Sensitivity studies should be conducted for significant fuel product variable properties using a mechanistic fuel performance code to assess the effect of fuel outside the specification on fuel particle failure probabilities during normal operation and accident conditions. The results of such sensitivity studies should be used in part to support the confidence levels selected for the NGNP production fuel fabrication QA statistical analysis procedures.

The fuel characterization methods and fuel fabrication statistical quality control procedures used for the NGNP/AGR fuel qualification program and NGNP reactor production fuel fabrication facility are follow-up items.

3.6 Demonstration that fuel performance requirements for *normal operations* are met by irradiating a statistically significant quantity of fuel at NGNP fuel design conditions, monitoring fuel irradiation performance, and conducting post-irradiation examinations (FQ2) (ST2/3(a))

3.6.1 Adequacy of accelerated irradiation testing[16]

Fuel performance data in the NGNP/AGR Fuel Program will be based solely on an accelerated irradiation testing program conducted in the ATR. In response to RAI questions, acceleration factors in the completed, ongoing, and planned AGR fuel irradiations were stated to range from 1.6 to over 3. The lack of fuel performance data obtained in real-time HTGR neutron environments is a concern. This concern is based on the questionable adequacy of data generated solely in accelerated irradiation environments. These working group concerns are heightened in view of the fact the normal fuel operating condition parameters (temperature, burnup, fluence) targeted for the NGNP UCO prismatic fuel are significantly more demanding than the conditions targeted in the German program and elsewhere for UO_2 pebble fuel. The data generated will be used to refine/develop, verify, and validate models and codes designed to predict fuel performance and fission product transport under all normal operating and accident conditions in an actual NGNP plant. The adequacy of the AGR data is particularly questionable with regard to time-at-temperature dependent phenomena such as fission product corrosion and attack of coating layers (e.g., Pd, Cs). Such phenomena can reduce the retention of metallic fission products in particles with structurally intact coatings and also weaken the coating layers.

Prior HTGR fuel qualification programs (US, Germany, Japan, and China) have employed fuel irradiation testing in the real-time neutron environments of HTGRs as well as in the accelerated neutron environments of MTRs. This historic methodology has proven effective for evaluating the performance of prior HTGR TRISO fuel designs in a reasonable period of time. The current NGNP/AGR Fuel Program proposes to develop and qualify an advanced UCO fuel concept with significantly improved fuel fabrication characteristics and excellent in-service performance under demanding operating and accident conditions of high temperature, burnup, and fluence. Performance data gathered in-reactor and during post-irradiation testing are being used to refine and develop predictive fuel performance/fission product transport models/computer codes applicable to all normal operations and all perceived accident conditions.

Fuel performance and fission product behavior data obtained under real-time NGNP irradiation conditions should be an essential component of the UCO fuel performance database. Test specimens obtained from real-time irradiation environments contain the proper mix of fission/activation products generated under actual irradiation conditions. Such fully prototypic data should thus be considered essential for adequately understanding fuel performance and for developing predictive models of fuel radionuclide retention and transport. Potential fuel performance issues to be more fully addressed by fully prototypic irradiation testing would include, among others, those associated with plutonium burnup and time-at-temperature effects from palladium, silver, rare earths, and cesium.

The working group notes that the NGNP prototype can be used to address all issues mentioned here and in Sections 3.2.1, 3.6.2, and 3.6.3 concerning the non-prototypicality of MTR irradiation conditions (i.e., HTGR fuel irradiation times, neutron spectra, path dependences, operating condition uncertainties). Such uses of the NGNP prototype are further discussed in Section 3.13. The inclusion of a suitably designed post-irradiation fuel inspection and testing

[16] Related RAIs: F1M/1, F2/M2, F3/M3, B49, B57, B58, B61, B69, B73, B82

program for the NGNP prototype can provide the important confirmatory fuel performance data for the UCO TRISO coated fuel particle design. Periodic inspection of irradiated test fuel based on detailed post-irradiation examinations and accident test simulations (modeled after the German accident testing program) can provide real-time performance data in the UCO database. Subsequently, early data would be available to refine the fuel performance/fission product transport models and codes and verify their predictive results. Validation of the NGNP fuel performance/fission product transport models and codes may require an effort completely independent from data gathering inspections. Independent code modeling predictions, followed by an independent evaluation of reference fuel performance under normal operating and accident condition simulation carried out on irradiated fuel from the NGNP prototype, appear to be feasible. Satisfactory completion of such a prescribed phase in a post-irradiation fuel inspection and testing program achieved with irradiated fuel from the NGNP prototype is considered a final confirmatory step in the NGNP UCO fuel qualification program. The working group considers this issue a major follow-up item.

3.6.2 Adequate plutonium generation and burnup in AGR fuel test irradiations[17]

Neither the white papers nor their supporting reference documents included information on plutonium burnup in the NGNP core or the AGR fuel test irradiations. As noted above in Subsection 3.2.1, the working group's view is that plutonium burnup should be among the normal operating service condition parameters specified for NGNP fuel. The following paragraphs describe the technical basis for that view.

When 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 particle performance and radionuclide 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, 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 resulting inventories of fission and activation products 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 degrade the integrity and retentiveness of TRISO fuel particles. In particular, the 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 is 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 plutonium fission fraction will vary with changes in the neutron energy spectrum. An HTGR spectrum tends to 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

[17] Related RAIs: F1/M1, F3/M3, F4/M5, F5/M6, F9/M11, M12, F10/M13, B44, B46, B47, B49, B57

near the 0.3 eV thermal 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 and retentiveness.[18] 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."[19]
- 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."[20] One could further hypothesize that the effects of silver diffusion on SiC grain boundaries could also increase grain boundary diffusion of cesium.
- Information needed for evaluating the effects of different neutron energy spectra in MTRs versus HTGRs includes 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 high-burnup irradiations of TRISO fuel in MTRs, one could increase the plutonium burnup fractions by doing some combination of the following:
  - Reducing the tested 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 tested fuel kernels with PuO_2/PuCO.

In partial response to related NRC questions, the Project provided TEV-1022,[21] a technical report with preliminary calculation results showing that, at a total fuel burnup of about 20%

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[18] *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.*

[19] *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.*

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

[21] J. Maki, J. Sterbentz, "Response to Questions about the Applicability of the AGR Test Results to NGNP Fuel," Technical Evaluation Study, TEV-1022, INL, September 30, 2010.

FIMA, plutonium burnup is 63% higher, palladium inventory 49% higher, and silver inventory 66% higher in a preliminary NGNP prismatic core design than in the AGR-1 test irradiations performed in the ATR. The Project has not provided requested results of similar calculations for the present AGR-2 irradiations nor any of the subsequent irradiations and the planned series of AGR tests. This is a follow-up item. The Project also stated that its approach to increasing plutonium burnup in the AGR irradiation tests relies solely on using neutron absorbers in the test rig to effectively harden the thermal spectrum by reducing the neutron flux in the lower range of the ATR thermal energy spectrum. In view of the AGR-1 analyses noted above, the working group presently views this approach as unlikely to adequately address the issues raised in this section.

The Project also provided a requested summary of the current state of knowledge on how palladium, silver, and rare earth fission products can affect TRISO fuel performance. On reviewing the information provided and noting the currently limited understanding of governing phenomena, it is the working group's view that plutonium burnup, time at operating temperature, and particularly palladium time-at-temperature are important parameters that should be considered in the irradiation testing of TRISO fuel. This issue area is considered a major item for follow up.

Finally, the working group notes that the Project has generally not shared with NRC its pre-test predictions of fuel irradiation conditions (i.e., flux spectra, Pu burnup, etc.) and fuel performance in the past and current AGR irradiation tests.[22] Post-test analyses of AGR-1 irradiation conditions were provided in TEV-1022 only in delayed partial response to a specific NRC request. The Project has not responded to the working group's RAI requesting pre-test predictions of the recently completed AGR-2 irradiation nor of any future AGR irradiations. The Project should freely share all test design and pre-test predictions of AGR irradiation conditions and irradiation fuel performance. This will enable NRC to evaluate with greater confidence the predictive capabilities of the Project's developmental fuel performance models as well as issues of test applicability and adequacy. This issue is an item for follow up.

3.6.3 Evaluation of irradiation test conditions[23]

As noted in a related RAI question, given the central importance of TRISO fuel performance to the NGNP safety case, the working group has given consideration to performing independent NRC analyses of AGR test irradiation conditions and associated fuel burnup isotopics and would be willing to pursue arrangements for gaining access to the detailed ATR information that would be needed for doing so. This is an item for follow up.

In presenting information on the AGR-1 and AGR-2 irradiation tests, the Project has noted that many of the thermocouples used to monitor the control of irradiation temperatures failed as the test irradiations progressed. In response to a related RAI question on how such thermocouple failures are accounted for in evaluating irradiation temperatures and associated uncertainties, the Project explained how thermocouples embedded in the graphite sample holders are used in conjunction with detailed analytical models to determine fuel temperatures in the AGR

[22] Related RAIs: B62, B63

[23] Related RAIs: F5/M6, B82

irradiation tests. After briefly describing a general approach to evaluating temperature uncertainties, the Project's RAI response noted that detailed uncertainty calculations for the AGR-1 irradiation test were then underway and should become available in early 2012 along with detailed reports of the final AGR-1 physics and thermal analyses and supporting sensitivity analysis. Further assessment of AGR fuel irradiation temperature uncertainties and how they are affected by thermocouple failures is an item for follow up.

3.7 Demonstration that fuel performance requirements for *accident conditions* are met by safety testing a statistically significant quantity of irradiated fuel at NGNP accident conditions and monitoring fuel accident performance (FQ2) (ST2/3(a))

3.7.1 Applicability of post-irradiation heating tests to fuel performance in HTGR accidents[24]

The NGNP/AGR Fuel Program, like the earlier German TRISO fuel program on which it builds, uses out-of-reactor post-irradiation heating tests to develop data on TRISO fuel performance in accidents. The NGNP/AGR Fuel Program's accident condition heating tests are expected to be performed weeks or months after ATR high-power irradiations are completed. However, the peak fuel temperatures in an HTGR heatup accident are expected to occur during high-power irradiation in the case of reactivity excursion events and within about a day or so after high power irradiation ends due to active or passive shutdown in the case of events with depressurized loss of forced cooling. Some MTRs have the capability to heat-up the fuel at power or within a day or so after high power irradiation is stopped, thereby more closely simulating actual fuel radiochemical conditions. Some historic HTGR fuel qualification safety tests have been conducted in such MTRs. It is therefore important to understand the extent to which delayed fuel heatup testing reproduces or bounds the physical phenomena that could potentially degrade TRISO fuel performance in actual HTGR heatup accidents.

