
October, November, and December 2016

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Advanced Reactor Technologies (ART):
Very High Temperature Reactor (VHTR) Research and Development (R&D) Quarterly Report

October, November, and December 2016

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INL ART TDO Program
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Advanced Reactor Technologies (ART):
Very High Temperature Reactor (VHTR) Research and Development (R&D) Quarterly Report

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Approved by:
Diane V. Croson
INL ART TDO Deputy Director
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<td>three-dimensional</td>
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<tr>
<td>AGC</td>
<td>Advanced Graphite Creep</td>
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<td>AGR</td>
<td>Advanced Gas Reactor</td>
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<td>AL</td>
<td>analytical laboratory</td>
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<td>ART</td>
<td>Advanced Reactor Technologies</td>
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<tr>
<td>ASME</td>
<td>American Society of Mechanical Engineers</td>
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<tr>
<td>BOC</td>
<td>beginning of cycle</td>
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<td>BWXT</td>
<td>BWX Technologies</td>
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<tr>
<td>CCCTF</td>
<td>Core Conduction Cooldown Test Facility</td>
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<tr>
<td>CRP</td>
<td>Coordinated Research Project</td>
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<td>EBSD</td>
<td>electron back-scatter diffraction</td>
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<tr>
<td>EDS</td>
<td>energy-dispersive spectrometer</td>
</tr>
<tr>
<td>EOC</td>
<td>end of cycle</td>
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<tr>
<td>FIMA</td>
<td>fissions per initial heavy metal atom</td>
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<td>HFEF</td>
<td>Hot Fuel Examination Facility</td>
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<td>HFIR</td>
<td>High Flux Isotope Reactor</td>
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<td>HOPG</td>
<td>highly oriented pyrolytic graphite</td>
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<td>high-resolution transmission electron microscopy</td>
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<td>HTTF</td>
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<td>high temperature vessel</td>
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<td>IAEA</td>
<td>International Atomic Energy Agency</td>
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<tr>
<td>IMGA</td>
<td>irradiated microsphere gamma analyzer</td>
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<td>INL</td>
<td>Idaho National Laboratory</td>
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<tr>
<td>IPyC</td>
<td>inner pyrolytic carbon</td>
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<tr>
<td>$K_{ic}$</td>
<td>fracture toughness</td>
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<tr>
<td>LWR</td>
<td>light water reactor</td>
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<tr>
<td>MCNP</td>
<td>Monte Carlo N-Particle Transport</td>
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<tr>
<td>MHTGR</td>
<td>modular high temperature gas-cooled reactor</td>
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<td>MPO</td>
<td>memorandum purchase order</td>
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<td>MURR</td>
<td>Missouri University Research Reactor</td>
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<td>Nuclear Energy Agency</td>
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<td>Full Form</td>
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<tr>
<td>NRC</td>
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<td>OECD</td>
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<td>OPyC</td>
<td>outer pyrolytic carbon</td>
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<tr>
<td>ORNL</td>
<td>Oak Ridge National Laboratory</td>
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<tr>
<td>PBMR</td>
<td>Pebble Bed Modular Reactor State Owned Company Ltd</td>
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<tr>
<td>PED</td>
<td>precession electron diffraction</td>
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<tr>
<td>PGS</td>
<td>precision gamma scanner</td>
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<tr>
<td>SCALE</td>
<td>Standardized Computer Analyses for Licensing Evaluation</td>
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<tr>
<td>SEM</td>
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<tr>
<td>SFF</td>
<td>Specialty Fuels Facility</td>
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<tr>
<td>STEM</td>
<td>scanning transmission electron microscopy</td>
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<tr>
<td>TAVA</td>
<td>time-average, volume-average</td>
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<tr>
<td>TBD</td>
<td>to-be-determined</td>
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<tr>
<td>TCT</td>
<td>technical coordination team</td>
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<tr>
<td>TEM</td>
<td>transmission electron microscopy</td>
</tr>
<tr>
<td>TRISO</td>
<td>tristructural isotropic</td>
</tr>
<tr>
<td>TTAF</td>
<td>Test Train Assembly Facility</td>
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<tr>
<td>UAM</td>
<td>Uncertainty Analysis in Modeling</td>
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<tr>
<td>UCO</td>
<td>uranium carbide/oxide</td>
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<tr>
<td>UO₂</td>
<td>uranium oxide</td>
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<tr>
<td>VHTR</td>
<td>very-high temperature reactor</td>
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<tr>
<td>WG</td>
<td>with grain</td>
</tr>
<tr>
<td>XRD</td>
<td>X-ray Diffraction</td>
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Advanced Reactor Technologies (ART):
Very High Temperature Reactor (VHTR) Research and Development (R&D) Quarterly Report

1. MAJOR ACCOMPLISHMENTS

1.1 Fuels Development

Highlights of Fuels Development activities during October through December 2016 are as follows:

1.1.1 Fuels Development

Highlights of High Temperature Materials Development activities during October through December 2016 are as follows:

October

- Completed corrective actions to allow resumption of uranium-handling operations in the Specialty Fuels Facility (SFF) at BWX Technologies (BWXT).
- Completed third and final pre-production tristructural isotropic (TRISO) coating run, 93166, at BWXT.
- Established parameters for the first production TRISO coating run, 93167. Kernels were charged and the run initiated, but an exhaust-line leak resulted in the run’s being terminated with only a partial buffer layer being deposited.
- Completed leach-burn-leach (LBL) of 1800°C safety-tested AGR-2 Compact 5-4-1 particles at ORNL.
- Completed gamma scanning with the Irradiated Gamma Microsphere Analyzer (IMGA) of particles deconsolidated from Advanced Gas Reactor (AGR-2) UCO Compact 5-4-1.
- Completed cross-sectioning and optical microscopy on special particles from AGR-2 Compact 3-4-2, chosen to investigate defects identified by x-ray, randomly-selected particles from as-irradiated AGR-2 UCO Compact 6-3-3, and randomly-selected particles from as-irradiated UO₂ Compact 3-3-1.
- Issued post-irradiation examination (PIE) plan for 1800°C safety testing of AGR-2 UO₂ Compact 3-4-1; safety test will be started in November 2016 at Oak Ridge National Laboratory (ORNL).
- Completed shipment of the fifth set of four AGR-2 compacts from Idaho National Laboratory (INL) to ORNL.
- Completed a trial radial deconsolidation of an un-irradiated AGR-3/4 fuel compact at the Analytical Laboratory (AL) at INL. Video was taken of the entire process for use in optical-image analysis software intended to allow continuous measurement of the compact radius throughout deconsolidation.
- Performed sieving and boiling/re-sieving of debris and particles from radial deconsolidation in attempts to retrieve the small designed-to-fail (DTF) particles from the AGR-3/4 compact.
- Completed Sr-90 separation and counting of AGR-3/4 Capsule 7 sink ring, and preliminary results were received from AL.
- Initiated analysis of the gamma count data from AGR-3/4 fluence wires at Pacific Northwest National Laboratory (PNNL).
• Continued installation and testing of the out-of-cell gamma counting station in Hot Fuels Examination Facility (HFEF).

• Issued Phase I and II qualification plan for physical sampling of AGR-3/4 inner and outer graphite rings.

• Initiated replacement of the 1M window at the HFEF. The old window tank unit has been removed and a new unit installed. Completion of the window replacement is expected early November.

• Implementation of the new HFEF Documented Safety Analysis (DSA) is on scheduled with an implementation date of November 15, 2016.

• Completed two low-enriched uranium shipments of pre-production AGR-5/6/7 TRISO particles from BWXT to ORNL for measurement of anisotropy, characterization of uranium dispersion/defective inner pyrolytic carbon (IPyC) fraction, and x-ray inspection for any other anomalies.

• Completed repair of high-temperature heat treatment furnace and initiated heat treatment of three BWXT TRISO particle samples for uranium dispersion/defective IPyC analysis.

• Prepared and reviewed 2016 International Topical Meeting on High Temperature Reactor Technology (HTR2016) presentations.

• Binh Pham presented the paper “Gap Size Uncertainty Quantification in Advanced Gas Reactor TRISO Fuel Irradiation Experiments” at the 11th International Topical Meeting on Nuclear Reactor Thermal Hydraulics, Operation and Safety in Gyeongju, Korea.

November

• AGR program staff attended the HTR2016 in Las Vegas, Nevada, presenting technical papers on various aspects of the AGR program, including irradiation and post-irradiation examination.

• Several AGR program staff attended the Advanced Fuels Technology Technical Leads Meeting in Germantown, MD, November 15-16, 2016. AGR staff gave three presentations that provided program historical background and licensing perspectives, a high-level overview of program technical activities, and some detailed results from recent PIE on AGR-2 fuel compacts. A main objective of the AGR staff presence at this meeting was to introduce the rest of the managers and technical leads in the Nuclear Energy Office of Advanced Fuels Technologies (NE-42) to the work that has been ongoing in the AGR program over the last decade.

• Completed defective IPyC analysis (uranium dispersion) and enumeration of other anomalies on particle samples from AGR-5/6/7 pre-production coating batches 93164, 93165, and 93166 at ORNL.

• Completed optical microscopy and x-ray tomography on 1800°C safety-tested AGR-2 Compact 5-4-1 particles at ORNL.

• Completed optical inspection and gamma scanning of particles deconsolidated from AGR-2 uranium oxycarbide Compact 2-3-1 at ORNL using the IMGA.

