OFFICE OF NEW REACTORS

ASSESSMENT OF WHITE PAPER SUBMITTALS ON FUEL QUALIFICATION AND MECHANISTIC SOURCE TERMS (REVISION 1)

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) and its Idaho National Laboratory (INL) as required by Congress in Title VI, Subtitle C, of the Energy Policy Act of 2005 (EPAct). The mission of DOE/INL's NGNP 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, DOE/INL is considering a modular HTGR with either a prismatic block or pebble bed core and safety features described by DOE/INL 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.
>
> 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. 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, DOE/INL and the U.S. Nuclear Regulatory Commission (NRC) have been engaged in prelicensing interactions on technical and policy issues that could affect the

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

design and licensing of an NGNP prototype. The Commission encourages early interactions as stated in its Policy Statement on the Regulation of the Advanced Reactors:

> "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."

The DOE/INL has prepared a series of white papers on aspects of the HTGR design and safety basis to obtain NRC feedback on design, safety, technical, and/or licensing process issues that could affect NGNP deployment.

On July 21, 2010, DOE/INL submitted the two interrelated white papers that are the subject of this assessment:

INL/EXT-10-17686, "NGNP Fuel Qualification White Paper" (ADAMS accession number ML102040261,[2] referred to herein as the FQ white paper)

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

The cover letter for these submittals briefly states that these white papers summarize the planned approaches to fuel qualification and the development of mechanistic source terms. The letter states that the approaches 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 that will be submitted for licensing the NGNP prototype. The purpose of this assessment report is to provide the requested NRC feedback.

2. ASSESSMENT PROCESS

Development of the requested NRC feedback required the conduct and documentation of an assessment process in two phases, as described respectively in the two subsections below. Submittals, correspondence, meeting materials, and meeting summaries pertinent to the assessment process and other NGNP prelicensing activities are available in ADAMS under Docket No. PROJ0748.

DOE/INL did not submit revisions to the white papers during the assessment process; however, it indicated that any future NGNP prelicensing or licensing submittals related to topics in the white papers would incorporate revisions and clarifications based on comments from the NRC assessment.

[2] Note: Subsequent ADAMS accession number references omit "ADAMS accession number" for brevity.

2.1 Initial Assessment Phase

For the initial phase of the assessment process, the NRC assembled an assessment working group comprising several personnel from the NRC Office of New Reactors and Office of Nuclear Regulatory Research. Personnel from Brookhaven National Laboratory and Sandia National Laboratories assisted the NRC staff working group. Appendix A of this report lists the NRC staff participants and national laboratory participants.

The NRC working group began the initial assessment process by discussing the white papers with DOE/INL at a public meeting held September 1-2, 2010 (ML102590247, ML102700497). Routine biweekly conference calls between NRC and DOE/INL facilitated continuing coordination of all interactions related to NGNP, including those for assessing the subject white papers. By letter dated May 3, 2011 (ML111250375), DOE/INL 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 DOE/INL's request, the NRC staff did not assess those portions of the white paper that relate to DOE/INL's changing plans for pebble fuel, as further noted in Section 3 and Appendix B to this report.

Over the course of the initial assessment process, the NRC working group issued two sets of requests for additional information (RAI) related to the content and stated feedback objectives of the subject white papers. The working group issued the first set of 118 RAI questions on June 7, 2011 (ML111530271), and issued the second set of 82 RAI questions on July 25, 2011 (ML112030135). DOE/INL responded to the two sets of RAIs on August 10, 2011 (ML11224A060), and September 21, 2011 (ML11266A133), respectively. On October 19, 2011, the NRC working group held a public meeting with DOE/INL to clarify and discuss selected RAI responses that were deemed particularly important toward completing the initial assessment.

This assessment phase culminated with the issuance of an initial version of this NRC assessment report (ML120240671, and ML120240699) and an associated NRC letter to DOE on February 15, 2012 (ML120240682). The letter to DOE included a brief discussion pursuant to DOE's request for continued preapplication interactions on fuel, source terms, and other topics as they relate to four key licensing issues highlighted in the NGNP Licensing Strategy Report to Congress (ML082290017).

2.2 Follow-On Assessment Phase

The follow-on assessment phase was conducted through a series of public working meetings and conference calls with DOE/INL and included reviews of additional DOE/INL submittals provided to address issues and follow up items identified in the initial version of this NRC assessment report. As in the initial assessment phase, continuing assessment interactions were facilitated and coordinated through routine conference calls between NRC and DOE/INL. These routine calls were generally conducted on a biweekly basis, shifting to a weekly basis during the closing weeks of the process.

In a letter dated July 6, 2012 (ML121910310), DOE/INL clarified its overall objectives with respect to the four key issue areas acknowledged in NRC's February 15, 2012, letter to DOE. These four key issue areas are:

(1) Containment functional performance
(2) Licensing basis event selection
(3) Source terms
(4) Emergency preparedness

DOE/INL's July 6, 2012, letter provided a useful framework for coordinating and integrating the continuing assessment interactions for fuel qualification and mechanistic source terms with those for the three interrelated NGNP white papers on defense-in-depth approach, licensing basis event selection, and safety classification of systems, structures, and components.

The staff noted during the assessment process that the word "acceptable" as used in DOE/INL'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 has instead assessed the proposed approaches in terms of whether they are reasonable, thereby effectively replacing "acceptable" with "reasonable" in DOE/INL's feedback requests.

As indicated in Appendix A, participants in the follow-on assessment phase included additional staff from appropriate NRC program offices. This updated white paper assessment report presents NRC staff views that revise, clarify, and supplement the NRC working group views presented in the initial report.

3. ASSESSMENT RESULTS

This section of the assessment report presents the feedback developed by the NRC staff in assessing DOE/INL's white papers on NGNP fuel qualification and mechanistic source terms. Section 3.1 presents a high-level overview of the proposed approaches and how they were assessed 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 license application submittal. The NRC licensing 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. [3]

The assessment feedback on these white papers is preliminary because many issues identified by the staff cannot be addressed or resolved until more information about the NGNP design and the results of planned or necessary supporting safety research and development are available. Nonetheless, the staff believes that identifying these issues is beneficial to DOE/INL for its consideration of relevant insights in further developing the NGNP design and its safety basis.

The specific technical issues for which DOE/INL has requested NRC assessment feedback are presented in Section 6 of the respective white papers in terms of stated outcome objectives. As

[3] The term "Commission," as used in this document, refers to the five appointed NRC Commissioners, whereas the term "staff" refers to NRC career staff.

noted above, DOE/INL issued a letter dated May 3, 2011, retracting those stated outcome objectives that relate to its changing plans for pebble fuel (ML111250375). For convenience, and to facilitate referencing in the feedback discussions that follow, Appendix B of this report consolidates and enumerates DOE/INL'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 DOE/INL's stated outcome objectives as labeled in Appendix B.[4] As reflected in Appendix B, and consistent with DOE/INL's letter of May 3, 2011, the assessment feedback provided in this report 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 staff 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 DOE/INL's changing plans for pebble bed reactor fuel. Accordingly, DOE/INL's RAI response submittals simply acknowledged such pebble-specific RAI questions with a brief reply citing DOE/INL'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 available in related RAIs in the form of individual RAI questions and comments developed by the staff and associated responses provided by DOE/INL. Each topic is thus footnoted with a list of related RAIs.[5] Some RAIs are listed under one or multiple 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 DOE/INL had withdrawn from further review. The staff generally understands that DOE/INL 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 staff's overall assessment is that the proposed high-level approaches to NGNP fuel qualification and mechanistic source terms are generally reasonable, albeit with several

[4] 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.

[5] 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.

(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 report 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.

potentially significant caveats. This means that, subject to further consideration and resolution of details and issues noted subsequently in this assessment report, the staff'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, given the information provided by DOE/INL. A key issue to be addressed for NGNP fuel qualification concerns the need to supplement DOE/INL's planned fuel testing program with real-time fuel test irradiations in an HTGR-like neutron environment. This issue and others are discussed in subsequent subsections of this report.