The NGNP/AGR Fuel Program is planning to re-irradiate some irradiated fuel compacts just before the heating tests. In partial response to an RAI question, the Project noted that the fission power levels and irradiation times achieved in the planned re-irradiations will be much lower than those required to produce the inventories of short-lived fission products expected to be present during an HTGR heatup accident. The stated purpose of the planned re-irradiations is to produce short-lived radionuclides (e.g., I-131) in quantities large enough to permit their measurement during post-irradiation heating tests. The re-irradiation of the fuel thus allows data on short-lived radionuclide transport to be obtained, which would otherwise not be possible. The Project further indicated that, because the masses of fission product elements in HTGR fuel heat-up accidents will be dominated by stable and long-lived isotopes, the elemental inventories within the test fuels will be prototypical. However, no calculations of nuclide generation, depletion, and decay (e.g., ORIGEN code results) were provided to support this conclusion.

To assess the effects of delayed testing on fuel particle performance, a quantitative comparison of the respective inventories of all elements produced by fission, activation, and decay would first be needed to determine any substantial elemental inventory differences. This would then be used to assess how the respective differences in elemental inventories could potentially

[24] Related RAIs: F2/M2, F19/M25, B18

affect fuel particle performance and how fuel performance could be affected by other changes in fuel composition (e.g., species migration, chemical reactions, phase changes) that might be expected to occur during extended periods of post-irradiation cooling and decay. Assessment of the applicability of delayed fuel heating tests to fuel performance in HTGR accident conditions is a follow-up item.

Once the NRC has reviewed the available analytical and experimental bases for using delayed fuel heating tests to obtain experimental data on fuel performance in HTGR heatup accidents, it should be possible to assess whether any additional fuel heating tests during or shortly after irradiation might be needed. The working group notes that at-power heating tests have been performed on earlier U.S. designs of TRISO fuel.[25]

3.7.2 Scope of fuel performance testing for LBE accident conditions

Potential needs for specific fuel performance tests and data to address potential reactivity excursion events, moisture-ingress events, and air-ingress events are discussed in Sections 3.2.2, 3.10.5, and 3.10.6 of this document. Associated issues are noted in the respective sections as items for assessment follow up.

3.8 Irradiation and accident proof testing of NGNP fuel fabricated on the production lines of the NGNP fuel fabrication facility[26] (FQ2) (ST2/3(a))

FQ white paper Section 5.3.6, "Production-Scale Fuel Manufacturing Facility for NGNP UCO Fuel," states that the NGNP/AGR Fuel Program does not include implementation of a capability to mass-produce fuel for the NGNP, nor does it include qualification of fuel produced in an NGNP fuel-fabrication facility.

The FQ white paper references INL 200765, which discusses two fuel supply options for an NGNP with a prismatic block core. The first option calls for construction of a pilot fuel fabrication facility (FFF) at INL to produce UCO fuel for the NGNP. The second option calls for a portion of the initial core for NGNP to be produced using the current pilot-scale fuel line at B&W (with modifications), which is currently being used for fabrication of irradiation test fuel, and to subsequently build a larger fuel-fabrication facility to both complete production of the first core and to produce reload fuel.

The FQ white paper further states that both an irradiation proof test and post-irradiation heating tests (of fuel produced in the FFF) will be needed to demonstrate the acceptable performance of the fuel and thereby qualify fuel for the NGNP. To accomplish this, representative fuel compact samples will be taken from the FFF process line for an irradiation proof test and subsequent post-irradiation heating tests. The white paper states that it is expected that the proof test will be conducted in the ATR and will utilize the same test train design as used for the AGR-5/6 fuel-qualification test.

[25] "Postirradiation Examination of Capsules P13R and P13S," GA-A13827, GA Technologies, Inc., San Diego, CA, October 1976; and HTGR Technology Development Program, Annual Progress Report for Period Ending December 31, 1982, ORNL-5960, June 1983, Section 9.3.2, pages 207-209

[26] Related RAIs: F26/M31, B55

However, in response to an RAI question, the Project stated that "….if significant changes were made to the fuel production equipment or processes thus deviating from those used for the AGR qualification fuel, it is expected that an irradiation proof test of the mass-produced fuel for the initial core would be conducted by the Project and/or the NGNP fuel vendor. This proof test would include PIE and post-irradiation heating tests expected to be largely confirmatory of AGR-5/6 and AGR-7/8."

The working group's view is that fuel produced by the NGNP FFF is likely to involve significant differences in the fuel production equipment, processes, and characterization methods. Accordingly the working group believes that both irradiation proof testing and post-irradiation heating tests of fuel produced in the FFF should be conducted to demonstrate the acceptable performance of the FFF fuel and to qualify the FFF fuel for the NGNP reactor. It is anticipated that the FFF fuel irradiation qualification testing can be conducted on a schedule that would not adversely impact the NGNP prototype startup schedule. This is a follow-up item.

3.9 Definition of event-specific mechanistic source terms for NGNP[27] (ST1)

The MST white paper solicits NRC agreement that the proposed definition of event-specific mechanistic HTGR source terms is acceptable. In response to an RAI question, the Project provided a clarified definition of "event-specific mechanistic source terms" as follows:

- *HTGR Source Term* – Radionuclides released from the reactor-building of a modular HTGR plant to the environment.
- *Mechanistic HTGR Source Term* – A modular HTGR Source Term that is calculated using models that use first principle methods supported as needed by empirical confirmation to represent the mechanisms (phenomena) that affect the generation and transport of radionuclides in the plant.
- *Event Specific Mechanistic HTGR Source Term* – A Mechanistic HTGR Source Term that is calculated for a specific LBE.

The working group finds this clarification useful and makes two observations. First, while the definition of source term as the release of radionuclides from the reactor building to the environment is appropriate for accident consequence calculations and emergency planning, the radionuclide release into the reactor building is an important consideration in the regulatory examination of barrier-based defense in depth (DID). That is, the DID provided by the last physical barrier (containment or reactor building) to the release of radionuclides to the environment. Second, event specificity is implied with regard to calculating the source terms for the selected LBEs. The working group believes a conclusive assessment of HTGR mechanistic source terms includes consideration of events for which source terms are to be mechanistically calculated and which result in bounding source terms.

The regulatory examination of DID capabilities (see Title 10 Code of Federal Regulations, Part 100 (10 CFR 100)) requires that a large release of radioactivity from the reactor coolant system to the reactor containment be hypothesized, consistent with expectations of a major accident at the reactor facility. This regulatory requirement is "technology neutral," predicated on the

[27] Related RAIs: M4, M9, M65, M66, M82, M83, M84, M86, M87, M89, M113

potential for severe events that could result in substantial releases of radioactivity from reactor fuel. The requirement is imposed to ensure that the ability to mitigate potentially severe consequences is duly considered, in tandem with the ability to prevent severe core damage events, in evaluating the adequacy of DID measures in the design of barriers to radionuclide release from the nuclear plant. That is, appropriately severe events should be considered in developing the bounding mechanistic source terms for demonstrating compliance with 10 CFR 100 requirements, and in showing consistency with the safety expectations conveyed in the Commission Policy Statement on the Regulation of Advanced Nuclear Power Plants.

The intent of working group's RAI question on HTGR severe accidents and resulting source terms was to stress the point that severe events must be considered for calculating the bounding source terms. The Project is correct in noting that the LWR oriented containment source term definition invoking a severe accident with extensive fuel melting is not applicable to modular HTGRs. The definition more pertinent to modular HTGRs would be the severe event induced releases to the reactor building and to the environment of (a) radionuclides released from fuel elements resident in the core during the accident and (b) long-lived radionuclides that have gradually accumulated in the primary system over many years of normal operation. The working group believes that BDBE's significantly more severe than those considered to date in the white papers on MST and LBE selection should be evaluated for calculating bounding source terms.

Additional discussions of the working group's views on LBE selection and methods for developing mechanistic source terms to demonstrate adequate barrier DID are provided in the NRC assessment document for the three NGNP white papers that describe the risk-informed and performance-based licensing approach proposed by the Project.

In summary, the working group's view is that the Project's definition of event-specific mechanistic source terms for the HTGR is generally consistent with the traditional staff definitions. However, the working group believes that appropriate consideration should be given to all available barriers in the assessment of event-specific mechanistic source terms. This is a follow-up item. The outcome of fuel performance testing (both in-pile and out-of-pile) in the NGNP/AGR Fuel Program should provide additional insights in this regard.

3.10 Establishment and validation of models for fuel performance and radionuclide transport in fuel particles and fuel elements (ST2/3(a))

3.10.1 Diffusion data for release from fuel elements[28]

Use of effective diffusion coefficients

In SECY-93-092, the staff made the following recommendation on source terms for the MHTGR (i.e., a proposed modular HTGR design) and other advanced reactor designs then undergoing pre-application review:

[28] Related RAIs: M72, M76, M108

"Advanced reactor and CANDU 3 source terms should be based on a mechanistic analysis and will be based on the staff's assurance that the provisions of the following three items are met:

- *The performance of the reactor and fuel under normal and off-normal conditions is sufficiently well understood to permit a mechanistic analysis. Sufficient data should exist on the reactor and fuel performance through research development and test programs to provide adequate confidence in the mechanistic approach.*
- *The transport of fission products can be adequately modeled for all barriers and pathways to the environs, including specific consideration of the containment design. The calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured.*
- *The events conserved in the analyses to develop the set of the source terms for each design are selected to bound severe accidents and design-dependent uncertainties.*

The design-specific source terms for each accident category would constitute one component for evaluation the acceptability of the design."

In its staff requirements memorandum (SRM) of July 30, 1993, the Commission approved the staff's recommendation on source terms. In SECY 03-0047, the staff recommended that the Commission retain the guidance contained in the July 30, 1993, 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.