• Received results from gamma scanning of dissolved niobium flux wires from the AGR-2 experiment Capsules 1, 3, and 5. Both Nb-94 and Nb-93m were measured in the lab at ATR. These data will be used to determine the fluence in these capsules and for comparison to predictions from physics calculations.

• Performed trial AGR-3/4 radial deconsolidation using a matrix blank (matrix material with no particles) at INL. Optical measurement software algorithms were adjusted for different lighting conditions. Consistent dimensional data were obtained in fume hoods, through glass fume hood sashes, and on the benchtop.
- Analyzed video of rotating matrix blank through the hot-cell window at Cell 5 in the AL at INL. Optical measurement software algorithms were adjusted for different lighting conditions and preliminary results were reasonable; however, chromatic aberrations existed, which affected the results. Equipment and/or lighting will be modified to reduce chromatic aberrations and improve the ability of the software to distinguish edges and measure AGR-3/4 compacts at various increments during radial deconsolidation.

- Received inductively coupled plasma mass spectrometry (ICP-MS) results from trial AGR-3/4 radial deconsolidations on unirradiated compacts. Very low amounts of uranium were measured in the solutions, indicating that the DTF particles are still intact after deconsolidation.

- Received ICP-MS results from AL analyses of solutions from burn-leach of AGR-3/4 Capsules 1, 3, and 7 sink rings.

- Completed re-scanning of AGR-3/4 Capsules 5, 8, and 10 sink ring lids for gamma-emitting fission products.

- Oxidized and leached AGR-3/4 Capsule 8 sink ring. Leachate will be analyzed for fission products. During oxidation of Sink 8, a clean piece of graphite was also oxidized to judge cross contamination. If no cross contamination occurred, it may be possible to oxidize multiple graphite pieces simultaneously.

- Retrieved melt wires from AGR-3/4 Capsules 1 and 3 inner rings at the HFEF. Melt wires from inner rings of Capsules 7 and 10 could not be removed without risking ring damage.

- Loaded and began precision gamma scanner (PGS) analysis of the outer ring of AGR-3/4 Capsule 3 at HFEF. This scan will use two to four times as many angles in order to improve the tomographic reconstruction. Completion of this will mean two sets of inner and outer ring tomographs have been obtained: IR/OR-3 and IR/OR-8.

- Resumed normal operations at HFEF following implementation of the new Documented Safety Analysis (DSA).

- Submitted paper for publication in *Journal of the European Ceramic Society*: H. Wen and I. J. van Rooyen, “Distribution of fission products palladium, silver, cerium and cesium in the un-corroded areas of the locally corroded SiC layer of a neutron irradiated TRISO fuel particle.”

- Initiated transmission electron microscopy (TEM) and scanning transmission electron microscopy examination (STEM) on TRISO particle AGR2-223-RS06 Lamellae 4 (IPyC-SiC interface). The new part-time, post-doctoral researcher received his first TEM training on this lamellae and learned and practiced the electron-beam alignment, imaging in bright field mode and selected area diffraction pattern. Preliminary bright-field images and selected area electron diffraction (SAED) from some SiC grains and precipitates were obtained.

**December**

- Terminated TRISO particle coating run J52O-16-93170 approximately one minute into the buffer deposition when a flammable-gas detector alarmed. The source of the flammable gases, related to acetylene used to deposit the buffer layer, has been linked to a broken stem on the graphite gas distributor. Following the replacement of the distributor nozzle, BWXT could not achieve the required vacuum level on the coater to enable another run to be executed in December 2016.

- Repaired broken manipulator in hot cell 5 at the Analytical Laboratory (AL).

- Completed oxidation (burn) of AGR-3/4 Capsule 10 sink ring.

- Transferred AGR-3/4 sink rings 2, 6, 9, and 12 into hot cell at AL.
• Collected video of rotating compact through cell 5 window at AL for tuning radial deconsolidation image analysis capability.

• Held videoconference for the AGR technical coordination team (TCT).

• Completed 1700°C safety testing of AGR-2 UO₂ Compact 3-4-1 in the ORNL Core ConductionCooldown Test Facility (CCCTF).

• Completed deconsolidation, pre-burn leach, and matrix burn-leach of as-irradiated AGR-2 UCO Compact 5-4-2; particle survey with IMGA is in progress.

• Completed materialographic preparation and optical microscopy of random samples (31–42 particles in each sample) riffled from AGR-2 UO₂ Compact 3-3-1, AGR-2 UCO Compact 5-3-3, and AGR-2 UCO Compact 6-3-3 to observe average irradiated particle microstructure.

• Completed microstructural analysis of select particles from AGR-2 UO₂ Compact 3-3-1 and AGR-2 UCO Compact 6-3-3 with scanning electron microscope (SEM) equipped with an energy-dispersive spectrometer (EDS).

• Completed x-ray tomography on three additional particles that released cesium during 1600°C safety testing of AGR-2 UO₂ Compact 3-4-2 due to CO corrosion of the SiC.

• Completed author checks, updates, technical editing and peer reviews on report INL/EXT-15-36281, “Advanced Electron Microscopy and Micro Analytical Technique Development and Application on Irradiated TRISO-Coated Particles from the AGR-1 Experiment.”

• Prepared scope of work for the FY 2017 high-resolution transmission electron microscopy (HRTEM) work at Los Alamos National Lab (LANL) on fuel kernels.

• Held phone interviews with two graduate intern candidates to identify a candidate to perform his/her PhD experimental work on the UCO kernels using STEM and EDS. A candidate of University of Florida was identified. It is estimated that this candidate will start in March 2017 at CAES.

• Prepared a draft abstract for the 4th Workshop on High Temperature Gas-cooled Reactor Silicon Carbide Materials Properties to be held in China during June 2017. Working title of presentation is “Fission product composition and distribution in the SiC layers of neutron irradiated AGR-1 TRISO fuel particles,” I. J. van Rooyen, C. Parga, J. Rosales, T. M. Lillo and K. Wright

• Completed measurement of pyrocarbon anisotropy on AGR-5/6/7 pre-production coating batch 93166A using the 2-modulator generalized ellipsometry microscope (2-MGEM) at ORNL.

• Completed x-ray tomography on particles from AGR-5/6/7 pre-production coating batches 93164, 93165, and 93166 previously identified with uranium dispersion due to defective IPyC analysis or other anomalies of interest.

• Completed shipment of three AGR-5/6/7 production batches from BWXT to ORNL for quality control and acceptance determination of pyrocarbon anisotropy and defective IPyC fraction.

• Completed 1800°C heat-treatment and x-ray radiography of 120,000 particles from each of three AGR-5/6/7 production batches (93168A, 93169A, and 93170A) as part of the defective IPyC analysis.

1.1.2 High Temperature Materials Development

Highlights of High Temperature Materials Development activities during October through December 2016 are as follows:
**High Temperature Design**

- Issued sections of the Alloy 617 high temperature code case related to the physical properties and time dependent properties sections of the code case are out for ballot for approval by Working Group Allowable Stress Criteria.

- Completed scoping tests of Alloy 617 fatigue behavior to examine the maximum necessary mean stress correction for the design fatigue curve at 1100°F. Two baseline tests and corresponding tests with the maximum calculated mean stress for comparison are done. Analysis of the results indicates that the current mean stress correction proposal based on the ASME methodology is far too conservative.

**Resolution of Regulatory Issues**

- Completed analysis of the local plastic zone at the notch of failed v-notch specimens using electron back-scatter diffraction (EBSD). Testing of v-notch creep rupture specimens at 750 and 1000°F are continuing.

- Continuing work on Bridgman type notch specimen (with u-shaped notches) at 800°C.

**1.1.3 Graphite Development and Qualification**

Highlights of Graphite Development and Qualification activities during October through December 2016 are as follows:

**October**

- Continued a small scoping study to determine the effects of oxidation on graphite performance. Post oxidized specimens are being tested for changes to mechanical strength, thermal conductivity, and stiffness modulus. Specimens are being oxidized to 5 and 10% mass loss at three temperature levels (550°C, 650°C, and 750°C) to determine the changes in material properties in two different grades.

- Initiated collaboration with Missouri University Research Reactor (MURR) to perform a flux wire irradiation experiment designed to verify the INL’s Monte Carlo N-Particle Transport (MCNP) Code responsible for calculating Advanced Graphite Creep (AGC) dose levels continues. This is critical for determining the irradiation dose uncertainty calculations for the AGC irradiations.

- Continued baseline material property testing activities. The current work focuses on completing the unirradiated material property testing for all remaining partial billets (billets that have had testing initiated, but for which tests are not quite complete). Currently five partial billets are still undergoing testing, and are nearly (i.e., 75–80%) complete.

**November**

- Completed a draft technical memorandum (TM) reporting the Critical stress intensity factor, $K_{ic}$, (fracture toughness) of several graphite grades. This was prepared and reviewed for accuracy. The fracture toughness values resulting from this small study will be utilized in the ASTM International nuclear graphite standard specifications.

- Initiated PIE testing at ORNL of graphite specimens from the high-temperature vessel (HTV) capsule. Dimensional change inspection of the specimens has been completed. SiC temperature monitors from the 900°C (design temperature) sections of the capsule have been read, and the temperature-monitor thermal-expansion data, analyzed. The four temperature monitors gave an average irradiation temperature of 869°C, suggesting that the capsule was operating slightly cooler than design calculations suggested.