In addressing DOE/INL's requests for feedback, the assessment comments provided herein are intended to facilitate continuing efforts by DOE/INL 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 by DOE/INL 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 NRC staff'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 DOE/INL's ongoing NGNP Advanced Gas Reactor (AGR) Fuel Development and Qualification Program[6] (hereafter 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 the 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

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

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 writing this assessment report, the AGR-1 irradiation tests of preliminary fuel designs fabricated with developmental equipment, processes, and controls had been completed, the AGR-1 post-irradiation examinations and safety tests were in progress, and the AGR-2 and AGR-3/4 irradiation tests were underway.

3.1.1.1 Proposed Approach to Fuel Development and Qualification

DOE/INL'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. It bears noting that the Chinese program imported and re-used the same fuel fabrication line that had been used in the German program. 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.

A central strategy of DOE/INL'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. DOE/INL 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 involves 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. Both unheated and heat-tested irradiated fuel samples undergo PIE.

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 provide an early indication of the effectiveness of the NGNP/AGR Fuel Program in implementing this strategy. 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 normal operation service design conditions projected for the NGNP design. However, at the time of this assessment, postirradiation accident heating (i.e., safety) tests on the AGR-1 fuel had not provided sufficient failure rate data to support firm preliminary conclusions on the accident performance of the fuel under development 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.

3.1.1.2 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 DOE/INL 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

The International Atomic Energy Agency (IAEA) compiled an international database of HTGR fuel-related radionuclide transport data in the 1990s and summarized the database in IAEA-TECDOC-987, “Fuel Performance and Fission Product Behavior in Gas-Cooled Reactors,” issued November 1997. DOE/INL 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 through 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 will be compared to the international database values in IAEA-TECDOC-978. AGR-3/4 will be used to develop data necessary 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. DOE/INL 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 the helium purification system, and reactor building venting are other aspects of modeling radionuclide transport in the primary circuit and reactor building.

The DOE/INL 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.

DOE/INL used an expert elicitation process to conduct an assessment that it characterized as a conceptual phenomena identification and ranking tabulation (PIRT) on the effects of moisture ingress on the HTGR performance in February 2011. The major phenomena and issues that are of high importance and that require more attention, as noted by DOE/INL, are as follows:

- characterization of graphite properties and performance under both short- and long-term exposure to moisture
- investigation into the importance of the plateout and resuspension of radionuclides in the primary coolant system
- development of an accident code for a system that can simulate phenomena associated with moisture ingress
- additional scoping analysis to further identify phenomena and sequences that are important to the plant performance

DOE/INL, in collaboration with NRC, also conducted an HTGR dust workshop in March 2011. A document that describes potential HTGR dust safety issues and research and development needs was prepared based on 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 mainly 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:

(1) Establishment of a fuel-product specification

(2) Implementation of a fuel-fabrication process that can meet the specification

(3) Implementation of statistical QA procedures to demonstrate that the specification has been met

(4) Irradiation of statistically sufficient quantities of fuel with the monitoring of in-pile performance and PIE to demonstrate that normal operation performance requirements are met

(5) 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 NRC staff's preliminary view that the elements identified by DOE/INL 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 DOE/INL. The following additional element should be identified as a necessary first step:

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

Although Section 4 of the FQ white paper does address fuel service conditions and performance requirements, the staff believes 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 therefore be highlighted as a key element of the fuel qualification program. Section 3.2 discusses the basis for this view.[7]

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. DOE/INL has stated that the production-scale and production-line equipment and processes are planned to be effectively identical and that ongoing and planned NGNP/AGR Fuel Program activities will show they produce fuel of the same quality and variability. Future NRC review will thus be necessary to

[7] Related RAIs include 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, and B78.

confirm achievement of this stated goal. Pending confirmatory conclusions of the future review, the staff notes a potential need to supplement the planned fuel qualification program with "proof testing" of fuel fabricated on the NGNP fuel facility production lines. Such proof testing would be necessary should the staff's future review efforts identify needs for additional assurance that production-line fuel exhibits irradiation and safety performance equivalent to that of fuel fabricated with developmental production-scale equipment, procedures, quality controls, etc. A comprehensive fuel qualification program may thus be found to require the following element:[8]

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

In addition, the staff 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. In accordance with Title 10 of the *Code of Federal Regulations* (10 CFR), Section 50.43(e)(2), it is noted that such reliance on prototype programs to comply with testing requirements may necessitate the incorporation of special design features and instrumentation in the NGNP prototype that, subject to successful demonstration of testing compliance, would not be required in subsequent NGNP plant designs. As indicated in several of the subsections that follow, the staff broadly observes that reliance on NGNP prototype testing may be necessary to adequately demonstrate design safety features associated with fuel, core, and reactor system performance. Such prototype testing appears to be needed for fuel qualification in particular, given that the presented NGNP/AGR Fuel Program does not include real-time fuel irradiation testing in an HTGR-like neutron environment. The following element may therefore be identified as a necessary step for NGNP fuel qualification as well as mechanistic source term development:[9]

- 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 NRC staff's preliminary assessment is that the five fuel qualification program elements identified by DOE/INL, if supplemented by the additional program elements stated above, may constitute a reasonable structure for NGNP fuel qualification and related aspects of mechanistic source terms development. Finally, the staff observes that it may be both possible and desirable to address potential needs for irradiation proof testing in the NGNP prototype, thereby effectively combining the last two program elements into one.

The remaining sections present the NRC staff'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:

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

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[8] Related RAIs include F26/M31 and B55.

[9] Related RAIs include 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, and B80.

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

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

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

(5) 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))

(6) 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))

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

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

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

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

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

(12) 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.

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 NRC staff believes 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 effort 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 attention by DOE/INL and the NGNP applicant, commensurate with their importance.

3.2.1 Fuel Service Conditions for NGNP Normal Operations

For the NGNP prismatic block core design, DOE/INL anticipates that fuel service conditions for normal operations (i.e., peak fuel temperature, burnup, and fluence) will be significantly more

demanding than those associated with past and current HTGR test reactors and power reactor designs.

Currently, like the NGNP design itself, the fuel design service conditions for NGNP normal operations are not yet finalized and will have to be further specified by a future applicant. The normal operating fuel service conditions addressed in the NGNP/AGR Fuel Program's normal operation irradiation tests are presently based on what DOE/INL 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 have been 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 DOE/INL 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[10]

The NRC staff believes 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)

The NRC staff views plutonium burnup 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 staff'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.

Parameter Path Dependence[11]

In addition, the staff believes that additional information on how 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

[10] Related RAIs include 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, and B66.

[11] Related RAIs include F3/M3, B5, B47, and B48.

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 DOE/INL's proposed reliance on accelerated test irradiations in the ATR that address predominantly the higher ranges of fuel operating temperature, burnup, and fluence.

In follow-on discussions, DOE/INL stated that it has not seen evidence of parameter path dependence for normal fuel operating conditions but acknowledged the need to further evaluate this issue based on NGNP core design information yet to be established (ML12132A467). The staff notes that, once a reference NGNP core design has been sufficiently established, calculated end-of-cycle core maps of fuel irradiation parameter combinations (i.e., fuel temperature, burnup, fluence) should be provided and compared for coverage against the tested irradiation parameter combinations realized in the NGNP/AGR Fuel Program. The potential significance of any "path dependence" coverage gaps thus found should then be analyzed and evaluated using validated phenomenological models of TRISO fuel performance under operating conditions and accident conditions.

Operating Condition Uncertainties and Anomalies[12]

The staff 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 therefore the NRC staff'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 staff 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. In related RAI responses, DOE/INL has acknowledged this issue as one that should be addressed as detailed design information is developed for any future NGNP application. The staff agrees but also notes that bringing earlier attention to this issue could benefit the timely development and qualification of advanced sensor systems for NGNP prototype monitoring, surveillance, and testing, and that basic technical requirements for such sensor systems may prove largely generic to all modular HTGR design variants.