NUREG/CR-6844, Vol. 1, "TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables for Fission Product Transport Due to Manufacturing, Operations, and Accidents," discusses a range of mechanisms identified by researchers as potentially playing an important role in the transport of fission products within the constituent materials of TRISO coated particle fuels. These include vapor transport via Knudsen diffusion for gaseous fission products; intercalation of alkali and alkali-earth fission products like Cs and Sr in the PyC layers; and grain boundary diffusion, surface diffusion, and bulk diffusion. Trapping mechanisms and temperature gradient driven diffusion (i.e., Soret effect) have also been observed and modeled.

The Project proposes that the model for transport of long-lived fission products in the coated particle and surrounding fuel element materials be simplified into a single transport equation using effective diffusion coefficients. The modeling consists of solving Fick's second law equation for concentration gradient driven diffusion with an effective diffusion coefficient for each fission product species. The effective diffusion coefficient for each species would generally be represented by an Arrhenius type equation as a function of temperature. The proposed approach for modeling fission product migration through the constituent fuel materials in the diffusion does not explicitly separately model all the phenomena and mechanisms in the previous paragraph.

The Project states that many different approaches have been used to characterize radionuclide transport in HTGRs that range from laboratory measurements to reactor surveillance programs at the seven HTGRs that have been built and operated, to atomistic modeling on supercomputers in recent years. While the approaches have been diverse, the transport models and material property correlations used to predict radionuclide transport in support of reactor design and safety analysis are, in general, based upon experimental data that have been correlated with phenomenological models based on first principles. Often, correction factors are added to the first principles model to account for irradiation effects. The NGNP Project notes that there is insufficient data to effectively and explicitly model all of the phenomena that have been postulated and submits that the several decades of experimental data acquisition, model development, code benchmarking and code validation based on an effective diffusion approach provides sufficient basis for the proposed approach.

It is the working group's view that the proposed use of effective diffusion coefficients in connection with the use of Fick' second law is generally reasonable, but should be confirmed. In this regard, the Project's response to a related RAI question stated that once the AGR-3/4 test data become available, alternative transport models would be considered to correlate the data if the current Fick's 2nd Law diffusion-based model is determined to be inadequate. The project further stated that if a more complex model is ultimately adopted, supplemental testing would likely be needed to obtain the supporting material property data. NRC reviews of the AGR-3/4 test data, diffusion coefficient analyses and assessment of the potential need for alternative transport models are follow-up items.

Issue resolution for flux-accelerated diffusion of metallic fission products during irradiation[29]

To a substantial degree, the German UO_2 test data for TRISO-coated particle diffusion rates published in IAEA-TECDOC-978 are based in on post-irradiation heating tests. However, in response to an RAI question raising the issue of flux-accelerated diffusion of cesium through intact SiC layers, the INL document TEV-1022 states that, "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) obtained from post-irradiation heating tests is not recommended because those coefficients do not consider the irradiation effects, either implicitly or explicitly." The Project states that it plans to pursue a critical review and analysis of the historical data on both in-pile and out-of -pile fission product diffusion in TRISO-coated particle fuel. For the AGR-2 and AGR-7 tests, the Project states that the release of fission products under irradiation will be measured via PIE, that these measurements will be analyzed to establish diffusion coefficients under irradiation, and that the resulting diffusion coefficients will be compared to the historic values from IAEA-TECDOC-978. NRC staff evaluation of the results of these planned actions is a follow-up item.

Additionally, for AGR-3/4 the irradiation, post-irradiation and safety testing, are intended to obtain data on fission product transport through NGNP fuel matrix and fuel element graphite with a known source of fission products in the fuel compact. This will allow measurements to evaluate the fission product gradient across the matrix and graphite surrounding the fuel compact via PIE. These gradients with knowledge of the irradiation temperature conditions will enable diffusion coefficients for the matrix material to be back-calculated. NRC reviews of the AGR-3/4 PIE data and diffusion coefficient analyses are also follow-up items.

[29] Related RAIs: F2/M2, M9, F16/M21, F21/M27, F36/M40, M78

Radionuclide transport in compact-to-graphite gap of the prismatic fuel element [30]

The Project states that for the calculation of event-specific mechanistic source terms for the prismatic core, the fuel compact-to-graphite gap is assumed to have no effect on the transport of gaseous fission products. Both the compact matrix and the fuel element graphite are relatively porous and provide very little resistance or holdup to the transport of fission gases (including halogens) released from the fuel particles. As such, any effect on the transport of fission gases of the compact-to-graphite gap is generally neglected with respect to mechanistic source term calculations.

The Project sates that in modeling metallic fission product transport during normal operations for event-specific mechanistic source terms, it is assumed that sorption equilibrium exists in the fuel compact-to-graphite gap. At equilibrium, fuel matrix sorption isotherms relate the metallic fission product vapor pressure in the gap and the solid phase concentration at the fuel compact surface. The isotherms are established experimentally. Similarly, at equilibrium, graphite sorption isotherms relate the metallic fission product solid phase concentration on the graphite fuel hole surface and the fission product vapor pressure in the gap. As such, the solid-to-gas phase vaporization and the gas-to-solid phase condensation of metallic fission products across the gap control the transport of metallic fission products across the gap. The temperature dependent sorptivity of the fuel compact matrix and the fuel element graphite control the transport of metallic fission products across the gap during normal operation and is credited and modeled in the calculation of event specific mechanistic source terms.

The NGNP/AGR Fuel Program Plan dated September 30, 2010, states that single-effects test data will be needed to develop and refine and sorptivity correlations in the fuel compact matrix and fuel element graphite with uncertainties within a factor of 10 at 95% confidence level. NRC review of the test data and the development of sorptivity correlations in the fuel compact matrix and fuel element graphite is a follow-up item.

For NGNP LBE transients, the effects of compact matrix and graphite sorptivity on metallic fission product transport across the gap are conservatively neglected. The working group views this approach as reasonable for use in the context of conservative consequence analysis.

<u>3.10.2 Modeling the transport of all radiologically significant radionuclides</u>[31]

The Project states that, while the analyses of fission product transport in modular HTGRs can include as many as 250 radionuclides (including all radiologically significant radionuclides), it is not necessary to collect data on all radionuclide species that are analyzed in the calculation of mechanistic source terms. The Project proposes to classify radionuclides and species into one of nine radionuclide classes which in most cases are based on the periodic table of elements. The Project proposes to develop experimental data on fission product transport for a representative radionuclide in each class (e.g., Cs-1 37 for alkali metals, 1-131 for halogens) and apply the fission product transport data and models to the other radionuclides in the class. This approach is similar to the approach taken for modeling fission product transport in LWR

[30] Related RAIs: M57, M62, M76

[31] Related RAIs: M58, M111

severe accident analysis. The working group's view is that the proposed approach is reasonable.

However, the Project states that the release of iodine (i.e., a halogen) from the fuel kernel will be assumed to be the same as the release of xenon (i.e., a noble gas) from the kernel, which is stated to be conservative based on historical measurements of iodine and xenon release from UO_2 and UC_2. The AGR fuel testing program is to confirm this for UCO fuel.

The working group views the approach to developing the experimental data needed for modeling the relatively large number of fission product species for NGNP mechanistic source terms as being reasonable.

3.10.3 Models and data for fuel particle performance during normal operation and heat-up accidents[32]

For prismatic core safety analyses, the Project proposes to use what they state to be a realistic, yet conservative, accident condition fuel performance (i.e., particle failure rate) model based on the model used by General Atomics. This model is described as the "1989 Goodin-Nabielek" model. The Project states that the model is based primarily upon German LEU UO_2 TRISO particle data. As post-irradiation heating data for the reference LEU UCO TRISO fuel become available from the on-going NGNP/AGR fuel development and qualification program, the 1989 Goodin-Nabielek model will be updated as appropriate. The model predicts how the retention of fission gases and volatile fission metals within the TRISO particles gradually degrades with time at high temperatures. Such degradation of fission product retention is evident from the German heating data for LEU UO_2 TRISO fuel.

The document, "Analysis of Fission Product Release Data for German Fuel Sphere HFR K3/3," ORNL/TM-12425, R.C. Martin, Oak Ridge National Laboratory, September 1993, provides a description and evaluation of the '1989 Goodin-Nabielek' model. The following observation is made in the document in connection with Figure 2 in the document with respect to the '1989 Goodin-Nabielek' model: "The fractional cesium release predicted by Eq. (3) (i.e., the 1989 Goodin-Nabielek model for the specific HFR-K3/3 irradiation and accident conditions) is compared with the experimental sphere release in Fig. 2. The agreement is good except for the initial 25 h of heating." Figure 2 shows that for the first 25 hours the model significantly over-predicts the fractional Cs-137 release data (by as much as an order of magnitude). However, the model under-predicts the experimental data after 25 hours of heating. As such, the basis for the statement that the model is "more realistic, yet conservative" for predicting accident heat-up particle failure fractions is unclear and is thus an item for follow up.

The subject reference also states that the SiC failure fraction associated with the 1989 Goodin-Nabielek model is given by:

$$\Phi = 1 - 2^{-(kt)m}$$

[32] Related RAIs: F32/M37, F49/M53, F50/M54, F51/M55, M76, M77, M91, M92, M93, M94, M96, B4, B7, B8, B10, B14, B23, B24, B25, B28, B38, B39, B43, B54, B62, B65, B67, B68, B70, B72, B79

Where m is the Weibull parameter, with m =2 recommended for heating test predictions and m=1.7 recommended for overall core performance predictions. No basis is provided for the use of m =2 for heating test predictions and m=1.7 for overall core performance predictions and the Project's response does not specifically discuss whether and how the NGNP/AGR Fuel Program will re-evaluate and/or revise the value of the Weibull parameter for the SiC layer of the NGNP UCO fuel particles for heating test conditions and core performance predictions. These are also items for follow up.

The subject reference also describes the range of validity and assumptions for the '1989 Goodin-Nabielek' model. The reference states that the decomposition frequency factor, k, shown in the equation above is represented by the Arrhenius equation:

$K= K_0\, e^{-Q/RT}$

where,

Q =activation energy for SiC corrosion

In the related RAI response, the Project does not specifically discuss whether and how the AGR fuel program will re-evaluate and/or revise the formulation of the decomposition frequency factor, k, for the SiC layer of the NGNP UCO fuel particles. This is therefore considered an item for follow up.