**December**

• Submitted a joint INL and ORNL written manuscript, titled “Understanding the Reaction of Nuclear Graphite with Molecular Oxygen: Kinetics, Transport, and Structural Evolution,” to *Journal of Nuclear Materials*. Authors are Joshua J. Kane (corresponding author), Cristian I. Contescu, Rebecca E. Smith, Gerhard Strydom, and William E. Windes.

• Initiated preliminary PIE testing of HTV graphite specimens. Dimensional and mass-change inspections have been completed. Stiffness (Young’s) modulus measurements have been initiated. Analysis of the preliminary volume and dimensional changes illustrate volume/dimensional change rate increases for AGC graphite grades at irradiation temperatures between 900°C and 1500°C.

• Completed fabrication of the graphite-sample-preparation glovebox. Glovebox passed inspection, and delivery of the system is expected in early January 2017.

• Attended the ASTM International meeting for subcommittee D02.F0, Manufactured Carbon and Graphite Products, in Lake Buena Vista, FL, December 6-7, 2016. Finalized methodology for new ASTM standard on Brazil nut (split-nut) mechanical test procedures. Laboratory participants were identified, and all samples and the custom test fixtures fabricated at the INL are being shipped to participants in January 2017.

• Completed significant modifications to ASTM standard C747, “Standard Test Method for Moduli of Elasticity and Fundamental Frequencies of Carbon and Graphite Materials by Sonic Resonance”. All changes to the standard have been balloted and approved. Standard will be issued in 2017.

• Continuing AGC-3 PIE testing. All testing is approximately 85% complete and is ahead of schedule. AGC-3 samples were irradiated over a dose range of 1–3.5 dpa at a temperature of 800°C.

• Continuing irradiation damage studies utilizing x-ray diffraction (XRD) and small-angle neutron scattering (SANS) techniques to ascertain the changes to graphite-crystal structure changes. Highly oriented pyrolytic graphite (HOPG) and polycrystalline specimens from AGC-1 irradiation are being analyzed over a range of irradiation temperatures.

1.1.4 Methods

Highlights of Methods activities during October through December 2016 are as follows:

**October**

• Completed re-installation of the gas circulator at the High Temperature Test Facility (HTTF) at Oregon State University (OSU).

• Continued with the definition of the depletion exercises required for Phase II of the International Atomic Energy Agency (IAEA) Coordinated Research Program (CRP) on High Temperature Gas-Cooled Reactor (HTGR) uncertainties. The results of three depletion cases were compared in October using the Scale/Keno and Serpent Monte Carlo codes, ranging from a single prismatic block to a one-third core model. These test cases will be reported to the CRP participants by the end of March 2017, and will be used to define the final specifications for Exercise I of Phase II.

**November**

• Completed an NQA-1 audit of the quality assurance program for the OSU HTTF; two findings were identified.
• Successfully collected data from a gas exchange test between the pressure vessel lower plenum and the reactor cavity simulation tank using a dual laser system from a Nuclear Energy University Program associated with the HTTF.

• Attended the HTR2016 conference in Las Vegas to present two papers on the HTGR Test Reactor and the IAEA CRP on HTGR Uncertainties. An ad hoc workshop of the CRP working group was also organized to plan the activities for 2017.

December

• Attended a fluoride-cooled high temperature reactor Phenomena and Ranking Table (PIRT) workshop at Georgia Tech in Atlanta. The workshop was focused on multi-physics phenomena that are simulated using coupled neutronics, thermal hydraulics and material properties, and was attended by experts from academia, national laboratories and the NRC.

• Discovered and removed accumulated water from the pressure vessel lower head and gas circulator housing on equipment at the High Temperature Test Facility (HTTF).

2. SIGNIFICANT ACCOMPLISHMENTS

2.1 Fuels Development

2.1.1 Fuel Development and Fabrication

2.1.1.1 TRISO Coating. The last of the pre-production runs, J52O-16-93166, was completed. Certified UCO kernels were used for the run so that it could be considered for inclusion in the TRISO lot for AGR-5/6/7, but because of the interruption in the coating process between the deposition of the inner pyrocarbon and the silicon carbide layers, this pre-production run is excluded from consideration.

Three TRISO particle coating runs for AGR-5/6/7 were successfully completed in November and characterization data are being generated. A sectioned particle from one of the production runs is shown in Figure 1.

![Figure 1. TRISO particle from batch J52O-16-93168.](image-url)
2.1.1.2 **Compacting Process Development.** Volumetric-feeder insert volumes have been estimated for the fabrication of compacts for AGR-5/6/7 with 25% and 40% TRISO packing fractions. A range of feeder inserts have been ordered to ensure that compacts of the desired length can be fabricated.

Resinated graphite powder intended for use in forming the AGR test compacts was analyzed for impurities. Data from previous compact fabrication were used to derive estimated decontamination factors to estimate the residual impurities that will exist in the compacts after thermal treatment. Two drums were tested from the powder lot. One drum was found to be lower in impurities than the other. Resinated graphite powder was obtained from the drum with the lower impurities for AGR-5/6/7 fuel compact fabrication.

2.2 Fuels Development and Qualification

2.2.1 Fuel Development and Fabrication

2.2.1.1 **X-ray Analysis of AGR-5/6/7 Pre-Production Coated Particle Batches.**

Determination of several important defect fractions in TRISO particles is a critical step in ensuring that TRISO fuel both meets quality control (QC) specifications and performs well in-reactor. The AGR-5/6/7 fuel specification includes an upper limit of 1E-4 (determined statistically to a confidence of 95%) on the population of particles classified to have “defective IPyC”. Whereas there are additional specified requirements on the properties of the IPyC (thickness, density, and anisotropy), a particle is specifically defined to have “defective IPyC” if it exhibits uranium dispersion from the kernel into the surrounding buffer or IPyC layers when imaged with X-rays. Particles with IPyC that is highly permeable, cracked, or otherwise compromised as a gas-retentive layer will allow intrusion of HCl gas during the initial stages of SiC deposition. This HCl can leach uranium out of the kernel and deposit it in the surrounding layers when the particles are heated, especially during the 1800°C heat treatment that is part of the compact production process.

Particles with defective IPyC are identified through a process which includes heat-treating particles using the same ramp rates, target temperatures, and hold times used for compact heat-treatment (20°C/min to 1800°C in vacuo with a one hour hold, followed by ≤20°C/min cooldown). Heat-treated particles are mounted in single layer arrays and imaged with low-resolution x-ray radiography to identify particles in which uranium from the kernel has migrated into the buffer or IPyC. While the primary purpose of this testing is the identification of particles with uranium dispersion due to defective IPyC, the radiography is also useful in identifying other anomalies which may be relevant to fuel performance or may highlight parts of the fabrication process that can be improved.

Defective IPyC analysis has been completed at ORNL on samples from three upgraded coating batches of pre-production TRISO particles produced by BWXT in preparation for AGR-5/6/7 fuel fabrication. Particles from each upgraded batch (93164A, 93165A, and 93166A) were heat treated, radiographed, and analyzed for any obvious defects or anomalies. Approximately 120,000 particles were analyzed from each coating batch to provide adequate statistical sampling and minimize the statistical penalty when determining the defect fractions at 95% confidence using the binomial distribution. A summary of the results is given in Table 1. Batches 93164A and 93166A exhibited Defective IPyC populations above the specified limit of ≤1E-4, while 93165A satisfied the specification.
Table 1. Results of defective IPyC analysis of samples from three upgraded coater batches.

<table>
<thead>
<tr>
<th>Coater Batch</th>
<th>93164A</th>
<th>93165A</th>
<th>93166A</th>
</tr>
</thead>
<tbody>
<tr>
<td>Number of particles in analyzed sample</td>
<td>119944</td>
<td>121032</td>
<td>114223</td>
</tr>
<tr>
<td>Number of particles with defective IPyC</td>
<td>25</td>
<td>2</td>
<td>18</td>
</tr>
<tr>
<td>Acceptance criteria (number of defective particles)</td>
<td>≤6 (fail)</td>
<td>≤6 (pass)</td>
<td>≤5 (fail)</td>
</tr>
<tr>
<td>Measured defect fraction in analyzed sample</td>
<td>2.1E-4</td>
<td>1.7E-5</td>
<td>1.6E-4</td>
</tr>
<tr>
<td>Minimum 95% confidence defect fraction(a)</td>
<td>≤3.0E-4</td>
<td>≤5.3E-5</td>
<td>≤2.4E-4</td>
</tr>
</tbody>
</table>

\(a\). Minimum specified fraction for the batch that each particular sample would have satisfied at 95% confidence.

Figure 2 shows four example radiographs of particles with obvious uranium dispersion that were classified as having defective IPyC. These radiographs are 2-dimensional projections formed by x-rays passing through nominally-spherical particles before striking an x-ray sensitive film, with brighter contrast indicating higher x-ray absorption. The well-defined bright circle in the center of each particle is the kernel and the outer bright ring is the SiC layer. In between these two features, the typical non-defective particle will be darker, with a slight contrast gradient moving from the perimeter of the kernel projection to the outer ring (as evident in the particles in each image surrounding the central defective particles). This contrast gradient is produced as a result of an increasing path length through the spherical SiC shell as the x-ray beam (normal to the image) strikes the surface of the sphere at increasingly-tangential trajectories. In particles with uranium dispersion, uranium located in the buffer and IPyC layers between the kernel and SiC causes additional x-ray absorption. Typically, the uranium dispersion is concentrated at the interface between the buffer and the IPyC layers resulting in an additional bright ring between the kernel and the SiC perimeter, as well as an additional contrast gradient inside this ring. However, uranium can sometimes move out into the IPyC layer resulting in brighter zones beyond the buffer perimeter.
Figure 2. Example radiographs of particles with uranium dispersion (centered in each image).