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., for accidents that involve coolant depressurization, reactivity excursion, detected/undetected fuel misloading, detected/undetected localized in-core coolant flow obstruction, etc.), maximum fuel particle oxidation (i.e., for depressurization accidents with air ingress), and maximum fuel kernel chemical attack (i.e., for moisture ingress accidents).

The accident service conditions assumed as the basis for AGR accident testing of the NGNP fuel are derived from what DOE/INL states to be a conservative assessment of the best

[12] Related RAIs include F3/M3, F6/M7, F7/M8, F10/M13, F11/M15, F22, F23/M28, and B29.

available information on the nuclear, thermal, and chemical environments 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. When finalized, it will thus be necessary to show that the accident service conditions 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[13]

It is the staff's view that, going forward, efforts by DOE/INL or NGNP applicant to establish 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 ingress events, air ingress events, and operating core "hot spot" events such as might result from detected/undetected core anomalies like fuel misloading or localized obstruction/starvation of core coolant flow.
- For analyzing reactivity excursions, as well as the potential for spatial xenon oscillations, the staff 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 and moisture 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 and exposed kernels 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., partial pressures of oxygen and moisture, temperatures, durations).
- 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:

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

The validity of this assumption should be evaluated by the staff in view of NGNP design and analysis details yet to be established. The staff 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

[13] Related RAIs include F22, F23/28, F33, F34/M38, F41/M45, B6, B22, B27, B30, B31, B32, B33, B64, and B66.

NRC staff what the potential severity would be of such power excursions or how they are being considered by DOE/INL 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 modular HTGRs have certain advantages in this regard over LWRs because of design-specific characteristics such as rod-ejection engineered safety features, rod-ejection mechanics, 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 reactivity power transient conditions should be qualified and assessed for that purpose. Should the results of any required reactivity excursion analysis reveal a need for transient or pulsed power fuel tests, this would add significant scope to DOE/INL's currently envisioned accident testing program for NGNP fuel.

Fuel accident conditions and performance requirements for (a) reactivity excursion events, (b) operating core "hot spot" events, (c) air ingress events, and (d) moisture ingress events should be further addressed as relevant NGNP design and analysis details are established. Reactivity excursions are further discussed in Section 3.10.5. Further observations pertaining to operating core "hot spots" are included in Sections 3.13.3 and 3.13.4.

3.2.3 Clarity and Adequacy of Fuel Performance Terminology[14]

The FQ and MST white papers define TRISO fuel particle failure based on fission gas release as a result of the 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 inservice fuel failure are specified in terms of quality requirements for all coating layers (as indicated in Table 16 of the FQ white paper).

- 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 Section 3.1.2 of the FQ white paper).

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 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

[14] Related RAIs include F15/M20, F24/M29, F48/M52, F49/M53, M86, M73, M115, B11, B13, B77, and B78.

tied to the fuel performance model being developed by DOE/INL, 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.

In describing these concerns in the initial assessment report, the NRC staff suggested that terms like “defective,” “failed,” and “functionally-failed” should be used to describe fuel particles in relation to the condition of individual coating layers and explain how fuel performance and radionuclide transport and release are considered and modeled in each case. DOE/INL subsequently responded in a public meeting by reporting that, in addition to the terms “intact” and “failed” used in the FQ white paper, future NGNP submittals would use the term “functionally degraded” to describe, for example, a fuel particle with intact PyC layers that retain fission gases and a defective or degraded SiC layer that allows the release of additional fission metals (e.g., cesium) (ML12132A467). The staff agrees that using this additional descriptive term can help bring necessary clarity to the evaluation and modeling of TRISO fuel performance.

3.3 Establishment of NGNP Fuel Designs with Product Specifications and Process Specifications for NGNP Fuel Fabrication[15] (FQ2) (ST2/3(a))

The objective of fuel design, fuel fabrication product specifications, and fuel fabrication process specifications is to produce fuel with the requisite high level of fuel performance and low level of fuel radionuclide releases during NGNP normal operations and LBEs (i.e., transients and accidents). Achieving these requisites is critical to enabling the safety analysis to show that the NGNP satisfies the top level NRC requirements in terms of dose consequences for occupational exposures, siting, 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, DOE/INL’s 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. DOE/INL states that these design margins (i.e., uncertainty factors) are largely based on engineering judgment. DOE/INL 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.

[15] Related RAIs include F43/M47, M62, M63, M64, M73, M115, B12, B15, B16, B33, B45, B52, B78, and B81.

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.

DOE/INL 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. DOE/INL 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, DOE/INL 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 fission metal releases more complex and uncertain, necessitating greater conservatism for fission metal releases than for fission gas releases and ultimately resulting in a factor-of-10 conservatism in developing the product specifications that affect fission metal releases from the fuel.

DOE/INL 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, DOE/INL 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 NRC staff 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.

At the time of the review, DOE/INL had finalized many but not all aspects of the NGNP fuel design (e.g., particle packing fraction in fuel compacts). DOE/INL 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.

3.3 Establishment of NGNP Fuel Designs with Product Specifications and Process Specifications for NGNP Fuel Fabrication[16] (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 operations and in accidents.

In this regard, DOE/INL 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 under accident heat-up conditions. DOE/INL 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 for normal operations and accidents.

DOE/INL's goal for fuel particle manufacture technology development is to achieve a fuel fabrication process that can produce 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, DOE/INL 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 with 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 with laboratory scale equipment and processes 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 DOE/INL has achieved considerable technical knowledge on the manufacture of fuel that can meet the NGNP fuel particle failure rate specifications for NGNP normal operation and heat-up accident conditions. The NRC staff'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 or encompass those used for the manufacture of the production fuel for the

[16] Related RAIs include F43/M47, M62, M63, M64, M73, M115, B12, B15, B16, B33, B45, B52, B78, and B81.

NGNP reactor. The staff 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.

3.5 Establishment and Implementation of a Fuel Fabrication Statistical Quality Control Procedure That Demonstrates That the Fuel Product Specifications Are Met[17] (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 (FFF) 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 in accordance with Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants," to 10 CFR Part 50, "Domestic Licensing of Utilization and Production Facilities."

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., the 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 staff'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, DOE/INL 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

[17] Related RAIs include F43/M47, B12, B15, B20, B21, B33, B34, B35, B36, B37, B40, B51, B53, B56, B60, and B63.

rejection of the population) becomes large. As such, for economic reasons, it may 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 could 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 could be used in part to support the confidence levels selected for the NGNP production fuel fabrication QA statistical analysis procedures.

In follow-on discussions, the NRC staff and DOE/INL agreed that completion of these items will be handled by the NGNP reactor and fuel vendors and the NGNP applicant. DOE/INL confirmed in a public meeting that the statistics of fuel characterization demand that the specifications be met with margin to keep the size of the sampled population of fuel particles manageable during production. Going forward, issues of concern to the NRC staff would include the extent to which NRC review and monitoring of fuel fabrication process parameters is needed and the extent to which fuel product characterization methods and procedures at the fabrication facility may vary from those used for the NGNP/AGR Fuel Program. DOE/INL offered to share any updated information related to these issues as it becomes available (ML12132A467).

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 Postirradiation Examinations (FQ2) (ST2/3(a))

3.6.1 Adequacy of Accelerated Irradiation Testing[18]

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, DOE/INL stated that acceleration factors in the completed, ongoing, and planned AGR fuel irradiations range from 1.6 to over 3. The lack of fuel performance data obtained in real-time HTGR neutron environments is of concern to the NRC staff. This concern is based on the questionable adequacy of data generated solely in accelerated irradiation environments. These NRC staff concerns are heightened in view of the fact that 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 (i.e., those in the United States, Germany, Japan, and China) have employed fuel irradiation testing in the real-time neutron environments of HTGRs

[18] Related RAIs include F1M/1, F2/M2, F3/M3, B49, B57, B58, B61, B69, B73, and B82.

as well as in the accelerated neutron environments of MTRs. This established 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 therfore be considered essential for adequately understanding fuel performance and for developing predictive models of fuel radionuclide retention and transport. Potential fuel performance issues that should 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 staff believes 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). Section 3.13 further discusses such uses of the NGNP prototype to meet testing requirements. The inclusion of a suitably designed post-irradiation fuel inspection and testing program for the NGNP prototype can provide the important confirmatory and supplemental 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 the NRC staff to be feasible. The NRC staff believes satisfactory completion of a post-irradiation fuel inspection and testing program achieved with irradiated fuel from the NGNP prototype is necessary to verify and supplement the technical basis for NGNP fuel qualification and mechanistic source terms code validation.