According to the above reference, the equations associated with the 1989 Goodin-Nabielek model were derived using Cs release data from particles with UO_2 kernels. The activation energy used for thermal decomposition is derived from heating data for un-irradiated particles with UO_2 kernels with the OPyC layer removed. The Project does not specifically discuss whether there is a plan to conduct tests to develop data on the activation energy for thermal decomposition using more prototypical particles with UCO kernels. This is an item for follow up.

The Project has provided no information on the fuel performance model that is proposed to be used in the prismatic core safety analysis to predict fuel particle failures during normal operation. However, the above reference further states in the "Range of Validity" section that although the model is derived from data obtained only at 1600 °C and above, the model could also be considered applicable at normal operating condition temperatures because the temperature dependence will render the model insignificant at lower temperatures. The reference also states that the fuel design data manual does not explicitly limit the range of validity of an earlier model using the same methodology being insignificant at these lower temperatures. This is an item for follow up.

The Project's response does not specifically discuss whether the 1989 Goodin-Nabielek model will be used over the range of fuel particle accident temperatures up to 1600 °C while the model is (will be) developed from accident condition data obtained at 1600 °C and above. Neither does the Project provide the basis for extrapolating the application of the model down to the lower accident fuel temperatures expected for the prismatic core designs. The Project does not explicitly state whether the NGNP/AGR Fuel Program will include the development of UCO fuel particle (SiC) failure data at lower accident condition temperatures (e.g., 1250 °C to 1600 °C) for use in developing the model to be applied to the NGNP. These important details are thus considered items for follow up.

The Project does not indicate in the white papers or its responses to RAIs whether the model will be used for predicting fuel particle (i.e., SiC) failure rates for the temperature range associated with prismatic core fuel operating temperatures. The use of the model for predicting particle failure rates during normal operation is thus an item for follow up.

With respect to degradation of the SiC layer due to corrosion, the Project's response to a related RAI states that the chemistry of UCO fuel ensures that the attack of SiC by rare earth fission products is prevented because those elements are in a stable oxide form. The Project further states that carbon monoxide (CO) production is also prevented as long as UC_2 and UO_2 are both present in the fuel kernel. The working group notes that with respect to SiC corrosion by CO during normal operation, the Project's response to a related RAI question uses a thermodynamic argument to discount the presence of CO within fuel particles with UCO kernels. It is not entirely apparent that thermodynamic properties determined under laboratory conditions will be directly applicable to materials exposed to long-term, intense irradiation, which is known to cause crystalline materials to evolve toward more amorphous states. This is a follow-up item.

In this regard, the Project's response to a related RAI states that post-irradiation heating tests performed in the past on LEU UCO TRISO particles indicate that the dominant corrosive mechanism for high-temperature failure is SiC corrosion by fission products rather than by CO. The working group expects that the NGNP/AGR Fuel Program will conduct PIE to confirm that high-temperature corrosive degradation of SiC in NGNP UCO fuel is predominantly due to corrosion by fission products rather than by CO. These are also items for follow up.

The experimental data base for the NGNP TRISO coated fuel particle performance (e.g., SiC failure) modeling is expected to represent a relatively small sample size (e.g., tens of thousands to hundreds of thousands of fuel particles) compared to the billions of particles in the NGNP core. To address uncertainties in the failure model caused by limited sample size, designers typically use statistical analysis to conservatively bound the failure data at different confidence levels (e.g., 50%, 95%). The Project does not describe any statistical analysis of the relatively limited fuel particle failure (or SiC decomposition) data that will be conducted for developing the 1989 Goodin-Nabielek model for application to NGNP licensing safety analysis. This is an item for follow up.

In the response to a related RAI question, the Project stated that, because the model used for prismatic modular HTGRs addresses more coated particle fuel failure modes, it lends itself to a more detailed evaluation of the confidence level associated with calculated fuel particle performance that is based on the level of uncertainty associated with each potential fuel failure mode. The Project does not describe how the NGNP/AGR Fuel Program test data will be used to assess the level uncertainty associated with each potential fuel failure mode. This is an item for follow up.

With regard to calculating dose consequences for NGNP LBEs, the Project does not describe how uncertainties in the 1989 Goodin-Nabielek model particle failure rate predictions will be applied to develop (a) the best estimate particle failure rates for calculating best estimate mechanistic source terms and (b) the conservative particle failure rates for the calculating conservative mechanistic source terms. This is an item for follow up.

With respect to the coated fuel particle barrier, the working group's view is that the fuel particle failure rate model used for mechanistic source term calculations should be shown to be realistic, but sufficiently conservative, relative to the experimental data. The model should be sufficiently conservative compared to the data involving the important phenomena that affect TRISO coated particle failure during accident heat-up conditions as well as normal operating conditions. This is an item for follow up.

3.10.4 Models and data for fuel particle performance during reactivity accidents[33]

HTGR reactivity insertion accidents involving large local kernel energy deposition can result in significantly higher local fuel particle failure rates and significantly higher fission product releases than HTGR core heat-up accidents. IAEA-TECDOC-978 Section 4.3 describes the Japanese and Russian reactivity initiated accident testing and associated failure fraction results. Test conditions are described in terms of kernel energy deposition (J/g U O_2) rather than kernel or particle fuel temperature. The results indicate that the failure fraction can become significant (i.e., >10-5) when kernel energy deposition reaches about 600 J/g U O_2. At about 1000 J/g UO_2 the failure fraction can reach 0.1.

In response to a request for additional information, the Project states that HTGR reactivity insertion events typically take place over minutes, and coated fuel particle thermal time constants are a small fraction of a second. Thus, the Japanese and Russian tests are stated to not be representative of HTGR reactivity insertion events. The Project further states that the fuel temperature history (time at temperature) is the most direct indicator of challenges to fuel performance and HTGR reactivity events do not produce fuel temperature histories that approach the severity of the depressurized loss of forced convection events with regard to presenting a challenge to fuel performance. The conditions of the Japanese and Russian tests are sufficiently far removed from conditions that could occur in either the prismatic or pebble bed NGNP that these cannot be considered as simulated reactivity insertion accident tests. The Project notes that since the NGNP designs have not progressed to the point of having detailed design information and associated safety analyses, illustrative results are presented for the earlier MHTGR design taken from the MHTGR Preliminary Safety Information Document (PSID) and MHTGR PRA. The results for the limiting design basis accident (DBA) and BDBE reactivity insertion events appear in the MHTGR PSID. The rod ejection accident was considered by the designer to be an incredible event due to the MHGR design features. The results for the limiting events analyzed for the MHTGR indicate that the maximum fuel temperatures would be less than in an MHTGR core heat-up accident. The selection and analysis of the energy deposition and the maximum temperature for the most limiting reactivity insertion accidents for the NGNP are not available. As such, confirmation of the lack of a need for fuel particle failure data for NGNP reactivity insertion accidents is a follow-up item. Reactivity insertion events are further discussed in Section 3.2.2 above.

[33] Related RAIs: F33, F41/M45, B6, B22, B27, B30, B31, B32

3.10.5 Models and data for accidents with attack by oxidants[34]

Hypothesized accidents of major concern for HTGRs involve the ingression of either water or air, and consequent oxidation of graphite and graphitic fuel matrix materials. Rates of oxidant reaction with graphite and graphitic matrix materials typically obey chemical kinetics at temperatures below about 1000 °C and are mass transport limited above about 1500 °C. At temperatures between 1000 and 1500 °C, there is mixed control of the rate of reaction. In the Project's responses to RAI questions on the nature of attack of oxidants on fuel particles, a plausible argument is made that oxidants will encounter much reactive material before they reach fuel particles despite the relatively rapid diffusion of oxidants through matrix materials.

The encountered material is hypothesized to greatly deplete the oxidant available to attack fuel particles. This argument ignores the fact that chemical kinetics of graphite oxidants can be catalyzed. Among the better catalysts are alkali metals and alkaline earths – that is, cesium and strontium that may have escaped the fuel particles and produced a "halo" around the fuel particles. One can conceive of preferential reactions at these catalyst sites that create pathways for rapid mass transport of oxidant to the fuel particles. It is not evident that catalysis of graphite oxidation has been considered in the analysis of either air or water intrusion accidents. Analysis of oxidant attack on matrix material will have to include mass transport consideration. Information that will be needed include:

- porosity
- tortuosity
- Knudsen permeability parameter
- Poisseiulle permeability parameter

Evaluation of attack by oxidants is thus considered an item for follow up.

The working group notes with interest the Project's statement on planned safety tests (radionuclide release at elevated temperatures) on compacts irradiated in graphite sleeves or on irradiated spherical fuel elements at various partial pressures of oxygen over a range of temperatures. This statement was in response to an RAI on air ingress test plans for pebble and prismatic fuels. The working group will remain interested in the status of these tests, and will welcome any information the Project can provide on water ingress testing.

Effects of air on particle coating layers[35]

As described in IAEA-TECDOC-978, air ingress has the potential to significantly increase the particle failure fraction above that associated with a depressurized loss of forced cooling accident due to the effects of oxidation of the particle coating layers. This has direct bearing on the estimate of event-specific source terms.

[34] Related RAIs: F34/M38, M61
[35] Related RAIs: F34/M38, M61, M74, M75, M81, M95

The Project states in a response to an RAI on this issue that the mixture of helium and air available for ingress into the primary system following a depressurization accident is expected to be only a few percent air and that the amount of ingress depends on break aspects such as size and location. The RAI response further states that a 5 mm thickness of graphite must first be permeated before oxygen reaches the fuel particles in either a prismatic block or pebble fuel element.

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

It is the working group's view that the planned integral safety tests of irradiated NGNP fuel at various partial pressures of oxygen over a range of accident temperatures are both appropriate and necessary to provide the need particle failure rate data for modeling particle failure during air ingress events. The working group also views the experimental study of SiO_2 formation versus SiO formation as a function of temperature and oxygen partial pressure as important in providing a qualitative and quantitative understanding and confirmation of the particle degradation phenomena for the integral test results. This is an item for follow up.

Effects of moisture ingress on releases form exposed kernels[36]

NUREG/CR-6844, Vol. 1, "TRISO-Coated Particle Fuel Phenomenon Identification and Ranking Tables for Fission Product Transport Due to Manufacturing, Operations, and Accidents," states that for water ingress events, although the behavior of intact particles is much the same as for heatup events, for particles with exposed kernels (i.e., failed particles) the kernel can be oxidized releasing much of its stored fission product inventory relatively quickly. The NUREG documents states that this effect appears to be burnup dependent. The NGNP/AGR Fuel 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.