Table 2 summarizes the various other anomalies that were noted during the defective IPyC analysis. The table includes an example of each identified anomaly, the number of particles in each sample with that anomaly, the measured fraction of particles with each anomaly in the analyzed sample, and a minimum 95% confidence anomaly fraction for the entire batch based on binomial distribution statistics and the measured sample. Also included in this table are results of the analysis of an additional 62444-particle sample from Batch 93164A that was heated to 2000°C for 1 h. This higher temperature heat treatment did not appear to impact the population of anomalies.
Table 2. Summary of other anomalies in samples from three upgraded coater batches.

<table>
<thead>
<tr>
<th>Defect Type</th>
<th>Particle Lot</th>
<th>Number observed</th>
<th>Measured fraction in sample</th>
<th>95% confidence limit for batch</th>
</tr>
</thead>
<tbody>
<tr>
<td>Faceted Kernel and/or Dimpled Coating</td>
<td>93164A</td>
<td>931</td>
<td>7.8E-3</td>
<td>≤8.2E-3</td>
</tr>
<tr>
<td></td>
<td>93164Aa</td>
<td>500</td>
<td>8.0E-3</td>
<td>≤8.7E-3</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>1474</td>
<td>1.2E-2</td>
<td>≤1.3E-2</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>1440</td>
<td>1.3E-2</td>
<td>≤1.4E-2</td>
</tr>
<tr>
<td>Single Notch</td>
<td>93164A</td>
<td>1659</td>
<td>1.4E-2</td>
<td>≤1.5E-2</td>
</tr>
<tr>
<td></td>
<td>93164Aa</td>
<td>718</td>
<td>1.1E-2</td>
<td>≤1.3E-2</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>73</td>
<td>6.0E-4</td>
<td>≤7.4E-4</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>91</td>
<td>8.0E-4</td>
<td>≤9.5E-4</td>
</tr>
<tr>
<td>Double Notch</td>
<td>93164A</td>
<td>149</td>
<td>1.2E-3</td>
<td>≤1.5E-3</td>
</tr>
<tr>
<td></td>
<td>93164Aa</td>
<td>41</td>
<td>6.6E-4</td>
<td>≤8.6E-4</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>5</td>
<td>4.1E-5</td>
<td>≤8.7E-5</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>4</td>
<td>3.5E-5</td>
<td>≤8.1E-5</td>
</tr>
<tr>
<td>Irregular Kernel</td>
<td>93164A</td>
<td>164</td>
<td>1.4E-3</td>
<td>≤1.6E-3</td>
</tr>
<tr>
<td></td>
<td>93164Aa</td>
<td>41</td>
<td>6.6E-4</td>
<td>≤8.6E-4</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>82</td>
<td>6.8E-4</td>
<td>≤8.2E-4</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>135</td>
<td>1.2E-3</td>
<td>≤1.4E-3</td>
</tr>
<tr>
<td>Multi-kernel</td>
<td>93164A</td>
<td>6</td>
<td>5.0E-5</td>
<td>≤9.9E-5</td>
</tr>
<tr>
<td></td>
<td>93164Aa</td>
<td>1</td>
<td>1.6E-5</td>
<td>≤7.6E-5</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>0</td>
<td>0</td>
<td>≤2.5E-5</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>1</td>
<td>8.8E-6</td>
<td>≤4.2E-5</td>
</tr>
<tr>
<td>Defect Type</td>
<td>Particle Lot</td>
<td>Number observed</td>
<td>Measured fraction in sample</td>
<td>95% confidence limit for batch</td>
</tr>
<tr>
<td>--------------</td>
<td>--------------</td>
<td>----------------</td>
<td>----------------------------</td>
<td>--------------------------------</td>
</tr>
<tr>
<td>White Spots</td>
<td>93164A</td>
<td>49</td>
<td>4.1E-4</td>
<td>≤5.2E-4</td>
</tr>
<tr>
<td></td>
<td>93164A&lt;sup&gt;a&lt;/sup&gt;</td>
<td>8</td>
<td>1.3E-4</td>
<td>≤2.4E-4</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>26</td>
<td>2.1E-4</td>
<td>≤3.0E-4</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>9</td>
<td>7.9E-5</td>
<td>≤1.4E-4</td>
</tr>
<tr>
<td>Thin SiC Layer</td>
<td>93164A</td>
<td>13</td>
<td>1.1E-4</td>
<td>≤1.8E-4</td>
</tr>
<tr>
<td></td>
<td>93164A&lt;sup&gt;a&lt;/sup&gt;</td>
<td>2</td>
<td>3.2E-5</td>
<td>≤1.1E-4</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>6</td>
<td>5.0E-5</td>
<td>≤9.8E-5</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>11</td>
<td>9.6E-5</td>
<td>≤1.6E-4</td>
</tr>
<tr>
<td>Extra Layers</td>
<td>93164A</td>
<td>0</td>
<td>0</td>
<td>≤2.5E-5</td>
</tr>
<tr>
<td></td>
<td>93164A&lt;sup&gt;a&lt;/sup&gt;</td>
<td>1</td>
<td>1.6E-5</td>
<td>≤7.6E-5</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>0</td>
<td>0</td>
<td>≤2.5E-5</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>2</td>
<td>1.8E-5</td>
<td>≤5.6E-5</td>
</tr>
<tr>
<td>Missing Kernel</td>
<td>93164A</td>
<td>5</td>
<td>4.2E-5</td>
<td>≤8.8E-5</td>
</tr>
<tr>
<td></td>
<td>93164A&lt;sup&gt;a&lt;/sup&gt;</td>
<td>2</td>
<td>3.2E-5</td>
<td>≤1.1E-4</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>2</td>
<td>1.7E-5</td>
<td>≤5.3E-5</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>0</td>
<td>0</td>
<td>≤2.7E-5</td>
</tr>
<tr>
<td>Kernel Migration</td>
<td>93164A</td>
<td>2</td>
<td>1.7E-5</td>
<td>≤5.3E-5</td>
</tr>
<tr>
<td></td>
<td>93164A&lt;sup&gt;a&lt;/sup&gt;</td>
<td>1</td>
<td>1.6E-5</td>
<td>≤7.6E-5</td>
</tr>
<tr>
<td></td>
<td>93165A</td>
<td>0</td>
<td>0</td>
<td>≤2.5E-5</td>
</tr>
<tr>
<td></td>
<td>93166A</td>
<td>1</td>
<td>8.8E-6</td>
<td>≤4.2E-5</td>
</tr>
</tbody>
</table>

<sup>a</sup>. Heat treated to 2000°C.
The first five defect types listed in Table 2 appeared to be mostly-related to the same root cause—kernel fragmentation prior to coating. Kernel fragmentation has previously been studied in connection with the presence of fissures within the kernel structure (see monthly program reports from December through May 2014). These fissures were found to lead to fragmentation of the kernels during handling, sieving, or initial fluidization in the coater, and where fragmentation led to significant faceting, dimpled coating shapes were observed to form. The total population of particles that appeared to have some level of kernel damage ranged from 1.5–2.5%; however, additional damage may not be visible in the single-orientation radiographs acquired for the defective IPyC analysis. Minor shape irregularities due to small chips in the kernel surface are not expected to impact irradiation performance; however, AGR-1 PIE has indicated that severely dimpled coating shapes may be more prone to failure during safety testing. Multi-kernel particles were rare; these are usually the result of kernel fragments bonding to other kernel fragments or whole kernels.

A number of particles were classified as having abnormal white spots in the radiograph region between the kernel projection and the SiC perimeter. These did not share the same characteristic as particles with uranium dispersion as a result of defective IPyC and were not counted as such. There are several possible sources for the additional localized x-ray absorption that causes these white spots. Some could be small debris from kernel fragmentation or some other high-Z inclusion (molybdenum and stainless steel inclusions have been previously observed and identified). However, these white spots can also be caused debris on the surface of the particle or on the Kapton tape used to seal the single-layer particle arrays. Tape mounts were visually inspected for debris external to the TRISO coating and for a sample of ten particles with white spots in the radiographs, nine were found to have external debris corresponding to the observed white spots.

Table 3. Results of defective IPyC analysis of samples from three upgraded coater batches.

<table>
<thead>
<tr>
<th>Coater Batch</th>
<th>93164A</th>
<th>93165A</th>
<th>93166A</th>
</tr>
</thead>
<tbody>
<tr>
<td>Measured fraction of particles with kernel migration</td>
<td>1.7E-5</td>
<td>1.6E-5</td>
<td>8.8E-6</td>
</tr>
<tr>
<td>Pre-burn leach defect fraction</td>
<td>2.6E-5</td>
<td>1.3E-5</td>
<td>9.9E-6</td>
</tr>
</tbody>
</table>
Several particles with the most significant types of defects or anomalies were selected for additional analysis to further study their microstructure. These particles were extracted from the Kapton tape mounts used for bulk radiography and individually imaged using high-resolution x-ray tomography. The resulting tomographic data sets (2000 radiographs acquired at 0.09° rotations) provided a higher resolution than the original radiographs and tomographic reconstruction of these data sets allowed visualization of the particles without the shadowing effect caused by projecting a three-dimensional (3D) object onto a two-dimensional image. The tomographs shown below are two-dimensional cross sections of each particle obtained by determining the absorption of a 3D array of small volume elements (voxels). To allow for efficient analysis of multiple particles, particles were mounted in high-throughput holders that were not optimized for the highest-quality tomographic imaging, so numerous reconstruction artifacts can be seen in the tomographs that can be avoided if imaging conditions were optimized. The following images show the original film-based radiograph used to identify each defect or anomaly coupled with single plane tomographs acquired at higher resolution.