3.6.2 Adequate Plutonium Generation and Burnup in AGR Fuel Test Irradiations[19]

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

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

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., enriched to less than 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 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.[20] 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.”[21]
- Silver is also widely known to diffuse readily through SiC at moderately high fuel operating temperatures (e.g., greater than 1,100 degrees C). In the past, researchers have hypothesized that the cumulative effects of silver diffusion could alter the SiC grain boundaries. For example, W. Schenk et al. state the following: “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

[20] 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.

[21] 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.

precursors of SiC damage."[22] It could further be hypothesized that the effects of silver diffusion on SiC grain boundaries might 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.

- More representative (or conservative) fission product compositions in high-burnup irradiations of TRISO fuel in MTRs might be achieved by increasing the plutonium burnup fractions through some combination of the following actions:
  - Reduce the tested TRISO fuel's initial enrichment
  - Harden the MTR's thermal neutron spectrum
  - Increase the MTR's epithermal neutron spectrum
  - Replace some UO_2/UCO in the tested fuel kernels with PuO_2/PuCO.

In partial response to related NRC questions, DOE/INL provided TEV-1022,[23] a technical report with preliminary calculation results showing that, at a total fuel burnup of about 20% 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. In reaction to NRC comments provided in the initial assessment report, DOE/INL supplemented its earlier response to related questions by submitting ECAR-363[24] and ECAR-2066[25] (ML12298A516). The calculations presented in these reports appear predict plutonium burnup non-prototypicalities in the ongoing AGR-2 irradiation roughly similar to those presented in TEV-1022 for the AGR-1 irradiation.

Responses to associated RAI questions further indicated that DOE/INL's current approaches to increasing plutonium burnup in the AGR irradiation tests have relied 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 and AGR-2 analyses noted above, the staff presently views these approaches as unlikely to achieve HTGR-like levels of plutonium burnup.

DOE/INL's RAI responses also included a requested summary of the current state of knowledge on how palladium, silver, and rare earth fission products can affect TRISO fuel performance. The staff further considered these issues during the follow-on assessment phase in light of supplemental information and observations provided by DOE/INL in TEV-1620 (ML12268A032).[26] As summarized in TEV-1620, DOE/INL currently believes that emerging experimental evidence points to rare earths, palladium, and silver having little effect on NGNP

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[22] W. Schenk, D. Pitzer, and H. Nabielek, "Fission Product Release Profiles from Spherical HTR Fuel Elements at Accident Temperatures," Jül-2234, September 1988, p. 118.

[23] 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.

[24] J. Parry, G. Chang, "Physics Evaluations for the AGR-2 Experiment Irradiated in the ATR B-12 Position," ECAR-363, INL, April 19, 2010.

[25] J. Sterbentz, "Preliminary JMOCUP As-Run Daily Depletion Calculation for the AGR-2 Experiment in ATR B-12 Position," ECAR-2066, INL October 9, 2012.

[26] D. Petti, "Discussion of NRC FQ/MST Assessment Report Follow Up Items Related to Neutron Flux Spectrum and Effects of Silver, Palladium and Neutron Flux on Radionuclide Transport through Silicon Carbide," TEV-1620, TEM-103001-1, Rev. 03, INL, September 10, 2012.

TRISO fuel performance. Moreover, DOE/INL expects future results from the NGNP/AGR Fuel Program to further support this interpretation. The staff presently agrees with DOE/INL regarding the limited effects of rare earth fission products.

However, on reviewing the information provided, the NRC staff notes that the evolving phenomenological understanding of how palladium and silver interact with TRISO fuel coatings is still very limited. Continued research is needed to support a compelling explanation for the sporadic cases of palladium attack on SiC that have been reported internationally in the TRISO fuel technical literature. The NRC staff therefore continues to view plutonium burnup, time at operating temperature, and particularly palladium time-at-temperature as important parameters that should be considered in the irradiation and accident testing of TRISO fuel. The staff intends to further evaluate this issue as continuing research efforts by DOE/INL and others yield new insights into how plutonium fission products interact with coating layers on the TRISO fuel particles now being developed and tested by the NGNP/AGR Fuel Program.

3.6.3 Evaluation of Irradiation Test Conditions[27]

As noted in a related RAI question, given the central importance of TRISO fuel performance to the NGNP safety case, the staff has considered the possibility of performing independent NRC analyses of AGR test irradiation conditions, potentially including thermal analysis as well as nuclear analysis of associated fuel burnup isotopic compositions, and would be willing to pursue arrangements for gaining access to the detailed ATR information that would be needed for doing so. In follow-on discussions, DOE/INL noted that data needed for thermal modeling of the irradiations could be made available whenever requested but that special arrangements would be necessary for accessing the detailed information needed for nuclear analysis.

DOE/INL has reported that many of the thermocouples used to monitor temperature control in the AGR test irradiations to date have been found to fail as the irradiations progressed. In response to a related RAI question on how such thermocouple failures are accounted for in evaluating irradiation temperatures and associated uncertainties, DOE/INL explained how thermocouples embedded in the graphite sample holders are used in conjunction with detailed analytical models to determine fuel temperatures in the AGR irradiation tests.

As mentioned in the RAI responses, DOE/INL later provided more detailed information in INL-EXT-12-24761[28] and INL-EXT-12-25169,[29] which are technical reports respectively detailing the analysis of thermocouple data and the quantification of temperature uncertainties for the AGR-1 irradiation test (ML12205A039). It bears noting that the latter report estimates standard deviations between 45 and 60 ºC in the time-average peak fuel temperatures. Topics that will merit significant attention as the NGNP/AGR Fuel Program progresses include the evaluation of how such relatively large irradiation temperature uncertainties are (a) quantified, (b) affected by increasing thermocouple failures, and (c) conservatively treated in the contexts of fuel performance qualification and data development for use in developing and validating the analysis models for fuel radionuclide transport.

[27] Related RAIs include F5/M6 and B82.

[28] B. Pham, J. Einerson, "AGR-1 Thermocouple Data Analysis," INL/EXT-12-24761, May 2012.

[29] B. Pham, J. Einerson, G. Hawkes, "Uncertainty Quantification of Calculated Temperatures for AGR-1 Experiment," INL/EXT-12-25169, April 2012.

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 Postirradiation Heating Tests to Fuel Performance in HTGR Accidents[30]

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 HTGR accidents are expected to occur during high-power irradiation in the case of in-core helium flow obstruction events, fuel misloading events, and reactivity excursion events and 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. [31] 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 under off-normal and accident conditions in the NGNP prototype. This topic was not explicitly addressed in the FQ white paper or its supporting references.

The NGNP/AGR Fuel Program is planning to re-irradiate some irradiated fuel compacts just before the heating tests. In response to an RAI question, DOE/INL 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. DOE/INL's RAI response further asserted 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 by DOE/INL to support this conclusion.

The initial NRC assessment report noted that a quantitative comparison of the respective inventories of all elements produced by fission, activation, and decay should first be provided to explicitly show the degree of prototypicality for every element present in the fuel. Any substantial elemental inventory differences thus identified should then be evaluated with regard to their potential to affect fuel performance. Evaluation of applicability should further address the potential for fuel performance to 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.

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[30] Related RAIs include F2/M2, F19/M25, and B18.