The NGNP/AGR Fuel Program Plan, dated September 30, 2010, states that a fuel heating facility will be developed to extend the chemical environment capabilities to heating to 1,600°C in oxidizing atmospheres typical of air and moisture ingress events. The subject plan states further that one capsule in the AGR-5/6 test train will contain fuel compacts with designed-to-fail particles to support post-irradiation moisture ingress testing in the fuel heating facility. Temperatures in the range of 800 to 1,300°C (corresponding to pressurized cooldown conditions) and up to 1,600°C (corresponding to depressurized conditions) may be conducted. Partial pressures of water vapor in the range of 10 to 50,000 Pa are anticipated to capture behavior across a spectrum of water leaks.

[36] Related RAIs: F11/M15, F34/M38, M61, M80, M110, M118

The working group's view is that the moisture ingress safety testing of irradiated NGNP fuel over a range of accident temperatures and partial pressures of water vapor is both appropriate and necessary to provide the needed particle data for modeling release of iodine, metallic fission products and fission gases during moisture ingress events. The conduct of these tests, the analysis of the experimental data, and the modeling of the test results are major follow-up items.

3.11 Establishment and validation of models for radionuclide transport in the primary circuit and reactor building (ST2/3(b)-(d))

Models for radionuclide transport in the primary circuit and reactor building include plateout and liftoff of radionuclides from surfaces in the primary circuit; generation, accumulation, and re-entrainment of carbonaceous dust contaminated with radionuclides; distribution, condensation, plateout, and settling of radionuclides in the reactor cavity and the other volumes of the reactor building. The effects of moisture and air ingress on radionuclide transport, and the role of helium purification system as well as venting of functional containment are other aspects of modeling of radionuclide transport in the primary circuit and reactor building.

The Project's confidence in limited radionuclide release under accident conditions is predicated on a very high level of safety performance of the TRISO coated fuel particles. This focus on the TRISO fuel has ramifications on the approach to source term modeling. A great deal of discussion is provided in the white papers on experiments and modeling radionuclide release from the fuel. However, much less discussion is given to source term model development and verification beyond the fuel such as transport in the reactor system and behavior following release from the reactor system. Indeed, a major challenge in the accident analysis of modular HTGRs is the modeling of radionuclide transport within the core and the reactor coolant system over many years of normal plant operation before initiation of an accident or transient. This is an item for follow up.

Radionuclide transport behavior in the primary circuit and reactor building[37]

In the view of the working group, the Project correctly recognizes that some fractions of condensable radionuclides, including iodine and volatile fission metals, released from the core during normal operation and during accidents, will likely deposit on structural surfaces (plateout) within the primary circuit. The Project also recognizes that currently available correlations for the deposition behavior of radionuclides have large uncertainties resulting from lack of appropriate sorption isotherms.

Radionuclides that deposit in the primary circuit during normal operation will be partially re-entrained (liftoff) from the circuit during depressurization events. The correlations for predicting radionuclide re-entrainment during depressurization transients have large uncertainties and cannot be properly validated because the historic database is not extensive for HTGRs and has large scatter.

[37] Related RAIs: F35/M39, M14, M24, M79, M98, M99, M100, M101, M102, M103, M104, M105, M106, M107, M109, M114, M116, M117

The MST and FQ white papers indicate the NGNP/AGR Fuel Program plans to perform single effects tests in an out-of-pile helium loop to characterize fission product deposition on and re-entrainment from primary system surfaces (i.e., plateout and liftoff) under normal and off-normal HTGR conditions. However, the details are missing. The working group agrees with the Project that the extent of additional in-pile and out-of-pile testing needed to establish and validate plateout and liftoff models should be further defined. The working group encourages the Project to continue its dialogue with NRC and furnish additional information on the subject when it becomes available. This will enable NRC to further assess the adequacy of the Project's test plans for resolving this issue. This is an item for follow up.

The working group further agrees with the Project that data will be needed to develop and validate the fission product transport models in the reactor building under wet and dry conditions. The Project's assertion that the LWR-centric radionuclide transport models are not generally applicable and that new technology development activities need to be defined for HTGR is noted. The working group recommends further dialogue between the Project and NRC on this topic.

Generation of carbonaceous dust during the operational life of an HTGR is an additional area for evaluation. The Project's current strategy is to use calculational tools to determine the impact of dust on the behavior of fission products in the system. The Project states that it will consider inclusion of the effect of dust in the fission product transport testing plans only if the calculations show a major impact on the transport behavior. The working group questions the Project's confidence in the analytical results when not much is known about the dust behavior, and believes the analytical effort needs to be complemented with experimental plans. This is an item for follow up.

The Project states that as NGNP design and technology development proceeds, details of the low-pressure reactor building, including venting and the extent to which filtration systems are credited and modeled in mechanistic source term calculations, will be determined. NRC review of the radionuclide transport in the reactor building, including the potential crediting and modeling of filtration systems in mechanistic source term calculations, is a follow-up item.

In response to a related RAI, the working group believes that the Project correctly recognizes that the manner in which the mechanistic source term is calculated may be affected by any future Commission policy decision on containment functional performance requirements. The working group notes that the Project considers a vented reactor building as the best choice for accident mitigation. The working group further notes the Project's argument that the vented filtered reactor building is a preferred option only in those cases of lower quality fuel and higher than expected release of plateout activity during accidents. Because fuel performance has not yet been demonstrated and plateout and dust releases during an accident have not yet been quantified, it is premature for the working group to judge the relative merit of a vented-only reactor building in contrast to vented-and-filtered reactor building. This is an item for follow up.

3.11.1 Modeling the helium purification system in calculating NGNP mechanistic source terms[38]

The Project states that the helium purification system (HPS) is credited and modeled as part of calculation of the transport and release of radionuclides during normal operation and for those LBEs in which the HPS continues to operate and proposes to classify the HPS as non-safety-related with special treatment. The HPS is not credited and modeled in the analysis of DBAs since it is not safety-related. However, for many anticipated operational occurrences (AOOs) and selected BDBEs, the HPS is expected to continue to operate and be credited in the mechanistic source term calculation. The HPS is expected to contribute to the removal of radionuclides circulating in the helium coolant, including noble gases and tritium. The contribution of the HPS to the removal of circulating activity radionuclides is expected to be large for radionuclides with a long half-life (>>4.5 hours) but is expected to have no significant effect for radionuclides with a short half-life (<<4.5 hours).

In the response to an RAI question on this subject, the Project also described how HPS performance would be modeled in removing radionuclides to control the NGNP circulating activity.

It is the working group’s view that for the analysis of LBEs, the assumed circulating activity levels (i.e., LBE initial condition) should be the maximum circulating activity allowed by the NGNP technical specifications. The working group’s view is that the Project has described a reasonable approach to performance modeling of the HPS in removing circulating radionuclides during the transient phase of those AOOs, design basis events (DBEs), and BDBEs in which the system remains in operation. The HPS safety classification is discussed, in general terms, in the NRC assessment of the NGNP white paper on SSC safety classification.

3.11.2 Modeling the reactor building vent filtration system in calculating mechanistic source terms[39]

The Project states that studies were conducted to assess design options for the reactor building and the respective advantages and disadvantages of each option. The Project states that as NGNP design and technology development proceeds, details of the low-pressure reactor building, including venting and the extent to which filtration systems are credited and modeled in mechanistic source term calculations, will be determined. NRC review of the radionuclide transport in the reactor building, including the potential crediting and modeling of filtration systems in mechanistic source term calculations, is a follow-up item.

The Project conducted an assessment (referred to as a “conceptual PIRT”) of the effects of moisture ingress on the HTGR performance in February 2011. The major phenomena and issues of high importance and requiring more attention, as noted by the Project, are:

- Characterization of graphite properties and performance under both short and long-term exposure to moisture

[38] Related RAIs: M59, M79

[39] Related RAIs: M60, M79, M82

- Investigation into the importance of the plate-out and resuspension of radionuclides in the primary coolant system
- Development of a systems accident code capable of simulating phenomena associated with moisture ingress
- Additional scoping analysis to further identify phenomena and sequences that are important to the plant performance

The working group notes that the moisture ingress “conceptual PIRT” is a good start, but believes that the resulting product does not go far enough to identify and prioritize important phenomena for fission product release and transport in the primary circuit and the containment. This is deemed by the working group to be a necessary first step in developing and validating fission product transport models that incorporate the effects of moisture ingress. This is a follow-up item.

The Project, in collaboration with NRC, also conducted an HTGR dust workshop in March 2011. A document that describes potential HTGR dust safety issues as well as research and development needs was prepared, based upon the discussions at the workshop. What is missing in the document is a substantive discussion on what to do with the findings of the workshop. Again, the working group views this as a necessary first step in developing and validating fission product transport models which incorporate the contribution of dust. This is an item for follow up.

3.12 Application of mechanistic source term models in best estimate and conservative analyses of transients and accidents (ST2/3(a)-(d))

3.12.1 Proposed uncertainty evaluation methodology

The Project’s approach for accident consequence analysis relies on the calculation of event-specific mechanistic building-release source terms and dose rates, which is based on the current understanding of radionuclide generation and transport phenomena. To compensate for uncertainties in understanding of phenomenology, the Project proposed an uncertainty evaluation methodology as follows:

- The detailed calculational tools described in Section 4.5 and in Appendices D and E of the MST white paper are used to predict the best estimate, time dependent mechanistic source term for a given LBE. These include separate computer codes for calculating the initial radionuclide inventories within the fuel and within the helium pressure boundary and for modeling the off-normal event phenomena as described in the MST white paper.
- A simplified integrated model is constructed for use in the mechanistic source term and consequence uncertainty evaluation. Best estimate values for the input parameters are utilized in this consequence uncertainty model to predict the mechanistic source terms for comparison against those obtained with the detailed calculational tools in Step 1.
- When there is confidence that the results of the simplified model of Step 2 are within reasonable convergence with the results obtained using the detailed tools, uncertainty distributions are selected for each of the independent input parameters.

- The simplified consequence uncertainty model is then run tens of thousands of times in a Monte Carlo fashion to construct the uncertainty distribution for the mechanistic source terms.