The particle shown in Figure 3 exhibited the common uranium dispersion appearance consisting of an increasing contrast gradient extending out from the kernel to a bright ring that is usually coincident with the perimeter of the buffer/IPyC interface. However, the original radiograph also showed an IPyC layer that was unusually thick, with an additional partial ring of higher intensity halfway through the layer. The tomograph in Figure 3 clearly shows the benefit of 3D analysis. What appeared to be an unusually-thick IPyC layer was actually a normal IPyC layer deposited on top of a soot inclusion located between the buffer and IPyC. Not visible in the radiograph was a large protrusion where the soot inclusion was much thicker. The uranium dispersion was mostly concentrated at the surface of the buffer, between the buffer and the soot inclusion. The tomograph shows little additional x-ray absorption occurring within the bulk of the buffer, which illustrates how the appearance of the gradient region in the radiograph can be misinterpreted as uranium dispersion throughout the buffer region when it is actually caused by penetration of the x-rays through a dense spherical shell of uranium piled up at a specific radial location. The tomograph in Figure 3 also shows evidence that some uranium moved from the outer surface of the buffer, through the soot inclusion, to the inner surface of the IPyC; this was the source of the faint ring in the radiograph halfway between the buffer and the SiC. An IPyC crack at the edge of the thicker portion of the soot inclusion can be seen in the Figure 3 tomograph; this could be the defect in the IPyC that caused the uranium dispersion.

Figure 3. Particle with uranium dispersion from 93164A; radiograph (left) shows uranium outside of kernel with pileup at two radial locations, and tomograph (right) reveals a soot defect between the buffer and IPyC with uranium piled up at the outer boundary of buffer and inner boundary of IPyC.
Figure 4 shows another particle classified as having uranium dispersion that had an unusual region between the buffer and IPyC layers. Based on the x-ray data alone, it is not clear whether this region is a gap between the two layers that formed after coating or is a low-density soot inclusion similar to the particle in Figure 3. Three-dimensional inspection of the abnormal region did not reveal a protrusion like that in the previous particle, which would allow it to be reliably identified as an inclusion. As in the particle shown in Figure 3, the particle in Figure 4 exhibited uranium pile-up at both the buffer outer boundary and the IPyC inner boundary. Intermittent pileup of uranium 10–15 µm from the kernel surface was observed in numerous other particles with uranium dispersion. Additional regions of increased x-ray absorption were found at several location in the IPyC; the enlarged image at the bottom of Figure 4 shows several of these. Follow-up SEM analysis with EDS of a polished cross section is planned to further characterize these regions in the IPyC and determine whether the increased x-ray absorption is due to uranium or possibly some other high-Z material. The fan-shaped nature of these features indicates that they may be radiating outward from a source located at the IPyC/SiC boundary. The largest of these features shown in Figure 4 also shows some thinning of the SiC at the center of the fan-shaped region that suggests chemical degradation of the SiC, such as might occur in the presence of a transition-metal impurity. It seems likely that these features in the IPyC layer are related to the observed uranium dispersion, but additional data are needed to determine that relationship.
The particle in Figure 5 exhibited an above-average amount of x-ray absorption in the buffer layer. The tomographs of this particle show that there was more uranium throughout the buffer than the particles in the preceding figures. The buffer was also unusually thin (40–50 µm) while the IPyC was unusually thick (50–60 µm).

Figure 5. Particle with above average uranium dispersion from 93166A; radiograph (left) and tomograph (right) show uranium dispersion throughout the buffer layer, and enlarged tomograph (bottom) shows the IPyC layer was unusually thick.

The variety of types of uranium dispersion observed in the three pre-production batches suggests varied failure mechanisms. Figure 6 shows a particle in which the uranium pileup at the buffer/IPyC interface was not as well defined and uranium was dispersed into the IPyC layer. Figure 7 shows a particle in which the uranium dispersion was localized in certain regions of the buffer. Figure 8 shows a particle in which uranium dispersion only extended about 60 µm out into the buffer (note that buffer was thicker than average: 130–140 µm).
Figure 6. Particle with uncommon uranium dispersion from 93164A; low-resolution radiograph (upper left), tomograph (upper right), and high-resolution radiographs (bottom) all show uranium dispersion into the IPyC layer.
Figure 7. Particle with asymmetric uranium dispersion in the buffer layer from 93164A*; radiograph (left) and tomograph (right).

Figure 8. Particle from 93164 in which uranium dispersion was limited to inner half of buffer; radiograph (left) and tomograph (right).

Figure 9 and Figure 10 show particles for which the radiographs could be mistaken for uranium dispersion based on the contrast gradient moving from the perimeter of the kernel projection to a ring distinct from the outer SiC ring. However, the tomographs of these particles show that both of these particles actually had two SiC layers. Particles identified as having uranium-dispersion defects were reviewed to ensure that none of these particles with additional layers were incorrectly assigned.
Figure 9. Particle from 93164A* with extra SiC and OPyC layers in which the additional absorption due to extra layers could be mistaken for uranium dispersion; radiograph (left) and tomograph (right).

Figure 10. Particle from 93164A with extra SiC and OPyC layers in which the additional absorption due to extra layers could be mistaken for uranium dispersion; radiograph (left) and tomograph (right).

Figure 11 and Figure 12 show particles that had damaged TRISO coatings such that their kernels were not enclosed by gas-tight layers. As discussed in conjunction with Table 3 above, these particles are detected during the preburn stages of LBL analysis when their exposed kernel are dissolved in the analysis acid. During 1800°C heat treatment, the kernels in these particles reacted with the surrounding buffer and IPyC layers. The tomographs show that the kernels migrated through these layers toward a region where the SiC and OPyC layers are missing. The volume of the kernels was significantly reduced by the interaction with the surrounding buffer, which was also consumed.
Defective IPyC analysis is currently in progress on 120,000-particle samples from five AGR-5/6/7 production batches. These particles are being heat treated, x-ray imaged, and analyzed in a similar fashion to the preproduction batches described herein. More extensive tomography with improved image quality is planned for defective and anomalous particles from these production batches to provide a thorough understanding of the nature of defects in the AGR-5/6/7 fuel.
2.2.2 Safety Testing and Post Irradiation Examination

2.2.2.1 AGR-2 Fission Product Mass Balance Update. AGR-2 compacts, graphite compact holders, metallic capsule hardware, and holder spacers have been analyzed for gamma-emitting fission products. Additionally, activities of the beta-emitter Sr-90 have been measured from the metallic capsule hardware. Leach analysis of the spacers and burn-leach analysis of the graphite capsule holders is still required in order to complete the Sr-90 mass balance. Figure 13 and Figure 14 summarize the mass balance for select isotopes in AGR-2 Capsule 2 (UCO fuel, time-averaged, volume-averaged [TAVA] temperature 1194°C). Figure 15 and Figure 16 summarize the mass balance for select isotopes in AGR-2 Capsule 3 (UO₂ fuel, TAVA temperature 1045°C). Figure 17 and Figure 18 summarize the mass balance for AGR-2 Capsule 5 (UCO fuel, TAVA temperature 1101°C). Figure 19 and Figure 20 summarize the mass balance for AGR-2 Capsule 6 (UCO fuel, TAVA temperature 1074°C). The most notable result from the plots below is that the Sr-90 fraction measured on the Capsule 3 metallic hardware is higher than that measured on the metallic hardware from the hotter UCO capsules, and it is much higher than the measured Eu fractions which were below the minimum detectable activity for Capsule 3. The cause of this seemingly higher Sr-90 release is not known.

Figure 13. AGR-2 Capsule 2 (UCO fuel) Ag-110m and Eu-154 mass balance.

Figure 14. AGR-2 Capsule 2 (UCO fuel) Cs-134/137, Eu-155, Sb-125, and Sr-90 mass balance.
NOTE 1: The Eu-155 and Sb-125 values are reported as minimum detectable activities (MDAs) since no activity of either isotope was measured.
NOTE 2: Sr-90 analysis currently lacks data from the spacers and holders.
Figure 15. AGR-2 Capsule 3 (UO₂ fuel) Ag-110m and Eu-154 mass balance.

Figure 16. AGR-2 Capsule 3 (UO₂ fuel) Cs-134/137, Eu-155, Sb-135, and Sr-90 mass balance.
NOTE 1: The Eu-155 and Sb-125 values are reported as minimum detectable activities (MDAs) since no activity of either isotope was measured.
NOTE 2: Sr-90 analysis currently lacks data from the spacers and holders.
Figure 17. AGR-2 Capsule 5 (UCO fuel) Ag-110m and Eu-154 mass balance.

Figure 18. AGR-2 Capsule 5 (UCO fuel) Cs-134/137, Eu-155, Sb-135, and Sr-90 mass balance.

NOTE 1: The Eu-155 values are reported as minimum detectable activities (MDAs) since no activity was measured. For Sb-125, a fraction of 5E-7 was measured in the spacers, nothing was measured in the holders, and an MDA of 4E-5 was determined for the metallic hardware.