[31] "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

The staff's assessment of the applicability of delayed fuel heatup testing proceeded during the follow-on assessment phase with DOE/INL's submittal of technical evaluation study TEV-1543.[32] The submitted study presents the computed inventories, both at-power and as a function of cooling time, of all elements having concentrations within 5 orders of magnitude of the highest element concentrations in irradiated fuel. Plotted results show niobium to be the only such element that changes substantially (e.g., by more than a factor of 2) in concentration during post-irradiation cooling. It is stated that niobium is expected to be present in UCO fuel as NbC, which, having a very high melting point and very low vapor pressure, should remain fixed within the fuel kernel.

With regard to the potential for fuel composition and performance to be affected by other physical phenomena in the interim between at-power irradiation and heatup testing, it is noted in TEV-1543 that the greatly reduced fuel temperatures experienced after irradiation serve to effectively preserve the fuel microstructure produced during irradiation and keep atomic mobility within the structure effectively nil. TEV-1543 further notes that the irradiated fuel sample's temperature is raised to 1250°C at the start of heatup testing and is held there for 12 hours to reestablish thermal conditions simulating normal operation prior to increasing the temperature to simulate core heatup under accident conditions.

The NRC staff believes that the information provided in TEV-1543 adequately demonstrates the applicability of delayed fuel heatup testing to the evaluation of fuel performance in HTGR accidents involving fuel heatup either during or after at-power irradiation.

3.7.2 Scope of Fuel Performance Testing for LBE Accident Conditions

The NRC staff discussed potential fuel performance tests and data to address reactivity excursion events, moisture-ingress events, and air-ingress events in Sections 3.2.2, 3.10.5, and 3.10.6 of this report. Associated issues are noted in the respective sections as meriting particular attention in the context of future safety reviews for licensing.

3.8 Irradiation and Accident Proof Testing of NGNP Fuel Fabricated on the Production Lines of the NGNP Fuel Fabrication Facility[33] (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 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.

[32] J. Sterbentz, "Delayed Heatup Testing," TEV-1543, Rev 1, INL, June 11, 2012.

[33] Related RAIs include F26/M31 and B55.

The FQ white paper further states that both an irradiation proof test and post-irradiation heating tests (of fuel produced in the NGNP fuel fabrication facility) 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.

However, in response to an RAI question, DOE/INL 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 DOE/INL 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."

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 attribute variability for fuel made on fabrication facility lines may differ from that for fuel made on a single production-scale line.

In follow-on discussions concerning this issue, DOE/INL clarified its intent to avoid the need for proof testing by using mixed batches of fuel made on the single production-scale line for AGR-5/6 to simulate the variability of fuel made on the fuel fabrication facility lines for the NGNP prototype (ML12132A467). The technical basis for this variability simulation approach was not described in detail but necessarily relies in part on future activities of the NGNP/AGR Fuel Program and should therfore be evaluated and confirmed by the NRC staff when such activities have been completed.

3.9 Definition of Event-Specific Mechanistic Source Terms for NGNP[34] (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, DOE/INL provided a clarified definition of "event-specific mechanistic source terms" as follows:

- <u>HTGR Source Term</u>. HTGR source terms are radionuclides that are released from the reactor building of a modular HTGR plant to the environment.

- <u>Mechanistic HTGR Source Term</u>. A mechanistic HTGR source term is a modular HTGR source term that is calculated with 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.

[34] Related RAIs include M4, M9, M65, M66, M82, M83, M84, M86, M87, M89, and M113.

- <u>Event-Specific Mechanistic HTGR Source Term</u>. An event-specific mechanistic HTGR source term is a mechanistic HTGR source term that is calculated for a specific LBE.

The NRC staff 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 may be 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 staff believes a conclusive assessment of HTGR mechanistic source terms includes consideration of deterministically selected events for which source terms are to be mechanistically calculated and that result in bounding releases from the reactor coolant system to the reactor building. An example of this type of bounding event is alluded to in the SRM to SECY-93-092 with regard to events leading to air ingress with graphite oxidation. For the vented reactor building concepts proposed for NGNP, functional requirements for radionuclide retention by the final release barrier should be considered in two phases: the bounding moisture ingress event (e.g., SGT failure) followed by a bounding large break in the reactor pressure boundary.

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 predicated on the 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.

In a September 20, 2012, meeting between DOE/INL and NRC staff, DOE/INL discussed a proposed methodology for addressing bounding events for the siting source term. DOE/INL proposed to supplement LBE-derived events with insights from a best-estimate mechanistic evaluation of postulated bounding event sequences, taking the following into account:

- Such events should be physically plausible.

- Events selected may have frequencies below that BDBE region.

- Events and event evaluations should consider the intrinsic and passive characteristics and the safety behavior of modular HTGRs.

The intent of NRC staff'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. DOE/INL 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.

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

In summary, the NRC staff believes that DOE/INL's definition of event-specific mechanistic source terms for modular HTGRs is generally consistent with the relevant Commission-approved staff recommendations in SECY-93-092 and SECY-03-047. However, the staff believes that appropriate consideration should be given to all available barriers in the assessment of event-specific mechanistic source terms. 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[35]

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:

> 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.

[35] Related RAIs include M72, M76, and M108.

> The design-specific source terms for each accident category would constitute one component for evaluating 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.

DOE/INL 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.

DOE/INL 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. DOE/INL 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.

The NRC staff believes that the proposed use of effective diffusion coefficients in connection with the use of Fick's second law is generally reasonable but that the proposal should be confirmed through testing. For this reason, in its response to a related RAI question, DOE/INL stated that once the AGR-3/4 test data become available, it would consider alternative transport models to correlate the data if a determination is made that the current Fick's second law diffusion-based model is inadequate. DOE/INL further stated that if a more complex model is ultimately adopted, supplemental testing would likely be necessary to obtain the supporting material property data.

Issue Resolution for Flux-Accelerated Diffusion of Metallic Fission Products during Irradiation[36]

Much of the German UO_2 test data for TRISO-coated particle diffusion rates published in TECDOC-978 are based on postirradiation heating tests. However, in response to an RAI question that raises the issue of flux-accelerated diffusion of cesium through intact SiC layers, 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 postirradiation heating tests is not recommended because those coefficients do not consider the irradiation effects, either implicitly or explicitly." DOE/INL 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, DOE/INL states that it will (1) use PIE to measure the release of fission products under irradiation, (2) analyze these measurements to establish diffusion coefficients under irradiation, and (3) compare the resulting diffusion coefficients to the historic values from IAEA-TECDOC-978.

Additionally, the intent of the irradiation, postirradiation, and safety testing for AGR-3/4 is 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 to allow measurements for the evaluation of the fission product gradient across the matrix and graphite surrounding the fuel compact through PIE. These gradients with knowledge of the irradiation temperature conditions will enable the back-calculation of diffusion coefficients for the matrix material.

Radionuclide Transport in the Compact-to-Graphite Gap of the Prismatic Fuel Element[37]

DOE/INL 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.

DOE/INL states 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.

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[36] Related RAIs include F2/M2, M9, F16/M21, F21/M27, F36/M40, and M78.

[37] Related RAIs include M57, M62, and M76.

The NGNP/AGR Fuel Program Plan states that single-effects test data will be needed to develop and refine sorptivity correlations in the fuel compact matrix and fuel element graphite with uncertainties within a factor of 10 at a 95% confidence level.

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

3.10.2 Modeling the Transport of All Radiologically Significant Radionuclides[38]

DOE/INL 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. DOE/INL proposes to classify radionuclides and species into one of nine radionuclide classes that in most cases are based on the periodic table of elements. DOE/INL proposes to develop experimental data on fission product transport for a representative radionuclide in each class (e.g., Cs-137 for alkali metals, I-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 severe accident analysis. The staff's view is that the proposed approach is reasonable.