The consequence uncertainty model accounts for the release and transport of radionuclides from the fuel barriers, the helium pressure boundary, and the reactor building and finally to the atmosphere. The model treats the fuel elements, the helium pressure boundary, the reactor building and the plateout (deposition) in the reactor building as four separate volumes.

For Volume 1, the initial inventories of the key radionuclides in the fuel compacts and fuel element graphite as a result of normal operation are determined. Radionuclide release mechanisms, which are individually accounted for in the model, include:

- release by diffusion from fuel particles with intact coatings,
- release from particles with defective silicon carbide (SiC) coatings,
- release from particles with the SiC and both PyC coatings failed, referred to as exposed or bare kernels, and
- release from heavy metal contamination.

Similarly, for Volume 2, the helium pressure boundary, there is an initial inventory of radionuclides that is circulating and plated out on the primary circuit surfaces as a result of normal operation. The circulating and plateout activities are dependent on the fuel body inventory, the fraction of exposed kernels, and the heavy metal contamination fraction during normal operation. Volume 3, the reactor building, receives the radionuclides released from the helium pressure boundary. The release from the reactor building is converted to a dose by multiplying by the weather dilution factor, the breathing rate, if applicable, and the dose conversion factor of the radionuclide for each time interval.

The working group's view is that this overall approach is generally reasonable, subject to considerations noted in the following subsections.

<u>3.12.2 Comprehensiveness of proposed uncertainty models</u>[40]

The working group recognizes that the consequence uncertainty model is an important element of the proposed source term methodology. The Monte Carlo uncertainty analysis has gained increasing acceptance in the nuclear safety field due, in large part, to its ease of application. However, the analysis can have significant challenges. The difficulties commonly encountered are:

> *Definition of uncertain quantities*: Quantities sampled in an uncertainty analysis by definition are poorly known and some engineering judgment is involved in the sampling of the possible values of these quantities. Ideally, the engineering judgment should be transparent, scrutable, and built upon considerable experience and expertise of the analyst. Similar difficulties arise when parameters peculiar to a code or model and intended to account in some unspecified way for phenomena not modeled in the code are selected for

[40] Related RAI: M88

sampling. The only expertise on possible values is usually the experience of the code developer in this case.

Definition of uncertainty ranges: Definition of the range of values to be sampled is often quite difficult simply because there is so little data. Data that do exist may not be sufficiently "prototypic." Thus, a rigorous definition of the range of values to be sampled can easily be the hardest part of any uncertainty analysis.

Correlations among uncertain quantities: The assertion that sampled quantities are independent must be justified. More subtle correlations can exist and must either be addressed or their neglect must be justified. For example, it is not at all obvious that "fuel inventory", "circulating inventory", and "plateout inventory" (as tabulated in response to a related RAI question) can be independently sampled in a Monte Carlo uncertainty analysis.

The working group notes that the Monte Carlo uncertainty analysis proposed by the Project appears to address only parametric uncertainty. The regulatory community recognizes also "model uncertainty" and "completeness uncertainty". There is, of course, no practical way to quantify completeness uncertainty ("unknown unknowns"). There is, however, a growing trend of asking at least for some assessment of model uncertainty if not rigorous quantification of this uncertainty. This is an item for follow up.

3.12.3 Context-specific uses of the terms "best estimate" and "conservative" [41]

The Project's "best estimate" calculations are described in the MST white paper as employing several conservative approximations and assumptions. The working group notes that this use of the term "best estimate" is potentially misleading in that the calculations in question would in fact yield dose consequence predictions that could be correctly described as conservative or pessimistic.

The working group acknowledges that the existence of conservatisms in so-called "best estimate" source term calculations remains merely an issue of semantics as long as the sole purpose of such calculations is to show that "best estimate" accident dose consequences are below a certain compliance or response threshold. However, when "best estimate" source term calculations are used (as the term implies) to provide realistic predictions of expected dose consequences, it may become necessary to replace the conservatisms noted in the white paper with realism.

In response to an RAI question, the Project confirmed that the proposed mechanistic source term calculations are intended for use in essentially all contexts, including emergency planning and all applications of risk assessment. The working group notes that realistic or non-biased source term predictions may be most appropriate in certain contexts of emergency planning and risk assessment. This is thus an item for follow up.

In general, the working group notes that discussions of "best estimate" and "conservative" analyses would benefit from maintaining clear distinctions between modeling assumptions and modeling approximations and their respective applications to (a) defining or modeling the events

[41] Related RAIs: M4, M17

themselves and (b) modeling the phenomenology of event progression and event consequences. For example, the analysis of a given event sequence may make use of pessimistic or worst case “conservative” assumptions about the event sequence itself in terms of system parameters and configurations (e.g., operating state, break timing, break size, break location, equipment failures, etc.) in conjunction with “best-estimate” phenomenological models that employ non-biased (i.e., realistic) approximations of physical phenomena in simulating the progression and consequences of the event sequence.

3.12.4 Analyzing mechanistic source terms for specific LBE categories[42]

As discussed in the NRC assessment of the LBE assessment white paper regarding its Outcome Objective 4: Acceptable limits on the event sequence consequences and the analysis basis for the LBE categories, the associated NRC working group’s continuing views are as follows:

- For AOOs, the dose calculation should realistically model all the systems, structures, and components (SSCs) modeled in the deterministic safety analysis of the AOO event sequence, but a conservative calculation of the mechanistic source term should be used to demonstrate that 10 CFR 20 dose limits are met. The working group believes this is consistent with SECY-03-0047 Issue 5, in which the staff recommended that a conservative event specific mechanistic source term calculation be used for AOOs. In the SRM for the SECY-03-0047, the Commission approved the staff’s recommendations related to this issue.
- For DBEs and DBAs, the working group believes that the proposal to conservatively calculate the mechanistic source terms and dose consequences for DBEs to demonstrate compliance with 10 CFR 50.34 is consistent with SECY-03-0047 Issue 5, in which the staff recommended that a conservative event specific mechanistic source term calculation be used for DBEs. In the SRM for the SECY-03-0047, the Commission approved the staff’s recommendations related to this issue. This approach is also consistent with Table 6-3 in NUREG-1860.
- For BDBEs, the dose calculation should realistically model all the SSCs modeled in the deterministic safety analysis of the BDBE event sequence, and a best estimate calculation of the mechanistic source term should be used to demonstrate that the BDBE dose limits are met. However, in SECY-03-0047, the staff recommended that a conservative source term be used for the purpose of siting and containment decisions. The staff further stated that the 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. In the SRM for the SECY-03-0047, the Commission approved the staff’s recommendations related to this issue.

However, to bound severe accidents, it is the working group’s view that events ranging in frequency from 10^{-5} to10^{-8} per reactor-year should also be considered for the purpose of siting and containment system design decisions. Events in that frequency range are defined by the Project as BDBEs. Where events in the frequency range of 10^{-5} to10^{-8} per reactor-year are considered for the purpose of siting and containment decisions (i.e., to ensure defense-in-depth is provided by the containment system design), a conservative analysis may thus be required.

[42] Related RAIs: M65, M66, M67, M68, M71

The working group believes that a Commission policy decision may be needed to support a final determination on how events in that frequency range will be considered for the purpose of siting and containment system design decisions (i.e., containment system design defense-in-depth). This is an item for follow up.

The Project proposes to use Monte Carlo methods to determine the overall effect of uncertainties on source terms (including the fuel failure fractions and fuel radionuclide releases) and off-site consequences and then use the resulting consequence distributions to provide a basis for judging acceptability and safety margins for a range of requirements. The Project therefore proposes that the model for failure probability of the NGNP's most important barrier to fission product release (i.e., the coated fuel particles) be modeled on a statistical basis to account for uncertainties about a mean in the particle failure probability. The working group believes this approach is generally consistent with SECY-03-0047 Issue 5, in which the staff recommended that the calculations should be as realistic as possible so that the values and limitations of any mechanism or barrier are not obscured. The use of realistic, but adequately conservative, models of radionuclide release from TRISO coated fuel particles for predicting event-specific mechanistic source terms is discussed in Section 3.10.

3.12.5 Peer review of NGNP mechanistic source terms[43]

The NRC assessment of the NGNP LBE selection white paper provides additional preliminary NRC views on the selection of LBEs and the calculation of the event-specific mechanistic source term for the events in each LBE category. Included are views on the potential need for peer review of the NGNP approach to mechanistic source terms, as described in the following paragraph.

In the MST white paper the Project states that, at present, there are no plans for conducting a peer review of NGNP mechanistic source terms analogous to the peer review conducted for the LWR Alternate Source Term. However, the NGNP white paper on probabilistic risk assessment (PRA) makes reference to the American Society of Mechanical Engineers (ASME) and American Nuclear Society (ANS) standard, "Technology Neutral Probabilistic Risk Assessment Standard for Advanced Non-LWR Nuclear Power Plants," dated July 2011. The PRA white paper states that it is expected that a trial use version of that ASME/ANS PRA standard will be approved in advance of the completion of the development of the COLA. The draft ASME/ANS standard is presently being prepared for ballot. The reference draft ASME/ANS PRA standard states that it is required that all PRA elements (including the mechanistic source term element) have a peer review. The need for a peer review of the NGNP mechanistic source terms is thus considered an item for follow up.

3.13 Establishment and implementation of NGNP prototype pre-operational and operational programs to verify and supplement the developmental technical bases for fuel qualification and mechanistic source terms[44] (FQ2) (ST2/3(a)-(d))

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[43] Related RAIs: M70, M72

[44] Related RAIs: F1/M1, F3/M3, F4/M5, F5/M6, F6/M7, F7/M8, F10/M13, M12, F13/M18, F14/M19, F15/M20, F16/M21, F22, F23/M28, B5, B29, B47, B49, B76, B80

3.13.1 Recommended use of prototype provisions to facilitate NGNP prototype licensing

The working group notes that licensing of an NGNP prototype was specified in the EPAct and reaffirmed in the DOE/NRC NGNP Licensing Strategy Report to Congress (2008). Relevant prototype licensing provisions are provided in 10 CFR 50.43(e) and 52.78(a)(24) and mentioned in various Standard Review Plan (SRP) sections. SRP Section 4.2, for example, refers to prototype testing in stating that for a fuel design that introduces new features the applicant should describe a more detailed surveillance program commensurate with the nature of the changes.