NOTE 2: Sr-90 analysis currently lacks data from the spacers and holders.
2.2.2.2 AGR-3/4 Fission Product Mass Balance Update. Table 4 and Table 5 summarize the measured isotope inventories in AGR-3/4 inner rings, outer rings, and sink rings. Sr-90 inventories were obtained for sink rings 1, 3, and 7 via the burn-leach method. Model predictions for the Sr-90 inventory in the sink rings were 0.0 for Sink 1, 9E-12 for Sink 3, and 1E-7 for Sink 7. Thus, the model significantly under-predicts the Sr-90 inventory in the rings. As a result, all sink rings will undergo burn-leach analysis for Sr-90. Model predictions for most other isotopes have generally been within a factor of 10 of the measured values. It is believed that the gaseous precursor to Sr-90 (Kr-90/Rb-90) is responsible for the higher measured Sr-90 inventories in the rings compared to the predicted values from the model, which does not account for this precursor.

Table 4. Gamma-emitting isotope inventory measured for AGR-3/4 inner rings using the Precision Gamma Scanner (PGS).

<table>
<thead>
<tr>
<th>Capsule</th>
<th>Type</th>
<th>Material</th>
<th>Inner Ring Inventory</th>
</tr>
</thead>
</table>

**Figure 19.** AGR-2 Capsule 6 (UCO fuel) Ag-110m and Eu-154 mass balance.

**Figure 20.** AGR-2 Capsule 6 (UCO fuel) Cs-134/137, Eu-155, Sb-135, and Sr-90 mass balance.

NOTE 1: The Eu-155 values are reported as minimum detectable activities (MDAs) since no activity was measured. For Sb-125, a fraction of 1.1E-6 was measured in the spacers, nothing was measured in the holders, and an MDA of 8.2E-5 was determined for the metallic hardware.

NOTE 2: Sr-90 analysis currently lacks data from the spacers and holders.

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Table 5. AGR-3/4 outer ring gamma-emitting isotope inventory measured using PGS. AGR-3/4 sink ring activities measured using rotating-drum gamma scanning at the Analytical Laboratory. Sr-90 analysis of the sink rings was accomplished via burn-burn leach of the sink rings.

<table>
<thead>
<tr>
<th>Capsule</th>
<th>Outer Ring Inventory (Capsule Fraction)</th>
<th>Sink Ring Inventory (Capsule Fraction)</th>
</tr>
</thead>
<tbody>
<tr>
<td></td>
<td>Ag-110m</td>
<td>Cs-134</td>
</tr>
<tr>
<td>1</td>
<td>0.0E+0</td>
<td>1.9E-5</td>
</tr>
<tr>
<td>2</td>
<td></td>
<td></td>
</tr>
<tr>
<td>3</td>
<td>1.4E-1</td>
<td>5.0E-4</td>
</tr>
<tr>
<td>4</td>
<td>4.2E-4</td>
<td>7.2E-5</td>
</tr>
<tr>
<td>5</td>
<td></td>
<td></td>
</tr>
<tr>
<td>6</td>
<td>2.1E-2</td>
<td>9.7E-4</td>
</tr>
<tr>
<td>7</td>
<td>7.0E-1</td>
<td>3.4E-3</td>
</tr>
<tr>
<td>8</td>
<td></td>
<td></td>
</tr>
<tr>
<td>9</td>
<td>4.0E-1</td>
<td>1.2E-3</td>
</tr>
<tr>
<td>10</td>
<td></td>
<td></td>
</tr>
<tr>
<td>11</td>
<td></td>
<td></td>
</tr>
<tr>
<td>12</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>
AGR-2 compacts will continue to provide additional statistics and better understanding of the fuel performance, several already-evident trends for fission product release are highlighted in the table and discussed below.

Table 6. AGR-2 safety tests in order of completion.

<table>
<thead>
<tr>
<th>Compact ID</th>
<th>Fuel Type</th>
<th>Irradiation Temp. (TAVA)</th>
<th>Average Burnup (FIMA)</th>
<th>Safety Test Temp.</th>
<th>Failed SiC</th>
<th>134Cs Release</th>
<th>110mAg Release</th>
<th>154Eu Release</th>
<th>90Sr Release</th>
</tr>
</thead>
<tbody>
<tr>
<td>3-3-2</td>
<td>UO₂</td>
<td>1062°C</td>
<td>10.5%</td>
<td>1600°C</td>
<td>≥6</td>
<td>2.1×10⁻³</td>
<td>1.7×10⁻²</td>
<td>3.8×10⁻³</td>
<td>1.4×10⁻³</td>
</tr>
<tr>
<td>3-4-2</td>
<td>UO₂</td>
<td>1013°C</td>
<td>10.7%</td>
<td>1600°C</td>
<td>≥26</td>
<td>9.3×10⁻³</td>
<td>1.1×10⁻²</td>
<td>3.2×10⁻⁴</td>
<td>2.7×10⁻³</td>
</tr>
<tr>
<td>2-2-2</td>
<td>UCO</td>
<td>1287°C</td>
<td>12.6%</td>
<td>1600°C</td>
<td>0</td>
<td>2.5×10⁻⁷</td>
<td>7.3×10⁻¹</td>
<td>4.7×10⁻²</td>
<td>4.1×10⁻²</td>
</tr>
<tr>
<td>5-2-2</td>
<td>UCO</td>
<td>1141°C</td>
<td>12.3%</td>
<td>1600°C</td>
<td>0</td>
<td>5.9×10⁻⁶</td>
<td>1.7×10⁻²</td>
<td>1.1×10⁻³</td>
<td>7.9×10⁻⁴</td>
</tr>
<tr>
<td>2-3-1</td>
<td>UCO</td>
<td>1296°C</td>
<td>12.6%</td>
<td>1600°C</td>
<td>0</td>
<td>~4×10⁻⁶</td>
<td>~2×10⁻²</td>
<td>~8×10⁻²</td>
<td>~9×10⁻²</td>
</tr>
<tr>
<td>5-4-1</td>
<td>UCO</td>
<td>1071°C</td>
<td>12.1%</td>
<td>1800°C</td>
<td>1-2</td>
<td>1.0×10⁻⁴</td>
<td>1.7×10⁻¹</td>
<td>6.0×10⁻³</td>
<td>2.3×10⁻³</td>
</tr>
<tr>
<td>2-3-2</td>
<td>UCO</td>
<td>1296°C</td>
<td>12.7%</td>
<td>1800°C</td>
<td>1-2</td>
<td>~3×10⁻⁴</td>
<td>~2×10⁻²</td>
<td>~2×10⁻³</td>
<td>TBD</td>
</tr>
<tr>
<td>3-4-1</td>
<td>UO₂</td>
<td>1013°C</td>
<td>10.6%</td>
<td>1700°C</td>
<td>High</td>
<td>~9×10⁻²</td>
<td>~8×10⁻²</td>
<td>TBD</td>
<td>TBD</td>
</tr>
</tbody>
</table>

a. Time-averaged, volume-averaged (TAVA) irradiation temperature is based on thermal calculations [Hawkes 2014, table 4].
b. Fissions per initial metal atom (FIMA) is based on physics calculations [Sterbentz 2014, table 6].
c. Safety test release fractions are based on the net amount of each isotope generated by the irradiation and do not account for any release during irradiation; 110mAg release during irradiation was significant from UCO compacts and must be considered to correctly interpret data.
d. Compact 3-4-1 safety test was terminated after 162 h at 1700°C due to high release, other tests were run for ~300 h at test temperature.

Europium release from UO₂ fuel compacts was lower than it was from UCO compacts. This result was expected based on the difference in kernel chemistry. Europium is present in the irradiated UO₂ fuel as an oxide, which is better retained in the kernel than the carbide form present in the UCO fuel. AGR-1 and AGR-2 PIE have shown that europium release from the compact during 1600°C safety testing is dominated by release of europium through intact SiC during irradiation. Europium released through intact SiC during irradiation is mostly retained in the OPyC and compact matrix graphite. When compacts are heated above the irradiation temperature during safety-testing, the europium in the compact matrix is gradually released from the compact. Because irradiation temperatures were ~150°C higher, AGR-2 Capsule 2 UCO compacts had more europium in the matrix at the start of the safety test than Capsule 5 compacts. This resulted in higher safety-test release from the Capsule 2 compacts. In addition, Compacts 5-4-1 and 2-3-2 exhibited higher safety-test release at 1800°C than similar compacts tested at 1600°C. This was at least partially due to significant release of europium through intact SiC during the safety test.

Trends for silver release are more difficult to deduce from the data in Table 6 because there is more variability in the pre-safety test silver inventories, both inside intact SiC and in the surrounding OPyC or matrix graphite. In all of the safety tests, silver in the OPyC or matrix graphite was rapidly released and mostly gone by the time the compact had been at the test temperature for 12 h. For the 1600°C safety tests, relatively little additional silver release was observed throughout the remainder of the tests; hence, silver release is more an issue of irradiation performance than performance at typical depressurization accident conditions and temperatures. However, there is evidence that additional silver is released through intact SiC during the 300-hour 1800°C safety testing. This resulted in a noticeably-elevated total silver release from Compact 5-4-1 (Table 6). A similarly-elevated total silver release was not detected from Compact 2-3-2 because the silver inventory at the start of the test was already depleted by release during irradiation (Compact 2-3-2 had been depleted to a fractional inventory of 7% at the end of irradiation, compared to a residual inventory of 86% in Compact 5-4-1). However, a slight increase in the silver

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release rate was detected in the latter half of the safety test that indicated that Compact 2-3-2 also exhibited measurable silver release through intact SiC at 1800°C.