However, DOE/INL states that the approach will assume that the release of iodine (i.e., a halogen) from the fuel kernel will be the same as the release of xenon (i.e., a noble gas) from the kernel. DOE/INL states that this assumption is conservative based on historical measurements of iodine and xenon release from UO_2 and UC_2. The NGNP/AGR Fuel Program will perform testing to confirm this assumption for UCO fuel. The staff believes that this assumption is potentially reasonable but notes that the testing results would be subject to future NRC review.

The NRC staff believes that DOE/INL's approach to developing the experimental data needed for modeling the relativeely large number of fission product species NGNP mechanistic source terms is reasonable.

3.10.3 Models and Data for Fuel Particle Performance during Normal Operation and Heatup Accidents[39]

This section of the initial NRC assessment report provided extensive assessment comments on DOE/INL's assumed use of the 1989 Goodin-Nabielek model for fuel performance, listing a total of 13 follow-up items related to various detailed aspects of the model. In subsequent follow-on discussions, DOE/INL stated that the Goodin-Nabielek model may not be used by the NGNP applicant and the staff agreed to reassess this topic in more general terms as reflected in the paragraphs that follow. The initial NRC assessment report (ML120240669) contains a detailed discussion of fuel particle performance models and data as framed in terms of the Goodin-Nabielek model.

[38] Related RAIs include M58 and M111.

[39] Related RAIs include 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, and B79.

With respect to degradation of the SiC layer due to corrosion, DOE/INL'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 bind with oxygen in the UCO kernel and remain in a stable oxide form. DOE/INL further states that carbon monoxide (CO) production and the resultant buildup of CO gas pressure in the fuel particle are also prevented as long as UC_2 and UO_2 are both present in the fuel kernel. The staff notes that with respect to SiC corrosion by CO during normal operation, DOE/INL'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.

In this regard, DOE/INL'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 staff 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.

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 fuel performance model caused by limited sample size, designers typically use statistical analysis to conservatively bound the fuel performance data at different confidence levels (e.g., 50%, 95%).

During the follow-on assessment phase, DOE/INL submitted three reports that provide previously requested pre-test code modeling predictions of fuel performance in the AGR-1 irradiation tests and heating tests and the AGR-2 irradiation tests.[40,41,42] The staff notes that both pre-test and post-test modeling predictions will be of continuing interest for their separate uses in test design, model improvement, and model validation, with emphasis shifting to the latter as the AGR test series progresses.

Going forward, the staff highlights the importance of evaluating how TRISO fuel performance models are developed and validated for predicting coating degradation and failure phenomena under normal, off-normal, and accident conditions. It bears reiterating in this context that the staff presently views the potentially degrading effects of plutonium fission products (e.g., Pd, Ag) as important, little understood, and less than fully addressed without test data from real-time irradiations in an HTGR neutron environment.

[40] EDF-5741, "AGR-I Pre-Test Prediction Analyses Using the PARFUME Code," Rev. 1, INL, May 2007.

[41] B. Collin, INL/EXT-12-26014, AGR-1 Safety Test Predictions Using the PARFUME Code, May 2012.

[42] K. Hamman, ECAR-1020, "AGR-2 Pre-Test Prediction Analyses Using the PARFUME Code for the U.S. Fuel Particles," Rev. 2, INL, May 2012.

3.10.4 Models and Data for Fuel Particle Performance during Reactivity Accidents[43]

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 those in HTGR core heatup accidents. Section 4.3 of IAEA-TECDOC-978 describes the Japanese and Russian reactivity-initiated accident testing and associated failure fraction results. Test conditions are described in terms of kernel energy deposition (J/g UO_2) rather than kernel or particle fuel temperature. The results indicate that the fuel failure fraction can become significant (i.e., greater then $1x10^{-5}$) when kernel energy deposition reaches about 600 J/g UO_2. At about 1,000 J/g UO_2, the failure fraction can reach 0.1.

In response to an RAI question, DOE/INL 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. The Japanese and Russian tests are therefore not representative of HTGR reactivity insertion events. DOE/INL 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. DOE/INL 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 the MHTGR Probabalistic Risk Assessment (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 MHTGR 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 determination of fuel energy deposition and maximum fuel temperature for the most limiting NGNP reactivity insertion accidents depends on NGNP design and analysis details that have not been established. Until this information is developed and reviewed, the staff will not be able to assess whether needs exist for fuel testing specific to NGNP reactivity excursions. Reactivity insertion events are further discussed in Section 3.2.2 above.

3.10.5 Models and Data for Accidents with Attack by Oxidants[44]

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 DOE/INL'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.

[43] Related RAIs include F33, F41/M45, B6, B22, B27, B30, B31, and B32.

[44] Related RAIs include F34/M38 and M61.

The assumption is that the encountered material greatly depletes the oxidants 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 (i.e., cesium and strontium that may have escaped the fuel particles and produced a "halo" around the fuel particles). Preferential reactions could possibly occur at these catalyst sites that create pathways for the rapid mass transport of oxidants to the fuel particles. Consideration of the catalysis of graphite oxidation in the analysis of either air or water intrusion accidents is not evident. An analysis of an oxidant attack on matrix material will need to consider mass transport and to include the following information:

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

The NRC staff notes DOE/INL'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.

During the follow-on assessment phase, DOE/INL submitted a research plan that contained more detailed information on the experiments that it intends to perform for moisture and air ingress.[45] The staff finds that the submitted experiment plan presents a reasonable approach for developing the data needed to model how air and moisture ingress can affect NGNP TRISO fuel performance and fission product transport. Ensuring that the experiments adequately envelop all LBEs that involve air or moisture ingress in the final NGNP design will be important. The sections below include more detailed staff observations and assessment comments on related topics.

Effects of Air on Particle Coating Layers[46]

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 issue directly affects the estimate of event-specific source terms.

DOE/INL stated in a response to an RAI question 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 millimiter (mm) thickness of graphite must first be permeated before oxygen reaches the fuel particles in either a prismatic block or pebble fuel element.

DOE/INL has outlined plans to conduct safety tests (fission product release at elevated temperatures) on compacts irradiated in graphite sleeves (to simulate the approximately 5-millimeter-thick web) or irradiated spherical fuel elements at various partial pressures of

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45 R. Hobbins, "Research Plan for Moisture and Air Ingress Experiments," PLN-4086, Idaho National Laboratory, Idaho Falls, ID, April 2012.

46 Related RAIs include F34/M38, M61, M74, M75, M81, and M95.

oxygen over a temperature range that has not yet been determined. It has also established 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 partial pressures of oxygen to confirm thermodynamic analyses.

The staff believes 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 required particle failure rate data for modeling particle failure during air ingress events. The staff also believes that the experimental study of SiO_2 formation versus SiO formation as a function of temperature and partial pressures of oxygen is important in providing a qualitative and quantitative understanding and confirmation of the particle degradation phenomena for the integral test results.

Effects of Moisture Ingress on Releases from Exposed Kernels[47]

NUREG/CR-6844 states that exposed kernels in failed particles can be oxidized in water ingress events (although the behavior of intact particles is much the same as for heatup events), thus releasing much of their stored fission product inventory relatively quickly. NUREG/CR-6844 further states that this effect is dependent on burnup. The NGNP/AGR Fuel Program has outlined additional tests that are necessary to characterize the effects of water ingress on fuel performance and fission product transport.

The NGNP/AGR Fuel Program Plan, states that a fuel heating facility will be developed to extend the chemical environment capabilities for 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.

The staff believes 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 required particle data for modeling the 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 will therefore be important in evaluating this issue.

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 reentrainment of carbonaceous dust contaminated with radionuclides; and distribution, condensation, plateout, and settling of radionuclides in the reactor cavity and the other volumes of the reactor building. Other modeling aspects of radionuclide transport in the primary circuit

[47] Related RAIs include F11/M15, F34/M38, M61, M80, M110, and M118.

and reactor building include the effects of moisture and air ingress on radionuclide transport and the role of the helium purification system and venting of functional containment.