Consistent with the DOE/NRC NGNP Licensing Strategy Report, the working group believes that the Project should employ prototype-specific plant design features and surveillance programs to facilitate effective resolution of technical issues for licensing. Viewed in conjunction with associated license conditions, technical specifications, and other regulatory controls, the primary purpose of such prototype-specific design features and programs would be to verify that initial and evolving NGNP operating conditions and performance elements (e.g., fuel performance) are consistent with those predicted and considered as the technical bases for licensing. Another purpose would be to supplement the technical bases for design, licensing, operations, and oversight.

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. Of particular concern is the potential for either inaccurately predicted normal conditions or undetected operating condition anomalies 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. This is an item for follow up.

3.13.2 Using prototype provisions to verify and supplement the developmental technical bases for fuel qualification and mechanistic source terms

The subject white papers seek NRC agreement that the presented technical approaches to fuel qualification and source term analysis and validation are acceptable. The working group's view is that the merits of these approaches and their implementation cannot be conclusively judged without considering the extent to which the resulting developmental technical bases will be verified and supplemented by prototype tests, surveillance, monitoring, and inspections to be performed in the NGNP prototype.

The Project should specifically address how design features, testing, and surveillance programs specific to the NGNP prototype will be used to verify and supplement the developmental technical bases now being established for NGNP fuel qualification and mechanistic source terms. Such prototype-specific programs would entail the conduct of pre-operational, startup, and operational tests, operational monitoring and surveillance, and periodic confirmatory measurements and inspections. This is a follow-up item.

3.13.3 Challenges and needs for verifying normal fuel operating conditions in HTGR cores

This topic has particular ramifications for NGNP in view of two essential attributes of HTGR technology:

- Accident source terms for modular HTGR designs are sensitive to normal core operating conditions.
- Inherent technical challenges make normal operating conditions in HTGR cores both difficult to measure and difficult to reliably predict.

Early accident releases can include significant contributions from long-lived metallic fission products (e.g., Cs-137, 30-year half-life) that accumulate in the primary system over decades of normal operation. Elevated normal fuel operating temperatures generally increase the diffusive release of Cs during normal operation and can weaken fuel particle coatings (e.g., due to Pd attack) during normal operations and in accidents.

For both pebble-bed and prismatic-block HTGRs, the ability to perform in-core 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. Real-time measurements of in-core peak operating conditions have therefore never been performed in any of the HTGRs operated to date. Interpretation of limited on-line measurements of coolant outlet temperature profiles outside the core, as well as post-irradiation examination of in-core meltwire probes, nevertheless suggests that core regions in past HTGRs operated at temperatures significantly higher than predicted.

Several factors contribute to the difficulty of predicting normal operating conditions in prismatic-block and pebble-bed HTGR cores. For one, the viscosity of gases like helium, unlike liquids, increases with temperature. This and the fact that the primary coolant flows downward in HTGR cores means that both viscosity and thermal buoyancy inherently act to reduce coolant flow to the hotter core regions where it is most needed during normal operations. These factors thus contribute to the development of helium bypass flows within and around the core and the evolution of operating hot spots[45] in core regions with higher fission power densities and/or more restricted coolant flow paths.

Additional factors affecting core operating conditions in prismatic-block HTGRs include their potential vulnerability to local “closed-lattice core” undercooling effects (e.g., from coolant hole obstruction or hole misalignment caused by block warping, shifting, or fracture) as well as their reliance on engineered power shaping achieved through fuel block shuffling and complex zoning of fuel and burnable poison. It further bears noting that prismatic-block cores will generally keep fuel in potential hot spots for many months at time.

The ability to reliably predict power shapes in HTGR cores faces particular challenges associated with:

- highly variable and uncertain local moderator temperatures,

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

- 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 flux mapping detectors.

Factors affecting pebble-bed HTGRs include the potential for reduced coolant flow in core locations with tighter random pebble packings. Pebble-bed HTGRs may be further affected by the potential for power shape aberrations associated with pebble flow profile uncertainties and the potentially destabilizing effects on pebble flow profiles caused by the strong temperature dependence of pebble-to-pebble friction in helium and by obstructions to local pebble flow caused by the debris resulting from occasional pebble breakage.[46, 47] In cases of pebble debris and locally obstructed pebble flow, the affected fuel populations may experience greatly extended core residence times that lead to excessive levels of fuel burnup and fluence.

Finally, core bypass flows directly affect normal core operating temperatures in both pebble-bed and prismatic block HTGRs. While such bypass flows cannot be directly measured, operating evidence suggests that they were underpredicted in past HTGRs. For example, predicted and actual core bypass flows in the THTR pebble bed reactor were reported as 7% and 18%, respectively.[48] It bears noting that the core bypass flow of helium through the gap openings between reflector blocks generally increases with operating time due to the irradiation-induced shrinkage of graphite. Core bypass flows and pebble flow velocity profile aberrations have been cited as major factors leading to higher than predicted peak core operating temperatures in the AVR and THTR pebble bed reactors.[49, 50]

The Project should develop approaches and plans for performing in-core measurements in the NGNP prototype to verify normal core operating conditions and demonstrate the adequate detection of operating condition anomalies. This is an item for follow up.

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[46] 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).

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

[48] 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).

[49] 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.

[50] 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.

3.13.4 Prototype testing and surveillance to verify and supplement the development technical bases for NGNP fuel service conditions and fuel performance[51]

The working group requested additional information on (1) how the regulatory requirements for technical specifications will be applied to the NGNP fuel design and (2) whether the technical specifications for NGNP 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)

In response the Project stated that a comprehensive set of technical specifications will be proposed by the license applicant for the NGNP to assure that safety-related systems, structures, and components meet design requirements throughout their service lifetimes. The Project stated that the technical specifications for FSV were examined for indications of the kinds of technical specifications that generally might be included in those for NGNP and many were relatively generic. These included limiting conditions for operation (LCOs) on primary and secondary coolant activity, and surveillances related to the plateout probe, primary reactor coolant radioactivity and secondary coolant activity. The technical approach to the FSV reactor core safety limit was complex and difficult to evaluate and is not considered by the working group to be applicable to the NGNP. It is the working group's view that the NGNP should have technical specification LCOs and surveillances that are generally similar to those for FSV. The working group's view is that an appropriate fuel or core-wide safety limit should be developed and included in the NGNP technical specifications and that the safety limit should be applicable to and be met for NGNP AOOs.

The Project also anticipates that the first-of-a-kind NGNP design, i.e., the NGNP prototype per the EPAct, would include special instrumentation systems to monitor fuel performance to ensure that it is consistent with the safety analysis. These would include:

- Ion chambers to continuously measure total gamma and beta activity in the primary coolant
- A sampling and analysis system to measure noble gas release-to-birth rate ratios
- Plateout probes to measure core release rates of condensable radionuclides such as I-131, Cs-137, and Sr-90
- Sampling stations and instrumentation to determine an overall mass balance for tritium
- Gamma scanning equipment to measure plateout activity on primary system surfaces

It is the working group's view that the above types of instrumentation should be included in the NGNP prototype, along with additional instrumentation as may be needed to address considerations discussed in the following paragraphs.

In response to specific RAI questions, the Project provided (a) general information on its limited university-based research efforts to date toward developing advanced in-core detector systems

[51] Related RAIs: F3/M3, F6/M7, F23/M28, F26/M31, F52/M56, M59, M60, M82, M102, B55

for HTGRs and (b) preliminary overview information on some of the types of surveillance and testing programs that the Project would envision for the NGNP prototype. The latter Project RAI responses provided a preliminary, high-level overview of envisioned startup testing programs, demonstration testing programs, and operational surveillance programs, all of which the working group would generally consider helpful or necessary, depending on NGNP prototype details.

Noted below are the working group's views on some additional areas where needs and opportunities may be found for conducting special operational surveillance and measurement programs in the NGNP prototype:

(a) As noted in Subsection 3.7.1, periodic PIE and accident heatup testing on fuel discharged from the NGNP prototype may be needed to supplement the developmental technical bases for fuel qualification and verify adequate fuel performance under actual HTGR operating conditions. 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., irradiation times and 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 ATR-based accelerated test irradiations used for developmental fuel qualification. Regarding the latter, it is noted that, based on information in TEV-1022, a technical report prepared by the Project in response to related NRC questions, one can conclude that the peak palladium time-at-temperature calculated for the AGR-1 accelerated fuel irradiation test is less than half the maximum value calculated for fuel irradiated in a representative prismatic-block NGNP core design.

(b) Specific measurements will likely be needed to confirm predicted core operating temperature and power profiles and fuel operating performance and to detect plausible core irregularities such as local core hot spots, fuel misloadings, pebble flow anomalies, block-stack motions, etc. Absent major advances in the development of in-core detector systems for HTGRs, core monitoring and confirmation may have to place significant reliance on near-core and ex-vessel detectors, PIE of discharged fuel, PIE of in-core melt-wire probes, PIE of in-core activation probes, and measurements of circulating, plateout, and dust activity.

The working group's view is that, going forward, a clearer understanding should be established regarding the full gamut of testing, monitoring, and surveillance programs and associated instrumentation systems envisioned for the NGNP prototype. Included would be a shared understanding of how such programs could be used to facilitate effective resolution of technical issues both generally and in the context of prototype licensing provisions. This would call for, among other things, information on any advanced in-core detectors to be developed and deployed, and, more generally, information on how measurement data will be calibrated and used to (a) address technical specifications and (b) verify and supplement the developmental technical bases for NGNP fuel qualification and mechanistic source terms. This overall topic is considered a significant follow-up item.

4. CONCLUDING REMARKS

The preceding sections have presented the NRC working group's detailed assessment comments in response to the Project's requests for feedback on the technical approaches presented it the NGNP white papers on fuel qualification and mechanistic source terms. As stated above in Section 3.1, the working group's overall preliminary assessment is that the proposed high-level approaches to NGNP fuel qualification and mechanistic source terms are generally reasonable, albeit with several potentially significant caveats. Subject to further consideration and resolution of the details and issues noted herein, the working group has identified no fundamental shortcomings that would necessarily preclude successful implementation of the presented high-level approaches towards developing the technical bases for related NGNP prototype licensing submittals.

It bears reiterating here that, consistent with the nature of the white papers, this NRC assessment feedback does not provide final staff positions or regulatory conclusions on any aspect of the NGNP design or technical safety basis. Such conclusions would be provided in the NRC staff's safety evaluations of future NGNP licensing submittals to determine whether or not the proposed NGNP design complies with NRC regulations. Completion of the NGNP prototype design and its developmental safety basis in accordance with this assessment feedback will not be sufficient justification for the design unless compliance with NRC regulations is also demonstrated.