Capsule 2 compacts exhibited higher 1600°C safety-test release of strontium than Capsule 5 compacts for the same reason as europium discussed above; increased strontium release rates in the latter half of the 1800°C tests indicated release through intact SiC in those tests similar to both europium and silver. Total strontium release was significant during safety tests of the UO2 fuel because of release from particles with failed SiC, as discussed below.

Cesium releases have been much higher during safety testing of the AGR-2 UO2 compacts than from safety testing of AGR-2 UCO compacts. All three AGR-2 UCO compacts tested at 1600°C exhibited zero SiC failure and very low cesium release as a result (measured cesium release from these tests was likely dominated by background contamination in the hot cell). Even at 1800°C, SiC failure was relatively low in the AGR-2 UCO compacts (1–2 particles out of ~3176 particles per compact). While initial cesium release from the two UO2 compacts safety tested at 1600°C was similar to the low values observed from the UCO fuel, after 50–100 h at 1600°C, cesium release increased dramatically. For the 1700°C safety test of UO2 Compact 3-4-1, the SiC failure fraction was so high that the safety test had to be terminated after 162 h at 1700°C to prevent excessive levels of radioisotope release. Preliminary measurement of the cesium release at the end of the Compact 3-4-1 test indicates ~9% of the total inventory of cesium was released, which could correspond to hundreds of particles (out of ~1543 per compact). Cesium release from AGR-2 UO2 fuel is connected with corrosion of the SiC layer by CO. The primary reason uranium carbide is mixed with uranium oxide in the UCO fuel kernel design is to control CO production to eliminate this deleterious effect (as well as to reduce internal pressure and eliminate the amoeba effect).

The IMGA was used to identify particles that released cesium in 1600°C safety-tested UO2 Compacts 3-3-2 and 3-4-2. This was discussed in detail in a previous quarterly report [INL/LTD-14-33591, Rev. 5]. Large holes were discovered in the SiC of several particles where it was exposed by an IPyC crack (Figure 21). Recent analysis has verified that these holes were caused by CO corrosion of the SiC. Evidence for CO corrosion includes characteristic SiC degradation and grain boundary attack where the SiC was exposed to CO gas by a crack through the IPyC (Figure 22), and Si depletion in the corroded zone accompanied by deposition of silicon oxide elsewhere in the particle (Figure 23). Survey with the IMGA of particles recovered from Compact 3-4-1 is planned to help enumerate the particles with failed SiC and identify particles for additional analysis with x-ray tomography and SEM.
Figure 21. (a) X-ray tomograph through midplane of 1600°C safety-tested AGR-2 UO₂ Compact 3-3-2 Particle SP04 showing acid leached kernel, and (b) 3D visualization of SiC surface showing cracks and large hole in SiC layer.

Figure 22. SEM micrographs showing CO corrosion of SiC adjacent to IPyC cracks in Compact 3-3-2 particles (IPyC cracks were verified by x-ray tomography but are not visible in the polish planes shown).
Figure 23. Micrographs obtained using secondary electron imaging (SEI) and back-scattered electron composition (BEC) showing a dense deposit located in the gap between buffer and IPyC in a particle that exhibited CO corrosion (top); and EDS maps of the deposit identifying primarily silicon and oxygen.

2.2.3.2 Radiation-induced Microstructural Changes in As-irradiated AGR-2 Particles. Materialographic sample preparation of irradiated TRISO particles typically involves mounting 2-3 particles in a miniature specimen holder custom designed for a Buehler Minimet 1000 grinder/polisher modified for hot cell use (Figure 24). Minimizing particle number simplifies preparation of high-quality polished sections and reduces sample activity to minimize radiation damage to the optical microscope and increase specimen handling options for advanced microscopies like SEM or TEM. A second version of the Minimet specimen-holder features a larger diameter specimen area and allows for 40-50 particles to be mounted in a single holder. This multi-particle holder allows high-quality polishing of larger, randomly-selected samples for better statistical sampling of the average particle microstructure.
Random samples from five as-irradiated AGR-2 compacts have been polished near midplane and examined with optical microscopy. Table 7 lists the irradiation conditions and number of sectioned particles with visible buffer fracture. The two Capsule 5 compacts both exhibited 86% visible buffer fracture in the plane of polish. By contrast, Compacts 2-2-3 and 6-3-3 exhibited very low buffer fracture. All four of these compacts were fabricated from randomly-riffled samples from a single coater batch; therefore, the observed performance is a function of the irradiation conditions. Compact 2-2-3 was irradiated at a similar dose to the Capsule 5 compacts, but at a higher temperature, which may have allowed more stress-relieving creep during irradiation. Compact 6-3-3 was irradiated to a lower dose and at a lower average temperature, so higher irradiation temperature is not the only influencing factor. Compact 3-3-1 was a different fuel batch, but had buffer deposited under similar conditions. No buffer fracture was seen in the particles from this compact. Figure 25 shows pictures of some of the multi-particle mounts. Of additional note is that no particles with fractures through the IPyC were observed in this survey; this is significant in that through-layer fracture of the IPyC has been observed to be a key contributing factor to SiC failure.

Table 7. Buffer fracture observed in AGR-2 as-irradiated particles.

<table>
<thead>
<tr>
<th>Compact ID</th>
<th>Fuel Type</th>
<th>Irradiation Temp. (TAVA)</th>
<th>Average Burnup (FIMA)</th>
<th>Fast Neutron Fluence (E&gt;0.18 MeV)</th>
<th>Number of Imaged Particles</th>
<th>Number with Visible Buffer Fracture</th>
<th>Fraction with Visible Buffer Fracture</th>
</tr>
</thead>
<tbody>
<tr>
<td>2-2-3</td>
<td>UCO</td>
<td>1261°C</td>
<td>10.8%</td>
<td>3.0×10^{25} n/m²</td>
<td>74</td>
<td>1</td>
<td>1.4%</td>
</tr>
<tr>
<td>5-2-3</td>
<td>UCO</td>
<td>1108°C</td>
<td>10.4%</td>
<td>3.0×10^{25} n/m²</td>
<td>88</td>
<td>76</td>
<td>86%</td>
</tr>
<tr>
<td>5-3-3</td>
<td>UCO</td>
<td>1093°C</td>
<td>10.1%</td>
<td>2.9×10^{25} n/m²</td>
<td>43</td>
<td>37</td>
<td>86%</td>
</tr>
<tr>
<td>6-3-3</td>
<td>UCO</td>
<td>1060°C</td>
<td>7.5%</td>
<td>2.1×10^{25} n/m²</td>
<td>44</td>
<td>0</td>
<td>0%</td>
</tr>
<tr>
<td>3-3-1</td>
<td>UO₂</td>
<td>1062°C</td>
<td>10.5%</td>
<td>3.5×10^{25} n/m²</td>
<td>35</td>
<td>0</td>
<td>0%</td>
</tr>
</tbody>
</table>
Figure 25. Multi-particle Minimet mounts of AGR-2 as-irradiated particles.
2.3 High Temperature Materials

2.3.1 ASME Code Qualification of Alloy 617

2.3.1.1 High Temperature Design. Two Code Cases are required in order to qualify Alloy 617 for use in construction of nuclear components for high temperature reactors, one for application in the temperature range where no creep occurs and the other at higher temperatures where time dependent allowables are necessary.

The low temperature Code Case, to allow use of Alloy 617 in Section III Division 5 Subsection HB Subpart A Class A and Subsection HC Subpart A Class B has been approved by Section II and III Committees and is under consideration for final approval by the Board on Nuclear Codes and Standards. This draft Code Case required a design fatigue curve, and low cycle fatigue testing was carried out at room temperature and 800°F to generate the data necessary to develop the design fatigue curve. Based on the measured properties, it appeared that either of the existing fatigue design curves I-9.2 or I-9.5 is appropriate for Alloy 617 and the Section III Working Group on Fatigue Strength recommended the use of I-9.5.

I-9.5 fatigue design curve includes a mean stress correction branch for cycles to failure above about 230,000. The mean stress correction is based on analysis using a Goodman diagram approach and it does not appear that any testing that included an applied mean stress was carried out.1

It is unknown what mean stress correction (if any) is appropriate for temperatures above 800°F but below the temperature where creep is anticipated to be significant.

Maximum Mean Stress Calculation

Data from elevated temperature low cycle fatigue testing of Alloy 617 by Yukawa2 were used to develop a fatigue design curve for the temperature range of 1000–1300°F (538–704°C). Yukawa fit the data using an elastic-plastic formulation

$$\Delta \varepsilon_t = A(N_f)^a + B(N_f)^b$$ (1)

where the first term represents inelastic strain and the second elastic strain. The fatigue data can also be fit to a form proposed by Diercks,3 similar to that of Langer,4 and typically used for room temperature data.

$$S_a = B N_f^{-b} + S_e$$ (2)

This method assumes an endurance limit, $S_e$, which is ~266 MPa according to the available data. The fitting parameters $b$ and $B$ are not the same variables used in Equation (1).

Although the fit for these two methods is similar in the range where there are data, the shape of the curve extrapolated to longer cycles is very different. Much of the extrapolated range falls outside the range of what is shown in the ASME B&PV Section III Division 5 Figure HBB-T-1420 (up to $10^6$ cycles), however the curves differ above about $5 \times 10^4$, and they have a significant effect on the maximum mean stress correction calculated.