DOE/INL's assumption of 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. The NGNP letter to NRC, CCN 228482 dated September 20, 2012, provides information on relative inventories of isotopes that are expected be (1) present in the core, (2) present in the helium coolant or plated out on helium pressure boundary surfaces, (3) released to the reactor building, and (4) released to the environment under various accident

Radionuclide Transport Behavior in the Primary Circuit and Reactor Building[48]

DOE/INL correctly recognizes that some fraction 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. DOE/INL also recognizes that currently available correlations for the deposition behavior of radionuclides have large uncertainties due to the 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.

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 NRC staff agrees with DOE/INL 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 staff further agrees with DOE/INL that data will be needed to develop and validate the fission product transport models in the reactor building under wet and dry conditions. The staff notes DOE/INL'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.

Generation of carbonaceous dust during the operational life of an HTGR is an additional area for evaluation. DOE/INL's current strategy is to use calculational tools to determine the impact of dust on the behavior of fission products in the system. DOE/INL 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 staff questions DOE/INL's confidence in the analytical results when not much is known about the dust behavior, and believes the

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

analytical effort needs to be complemented with experimental plans. However, during a July 24, 2012, meeting DOE/INL stated that the degree to which dust is produced in HTGRs with prismatic fuel is, based on historical data, significantly less than that for pebble bed designs. NRC stated at this meeting that future work on prismatic designs should include analyses to support the case for dust effects in those designs being negligible.

DOE/INL 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. Future review and assessment efforts should address radionuclide transport in the reactor building, including the potential crediting and modeling of filtration systems in mechanistic source term calculations.

Based on its response to a related RAI question, the staff believes that DOE/INL correctly recognizes that the manner in which mechanistic source terms are calculated may be affected by any future Commission policy decision on containment functional performance requirements. The staff notes that DOE/INL considers a vented reactor building as the best choice for accident mitigation. The staff further notes DOE/INL'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 staff to judge the relative merit of a vented-only reactor building in contrast to vented-and-filtered reactor building. It bears noting in general, however, that the staff has the option to require NGNP prototype design features that, subject to compliance with prototype testing requirements per 10 CFR 50.43(e)(2), may not be required for a standard NGNP design.

3.11.1 Modeling the Helium Purification System in Calculating NGNP MSTs[49]

DOE/INL 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 but is expected to have no significant effect for radionuclides with a half-life much less than a few hours.

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

The staff believes that the assumed circulating activity levels in the analysis of LBEs (i.e., LBE initial conditions) should be the maximum circulating activity allowed by the NGNP technical specifications. It is the staff' view that DOE/INL has described a reasonable approach for modeling the performance of the HPS in removing circulating radionuclides during the transient

[49] Related RAIs include M59 and M79.

phase of those LBEs in which the system remains in operation. Safety classification of the HPS is discussed, in general terms, in the NRC's 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[50]

DOE/INL states that studies were conducted to assess design options for the reactor building and the respective advantages and disadvantages of each option. DOE/INL 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.

DOE/INL 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 DOE/INL, 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 staff notes that the moisture ingress "conceptual PIRT" is a good start; however, it 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. The staff believes that these activities are a necessary first step in the development and validation of fission product transport models that incorporate the effects of moisture ingress.

DOE/INL, in collaboration with the NRC, conducted an HTGR dust workshop in March 2011. Based on discussions at the workshop, DOE/INL and the NRC concluded that a document that describes potential HTGR dust safety issues and research and development must be prepared. The staff believes that the preparation of such a document is also a necessary first step in the development and validation of fission product transport models that incorporate the contribution of dust.

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

DOE/INL'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

[50] Related RAIs include M60, M79, and M82.

uncertainties in understanding of phenomenology, DOE/INL proposed an uncertainty evaluation methodology as follows:

(1) The detailed calculational tools described in Section 4.5 and Appendices D and E to the MST white paper are used to predict the best-estimate, time-dependent mechanistic source term for a given LBE. These tools include separate computer codes for calculating the initial radionuclide inventories within the fuel and within the helium pressure boundary and for the modeling the off-normal event phenomena as described in the MST white paper.

(2) 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 used in this consequence uncertainty model to predict the mechanistic source terms for comparison to those obtained with the detailed calculational tools in Step 1.

(3) If confidence exists that the results of the simplified model in 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.

(4) 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 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 fuel elements), a determination of the initial inventories of the key radionuclides in the fuel compacts and fuel element graphite as a result of normal operation is made. The model accounts for the following radionuclide release mechanisms individually:

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

Similarly, for Volume 2 (the helium pressure boundary), an initial inventory of radionuclides is circulating and plated out on the primary circuit surfaces from 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 staff believes that this overall approach is generally reasonable, subject to considerations as noted in the subsections below.

3.12.2 Comprehensiveness of Proposed Uncertainty Models[51]

The staff 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 partly because of its ease of application. However, the analysis can have significant challenges. The difficulties commonly encountered are as follows:

- 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 the considerable experience and expertise of the analyst. Similar difficulties arise when parameters that are peculiar to a code or model and that must account, in some unspecified way, for phenomena that are not modeled in the code are selected for sampling. The only expertise on possible values is usually the experience of the code developer in this case.

- Definition of uncertainty ranges. Defining the range of values to be sampled is often quite difficult because not enough data exist. Existing data that may not be sufficiently "prototypic." Therefore, rigorously defining the range of values to be sampled can 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. These correlations must either be addressed, or their neglect must be justified. For example, "fuel inventory," "circulating inventory," and "plateout inventory" (as tabulated in response to a related RAI question) cannot obviously be independently sampled in a Monte Carlo uncertainty analysis.

The staff notes that the Monte Carlo uncertainty analysis proposed by DOE/INL 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.

3.12.3 Context-Specific Uses of the Terms "Best Estimate" and "Conservative" [52]

DOE/INL's "best estimate" calculations are described in the MST white paper as using several conservative approximations and assumptions. The staff 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.

[51] Related RAI: M88

[52] Related RAIs include M4 and M17.

The staff 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, DOE/INL confirmed that the proposed mechanistic source term calculations are intended for use in essentially all contexts, including emergency response and all applications of risk assessment. The staff notes that realistic or non-biased source term predictions may be most appropriate in certain contexts of emergency response and risk assessment.

In general, the staff 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 (1) defining or modeling the events themselves and (2) 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[53]

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

- For AOOs (or Anticipated Events (AEs), the dose calculation should realistically model all the SSCs modeled in the deterministic safety analysis of the AOO event sequence; however, the plant should use a conservative calculation of the mechanistic source term to demonstrate that it has met the dose limits in 10 CFR Part 20, “Standards for Protection against Radiation.” The staff believes this view is consistent with Issue 5 of SECY-03-0047, “Policy Issues Related to Licensing Non-Light-Water Reactor Designs,” dated March 28, 2003, in which the staff recommends that licensees use a conservative event-specific mechanistic source term calculation for AOOs. In the SRM for SECY-03-0047, the Commission approved the staff’s recommendations related to this issue. Moreover, the staff would expect the NGNP applicant to use a conservative calculation of mechanistic source terms to evaluate those AEs/AOOs used for defining the fuel and plant operational safetly limits and the technical specification basis.

- For DBEs and DBAs, the staff believes that the proposal to conservatively calculate the mechanistic source terms and dose consequences for DBEs to demonstrate compliance

[53] Related RAIs include M65, M66, M67, M68, and M71.

with 10 CFR 50.34, "Contents of Applications; Technical Information," is consistent with Issue 5 of SECY-03-0047 in which the staff recommends that the licensee use a conservative event-specific mechanistic source term calculation for DBEs. In the SRM for 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, "Feasibility Study for a Risk-Informed and Performance-Based Regulatory Structure for Future Plant Licensing," issued December 2007.