The working group's assessment comments are intended to facilitate continuing efforts towards achieving effective resolution of technical and policy issues for licensing the NGNP prototype. Many of the issues identified in this assessment can be addressed through the Project's ongoing and planned efforts. However, as noted in the assessment comments on several feedback topics and more broadly discussed in Section 3.13, it appears that many of the more challenging issues and uncertainties concerning fuel performance and source terms could be most effectively resolved through the use of prototype licensing provisions in conjunction with prototype-specific design features and special programs of operational surveillance, monitoring, testing, and inspection in the NGNP prototype.

The working group further believes that detailed consideration of such prototype provisions and programs could be beneficial to the Project in the near term. This view is based in part on noting that the anticipated scope and nature of such provisions and programs would seem to be largely generic to all modular HTGR design variants and, thus, largely insensitive to NGNP design details yet to be established. Among the potential benefits that may result from bringing focused attention to this area in the near term would be the extra time afforded to develop and qualify advanced sensor and surveillance systems for HTGR service conditions.

APPENDIX A

Participants in the NRC Working Group for Assessing the NGNP White Papers on Fuel Qualification and Mechanistic Source Terms

Listed alphabetically below are the NRC working group participants who contributed to assessing the NGNP white papers on fuel qualification and mechanistic source terms. Principal contributors are designated with an asterisk (*).

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APPENDIX B

NGNP Project Requests for NRC Feedback on Fuel Qualification and Mechanistic Source Terms

The NGNP Project has requested that the NRC provide feedback on the adequacy of its planned approaches to fuel qualification (FQ) and mechanistic source terms (MST) as the bases for future NGNP licensing submittals in these areas. The Project presented specific requests for feedback within its respective FQ and MST white papers and updated these requests in a letter dated May 3, 2011 (ADAMS accession number ML111250375).

The Project's initial requests for NRC feedback appear in Section 6 of the respective white papers in terms of stated "outcome objectives." These requests are paraphrased and numbered below for reference.

<u>Fuel Qualification – Stated Outcome Objectives</u>

The primary issues for which NRC feedback is requested include:

FQ1. Plans established in Section 5 for qualification of the UO_2 pebble fuel type are generally acceptable. These plans call for (a) utilizing German data for normal operation irradiation, and transient/accident heat-up conditions, and (b) performing additional confirmatory irradiation and safety tests on fuel manufactured at a qualified facility to statistically strengthen the performance database and demonstrate that the fuel performs at least as well as the German fuel upon with the UO_2 pebble fuel design is based.

FQ2. Plans established in Section 5 for qualification of the uranium oxycarbide (UCO) prismatic fuel type are generally acceptable based on the Advanced Gas Reactor (AGR) Fuel Development and Qualification Program.

Other activities and information may be necessary to support the qualification of both pebble-bed UO_2 and prismatic UCO fuels. Therefore, it is requested that the NRC either:

i. Confirm that the plans presented in this paper are generally acceptable, or

ii. Identify any additional information or testing needed to demonstrate adequate NGNP fuel performance.

<u>Mechanistic Source Terms – Stated Outcome Objectives</u>

Issues for Resolution:

It is requested that NRC either confirm that the plans for addressing the respective issues summarized below are generally acceptable, or identify additional information needs of the NRC or any areas in which the NRC believes that plans will not be sufficient to address applicable regulatory requirements and guidance:

ST1. Agreement that the definition of event specific mechanistic source terms for the HTGR is acceptable.

ST2. Agreement that the approach to calculate event specific mechanistic source terms for the HTGR technology is acceptable, subject to validation of the design methods and supporting data that form the bases of the calculations.

Specifically, this approach analyzes a functional containment comprising several barriers that limit the release of radionuclides to the environment (defined herein as the source term) for each postulated event, including normal operating conditions, abnormal operating conditions and accident conditions. The multiple barriers include individual fuel particle kernels and coatings, the fuel matrix and fuel element graphite, the helium pressure boundary (primary circuit), and a vented low-pressure reactor building. Design methods for determining radionuclide source terms, which include analytical tools used to calculate the performance of each of these barriers during radionuclide transport under event-specific conditions, are defined and supported by testing and analysis. These analytical tools are applied in calculations for normal operating conditions, abnormal operating conditions, DBA conditions, and BDBA conditions:

(a) Generation and transport of each radiologically significant species of fission product from the fuel kernel, through the TRISO particle coatings and fuel element graphite and into the reactor coolant as a function of as-manufactured quality of the TRISO fuel coatings (including heavy metal contamination) and postulated in-service and accident condition coating failure rates as a function of fuel burnup, power level, temperature (including time at temperature), and, where applicable, air and water contamination.

(b) The concentration and form of each radiologically significant species of radionuclide in the primary circuit (those released from the fuel elements) under steady-state full power and temperature operating conditions, including circulating activity and plateout of condensable radionuclides on primary circuit components; the effects of dust generation, fallout, and radionuclide absorption; radionuclide half-life; and operation of the helium purification system.

(c) The concentration and form of each radiologically significant species of radionuclide in helium released from the helium pressure boundary under depressurization events as a function of time considering the location and time-dependent rate of coolant release, reentrainment of accumulated dust, liftoff of plated-out radionuclides, and the effects of time-dependent air and/or moisture ingress on these parameters.

(d) The effects of radionuclide form, condensation, settling, vent-path configuration, and vent filtering, if any, on the time-dependent calculation of radionuclide transport through the reactor building and the source term release to the atmosphere for each event.

ST3. Agreement on the acceptability of the approach of the planned fission product transport tests of NGNP/ AGR Fuel Development and Qualification Program, as supplemented by the existing irradiation and post-irradiation heating data bases, to validate these fission product transport analytical tools.

In addition, the evolving nature of the Project's plans and requested NRC feedback for NGNP pebble fuel was noted in the introduction section of the FQ white paper as follows:

Pebble-bed Reactor – The qualification of UO_2 fuel particles is based on a combination of existing German low-enriched uranium (LEU) UO_2 test data and additional testing of fuel replicating the German design and fabrication process. The program for additional testing discussed in this paper was developed primarily to support a demonstration power plant to be constructed in South Africa. That project was recently cancelled, and the pebble-bed testing program may undergo significant changes. As revised testing plans are developed in the near future they will be described and discussed in the course of revising this paper.

The beginning of Section 5.2 of the FQ white paper further noted that:

The pebble-bed program in South Africa has been substantially altered during the production of this paper and may undergo additional changes. The material presented here does not reflect these recent changes and can be expected to be significantly revised in the course of discussions with the NRC staff.

Accordingly, in its letter of May 3, 2011, the Project provided the following updates:

"Following submittal of the white papers, the strategy for fuel acquisition for the NGNP (Ref. 3)[52] was revisited in light of the major change in fabrication options for pebble fuel. The updated strategy does not involve replication of German fuel, the basis for the PBMR (Pty) Ltd. approach, as described in Section 5.2.1 of the Fuel Qualification White Paper on fabrication and process control.

At present, the NGNP Advanced Gas Reactor (AGR) Fuel Development and Qualification Program (Ref. 4)[53] is focused on testing of LEU UCO TRISO fuel particles in compacts such as those used in prismatic HTGRs. However, the near term activities have been adjusted to incorporate scope supporting pebble fuel particles. Specifically, LEU UO_2 TRISO fuel particles generally consistent with the German particle design and produced by Babcock and Wilcox, AREVA and PBMR, (Pty) Ltd. are currently under irradiation in compacts in the AGR-2 test train in the Advanced Test Reactor (ATR) at Idaho National Laboratory.

Building on the ATR irradiations that are currently underway, updated information regarding the revised plan for pebble bed fuel qualification will be provided once that plan is established and those additional details are available. It is expected that the scope and objectives of the revised pebble bed fuel plan will build upon the existing plan (Ref. 4) and be adjusted for pebble bed fuel specific design and service. This would include irradiation and testing of sufficient quantities of fuel to demonstrate that statistical fuel performance requirements (particle failure fractions) are met without relying on the use of historical German data.

With regard to support of mechanistic source terms, a broad set of international experimental results on fission product transport in coated particle fuel has been produced, exchanged, and subjected to international review over several decades. A primary example of data exchange and review is a document produced by the International Atomic Energy Agency (Ref. 5).[54] In general there is considerable overlap in data, allowing comparison of results

[52] [3] D. Petti, et al., INL/EXT-07-12441, Rev. 2, "Updated NGNP Fuel Acquisition Strategy," December 2010

[53] [4] INL/PLN-3636, "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," September 2010

[54] [5] IAEA-TECDOC-978, "Fuel Performance and Fission Product Behavior in Gas-Cooled Reactors," November 1997

from parallel tests. The effort required to reproduce this broad set of data would be prohibitive and the data set is considered, by virtue of its extensive international exchange and review, to be sufficiently qualified for use in model development. Fission product transport models used and planned to be used by the NGNP project for source term predictions have been developed with consideration of this international database, including German data, for both the prismatic and pebble designs. The NGNP fuel development and qualification program incorporates testing to generate additional data for the prismatic fuel form for use in model development and validation of fission product transport codes. As noted above, it is expected that a program of comparable scope and objectives would be conducted for a pebble fuel design.

Therefore, the material in Section 5.2 of the Fuel Qualification White Paper should be withheld from review. In addition, the objectives in Section 1.3 and in Section 6 of the Fuel Qualification White Paper related to qualification of pebble fuel based on the PBMR, (Pty) Ltd. approach should be withheld from review. The NGNP Project plans to update both the Fuel Qualification and Mechanistic Source Terms white papers once the pending NRC requests for additional information (RAIs) are satisfactorily addressed."

The NRC working group adjusted its subsequent assessment efforts in accordance with the above Project updates. The assessment feedback provided in the body of this document thus addresses neither Objective FQ1 for pebble fuel qualification nor the directly related aspects of Objectives ST2 and ST3 for mechanistic source terms. The working group nevertheless included its previously developed RAI questions and comments specific to pebble fuel in the RAI sets that it subsequently submitted to the Project. This was done in recognition of the full or partial relevance that those questions and comments may be found to hold to the Project's changing plans for pebble fuel.