- For BDBEs, the dose calculation should realistically model all the SSCs modeled in the deterministic safety analysis of the BDBE event sequence, and the licensee should perform a best-estimate calculation of the MST to demonstrate that it has met the BDBE dose limits. However, in SECY-03-0047, the staff recommends that the licensee should use a conservative source term to make siting and containment decisions. The staff further states that the selection of events considered in the analyses to develop the set of source terms for each design must be done to bound severe accidents and design-dependent uncertainties. In the SRM for SECY-03-0047, the Commission approved the staff's recommendations related to this issue.

However, to bound severe accidents, it is the staff'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 DOE/INL 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 performance-based functional containment system), a conservative analysis may be required. The staff 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).

DOE/INL 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. DOE/INL 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 staff 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[54]

The NRC assessment of the NGNP LBE selection white paper provides additional preliminary staff views on the selection of LBEs and the calculation of the event-specific mechanistic source terms 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.

[54] Related RAIs include M70 and M72.

In the MST white paper DOE/INL 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 PRA references 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 combined license application. The draft ASME/ANS standard is presently being prepared for a second round of balloting. The reference draft ASME/ANS PRA standard states that all PRA elements (including the mechanistic source term element) must have a peer review.

3.13 Establishment and Implementation of NGNP Prototype Preoperational and Operational Programs To Verify and Supplement the Developmental Technical Bases for Fuel Qualification and Mechanistic Source Terms[55] (FQ2) (ST2/3(a)–(d))

3.13.1 Recommended Use of Prototype Provisions To Facilitate NGNP Prototype Licensing

The staff notes that the 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 appear in 10 CFR 50.43(e) and 10 CFR 52.78(a)(24) and in various sections of NUREG-0800, “Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition.” For example, Section 4.2, “Fuel System Design,” of Chapter 4, “Reactor,” of the NUREG-0800 refers to prototype testing by 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 staff believes that DOE/INL 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.

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55 Related RAIs include 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, and B80.

3.13.2 Use of Prototype Provisions To Verify and Supplement the Developmental Technical Bases for FQ and MSTs

The subject white papers seek NRC agreement that the presented technical approaches to fuel qualification and source term analysis and validation are acceptable. The NRC staff 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.

DOE/INL 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.

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:

(1) Accident source terms for modular HTGRs are sensitive to normal core operating conditions.

(2) 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 with a 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 cesium during normal operation and can weaken fuel particle coatings (e.g., due to a palladium 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[56] 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
- 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
- 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.[57, 58] 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.[59] 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

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

[57] 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).

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

[59] 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).

cited as major factors leading to higher than predicted peak core operating temperatures in the AVR and THTR pebble bed reactors.[60, 61]

DOE/INL 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.

3.13.4 Prototype Testing and Surveillance To Verify and Supplement the Development Technical Bases for NGNP Fuel Service Conditions and Fuel Performance[62]

The staff 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 DOE/INL 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. DOE/INL stated that the technical specifications for the Fort Saint Vrain (FSV) reactor 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 staff to be applicable to the NGNP. It is the staff's view that the NGNP should have technical specification LCOs and surveillances that are generally similar to those for FSV. The NRS staff believs that appropriate fuel or core-wide safety limits should be developed and included in the NGNP technical specifications and that the safety limits should be applicable to and be met for NGNP AEs/AOOs.

DOE/INL 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 strontium-90

[60] 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.

[61] 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.

[62] Related RAIs include F3/M3, F6/M7, F23/M28, F26/M31, F52/M56, M59, M60, M82, M102, and B55.

- Sampling stations and instrumentation to determine an overall mass balance for tritium

- Gamma scanning equipment to measure plateout activity on primary system surface

The NRC staff believes that the NGNP prototype should include the above types of instrumentation and additional instrumentation as needed to address considerations discussed in the paragraphs below.

In response to specific RAI questions, DOE/INL provided (a) general information on its limited university-based research efforts to date toward developing advanced in-core detector systems for HTGRs and (b) preliminary overview information on some of the types of surveillance and testing programs that DOE/INL would envision for the NGNP prototype. The latter DOE/INL RAI responses provided a preliminary, high-level overview of envisioned startup testing programs, demonstration testing programs, and operational surveillance programs, all of which the staff would generally consider helpful or necessary, depending on NGNP prototype details.

Noted below are the NRC staff 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 DOE/INL 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.

Going forward, the staff believes that the NGNP applicant should establish a clearer understanding of the full spectrum of testing, monitoring, and surveillance programs and associated instrumentation systems envisioned for the NGNP prototype. In addition, DOE/INL should establish 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 in accordance with 10 CFR 50.43(e)(2). Establishing such an understanding would

require information on the development and deployment of any advanced in-core detectors and an explanation of how DOE/INL will calibrate and use measurement data to address technical specifications and to verify and supplement the developmental technical bases for NGNP fuel qualification and mechanistic source terms.

4. CONCLUSION

The preceding sections have presented the NRC staff's detailed assessment comments in response to DOE/INL's requests for feedback on the technical approaches presented in the NGNP white papers on fuel qualification and mechanistic source terms. As stated above in Section 3.1, the NRC staff'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 staff has identified no fundamental shortcomings that would necessarily preclude successful implementation of the presented high-level approaches towards developing much of the technical bases for related NGNP prototype licensing submittals.

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 staff'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 DOE/INL'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 prototype testing per 10 CFR 50.43(e)(2) in conjunction with prototype-specific design features and special programs of operational surveillance, monitoring, testing, and inspection in the NGNP prototype.

The staff further believes that detailed consideration of such prototype provisions and programs could be beneficial to DOE/INL 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, therfore, 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

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

Listed alphabetically below are the NRC participants who contributed to assessing the NGNP white papers on fuel qualification and mechanistic source terms. Participation in the initial (1) and follow-on (2) assessment phases is indicated in parentheses after the name. Principal contributors for the respective phases are designated with an asterisk.

From the NRC Offices of New Reactors (NRO) and
Nuclear Regulatory Research (RES):

Sudhamay Basu, RES (1*)(2)
Thomas R. Boyle, NRO (1)(2)
David D. Brown, NRO (2)
Donald E. Carlson, NRO (1*)(2*)
Russell E. Chazell, NRO (2)
Nan-Pin D. Chien, NRO (2)
Jonathan DeGange, NRO (2*)
Hossein Esmaili, RES (1)
Michelle L. Hart, NRO (2)
Andrew J. Nosek, RES (1)(2)
Shie-Jeng Peng, NRO (2)
Stuart D. Rubin, RES (1*)
James J. Shea, NRO (1)(2*)
Christopher N. VanWert, NRO (2)
Joseph F. Williams, NRO (1)

From Brookhaven National Laboratory (BNL) and
Sandia National Laboratories (SNL):

Lap-Yan Cheng, BNL (1)(2)
Lynne Ecker, BNL (1)
Randall O. Gauntt, SNL (1)
Michael J. Kania, BNL consultant (1)(2)
Hans Ludewig, BNL (1)(2)
Dana A. Powers, SNL (1)
John U. Valente, BNL (1)
Robert Wichner, SNL consultant (1)
Michael F. Young, SNL (1)

APPENDIX B

DOE/INL Requests for NRC Feedback on NGNP Fuel Qualification and Mechanistic Source Terms

The DOE/INL 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. DOE/INL presented specific requests for feedback within its respective FQ and MST white papers and updated these requests in a letter dated May 3, 2011 (ML111250375).

Then DOE/INL'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 DOE/INL'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, DOE/INL provided the following updates:

"Following submittal of the white papers, the strategy for fuel acquisition for the NGNP (Ref. 3)[63] 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)[64] 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).[65] In general there is considerable overlap in data, allowing comparison of results from parallel tests.

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[63] [3] D. Petti, et al., INL/EXT-07-12441, Rev. 2, "Updated NGNP Fuel Acquisition Strategy," December 2010

[64] [4] INL/PLN-3636, "Technical Program Plan for the Next Generation Nuclear Plant/Advanced Gas Reactor Fuel Development and Qualification Program," September 2010

[65] [5] IAEA-TECDOC-978, "Fuel Performance and Fission Product Behavior in Gas-Cooled Reactors," November 1997

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