A hypothetical mean stress curve was developed using the process used to develop the I-9.5 mean stress curve.1 A paper by O’Donnell and Langer5 shows the development of the maximum mean stress correction for Zr in detail and the method has been used for analysis of Alloy 617. The method utilizes the Goodman diagram with maximum mean stress equation

$$S_a' = S_a \left[ \frac{S_u-S_a}{S_u-S_b} \right] f \text{or } S_a < S_b$$ (3)

where $S_a$ is the effective yield strength, determined using a 0.2% offset on the cyclic stress-strain curve (a plot of the strain amplitude vs the stress amplitude from plastic fatigue tests at various strain levels), $S_u$ is 591.5 MPa, the average UTS determined for the INL reference plate at 550 and 700°C, and $S_a$ is the zero mean stress amplitude from the fatigue curve, a fictitious stress obtained by multiplying the strain by the
elastic modulus. *Equation (3)* is specifically for “metallic materials below the creep range.” The lower temperature limit for creep of Alloy 617 is estimated at around 1100°F (550-600°C), right in the middle of the temperature range this fatigue curve is meant to represent. Because of the rapid stress relaxation in the creep regime, it is argued that no mean stress correction is needed for the two high temperature fatigue curves.

Figure 26 compares the two methods of calculating a fatigue curve for the 1000-1300°F data, with the Langer method shown as dashed lines, and the Yukawa method shown as solid lines. In order to make the comparisons, conversions between stress and strain are made using the elastic modulus of the maximum temperature for each curve. The maximum mean stress corrections, calculated according to *Equation (1)*, are shown in purple. The grey area indicates the portion that would be visible in the Division 5 HBB Code Case after the 2 and 20 factors are applied. The cyclic yield strength, $S_y$, is shown as a red square symbol for each temperature. Two symbols are shown on the 1000-1300°F curve, one is plotted at 450 MPa, the same value used for the low temperature curve, and the other is 500 MPa, the value from the 1000 and 1300°F cyclic stress strain curves. A maximum mean stress correction assuming $S_y=500$ MPa is more severe (plotted in a lighter shade of purple). A lower UTS also results in more severe maximum mean stress corrections. No matter how they are calculated, these maximum mean stress curves are quite severe compared to the zero mean stress curve and also to the 1-9.5 maximum mean stress correction. Maximum mean stress testing was undertaken to check these calculated curves.

![Figure 26. Fatigue curves comparing two calculation methods, with the Langer method shown as dashed lines, and the Yukawa method shown as solid lines. The maximum mean stress corrections for 1000-1300°F are shown in purple. The grey area indicates the portion that would be visible in the Division 5 HBB Code Case.](image)

**Maximum Mean Stress Testing**

The maximum mean stress for a given total strain range was calculated using the approach described in the with grain (WG) fatigue strength document by Jones and O’Donnell\(^6\) according to

$$ (\sigma_{mean})_{max} = S_u - \frac{\Delta \sigma}{2} $$

where $S_u$ is the ultimate tensile strength, and $\Delta \sigma$ is the stress range taken from a zero mean stress (baseline) fatigue test for the same strain range. The material exhibits significant cyclic strain hardening,
so $\Delta\sigma$ was taken at a point where the peak stress reaches a constant value. Tensile strain is applied to the specimen until the maximum mean stress is achieved and then it is cycled to achieve the total strain range on the tensile loaded specimen. The strain is shown schematically in Figure 27.

![Mean strain vs. Time](image)

**Figure 27.** Schematic of strain as a function of time for a maximum mean stress test for an ideal case where there is no pause between the initial loading and the cyclic portion of the test.

A temperature of 550°C in the low end of the temperature range of interest (538–704°C) was selected to ensure that it was below the creep range. An even increment in Celsius was selected so that a previously run tensile test could be used to estimate the strain required to achieve maximum mean stress. At 550°C, the ultimate tensile strength, $S_u$, is 625.6 MPa.

Baseline and maximum mean stress tests were run for total strain ranges of 0.36 and 0.48%. The baseline test with a total strain range of 0.36% ($\pm 0.18\%$) was selected to fall on the 1000-1300°F fatigue curve (without 2 and 20 “knock-down” factors) at 100,000 cycles. This cycle life is short enough to complete in a reasonable amount of time, but long enough that the mean stress effect being checked would be substantial (far from where the max mean branch forks off in Figure 26) if the calculated branch is correct. The test ran longer than predicted by the design curve and was suspended at 550,000 cycles. The 0.48% strain range baseline test was selected at the same test temperature to try to get a test that failed in fewer cycles, but slightly past the beginning of the predicted mean stress correction shown in Figure 26. Failure occurred for the 0.48% baseline test near 39,000 cycles.

The mean stress that could actually be tested was limited by the range of the extensometer, $\pm 2.2\%$. The test is started with the extensometer at near maximum compression, $-2.1\%$. As the specimen is heated the expansion causes some decompression and the extensometer and the remaining range is available for preloading in addition to the tensile portion of cycling, about 3.5%.

For a 0.36% strain-range mean stress test, the total available strain for preloading of the specimen is about 3.3% (3.5–0.18). According to the tensile test, a strain of 3.3% corresponds to about 283 MPa. The actual stress reached during preloading for this test was 278.2 MPa and was held for about a minute before cycling began with a mean strain of 3.36%. The 0.48% strain-range mean stress test was loaded to about 260 MPa and cycled with a mean strain of 3.26%, with no hold between preloading and cycling. Both of the maximum mean stress tests followed the tensile test very closely during initial loading.

The loading and cyclic portions of the test are both run in strain control, and the strain as a function of time plots in Figure 28, a and b, show the strain behaved as expected. The stress response (Figure 28, c and d) shows that the maximum stress never increases much above that experienced during initial loading. Also, a substantial compressive stress is required to obtain the minimum strains prescribed by the cycling. The tests have very little mean stress, although they have significant mean strain. The mean stress of the 0.36% test has an average value of about 23 MPa, and the mean stress of the 0.48% test has...
an average value of 8-9 MPa. The baseline tests have a mean stress of approximately zero, as expected, although for all tests the mean stress exhibits significant variation.

Figure 28. (a-b) Strain and (c-d) stress plotted as a function of time through the first 3 cycles of the mean stress tests for the 0.36% (a,c) and 0.48% (b,d).

The 0.36% maximum mean stress test stopped after about 225,000 cycles, although the specimen failed in the grip region. The 0.48% maximum mean stress test failed after about 29,000 cycles, also outside of the gage section. The baseline and mean stress tests are plotted with the predicted fatigue curves in Figure 29. For both strain ranges tested, the maximum mean stress test does have a shorter life, but all four tests fall above the 1000–1300°F fatigue curve, indicating little to no mean stress effect for this temperature.
Figure 29. Fatigue tests at 500°C (1022°F) are plotted with the fatigue curves without 2 and 20 “knock-down” factors.

2.3.1.2 Resolution of Regulatory Issues. The effects of geometric discontinuities and multi-axial stress is being studied through use of notched specimens. V-notched specimens, which include a straight gauge portion are being used to study notch strengthening/weakening effects, and U-notch specimens (two types, small and large radii) to understand the effects of multi-axial stress.

V-notch Testing Results

Four conditions have been studied, with two replicate specimens at each condition. Temperature varied from 750 to 1000°C, and initial applied stress from 20 to 145 MPa. Seven tests have been performed. Applied stress was chosen such that the expect creep rupture life of the specimens would be approximately 1000-2000 hours. In each case, rupture occurred at the straight gauge rather than the notch, confirming that the material at these conditions is notch-strengthening. The creep rupture life of each specimen was consistent with results from standard straight-gauge creep specimens, as shown in Figure 30. Results from an eighth test were discarded because the temperature overshot the set point (1000°C), so no replicate test exists for the 1000°C test. As all other tests showed no variation in results, no new conclusions were expected from the replicate 1000°C test, so a new replicate test was not performed.
Figure 30. Initial applied stress vs. Larson Miller Parameter (a parameter that combines temperature and rupture life) showing that all V-notch tests fall within the scatter of typical straight gauge creep tests.

To better understand the mechanism of notch strengthening, plasticity of the specimens was studied using EBSD. Optical micrographs taken in the previous quarter showed deformation within the grains adjacent to the notch tip. Deformation can be better seen through use of EBSD local misorientation measurements. Small changes in orientation of the crystal lattice are indicative of the presence of dislocations, these local misorientations can be used to qualitatively examine deformation. These results are compared to the damage at the V-notch in Figure 2. Misorientation measurements were made from the tip of the V-notch (or edge of the specimen) towards the center of the specimen. The black line in Figure 31 represents an area with no creep damage (the thick shoulder of the specimen). The green line is representative of large amounts of creep damage (the straight gauge of a specimen, taken from a 750°C test interrupted just prior to failure, at 2200 hrs). Every other color represents a different V-notch test. Each test, with the exception of the 750°C test, show an increased level of plasticity until ~150–200 µm away from the notch tip. Little sign of damage is shown beyond 200 µm. The 750°C test shows no damage at the notch tip, following the black line nearly exactly. A grain within the specimen was at elevated levels of misorientation, though this is not believed to be a result of creep deformation. These results show that there is very little deformation occurring near the V-notch. The reason for the notch-strengthening properties of the material at these conditions is likely due to the restriction on plasticity imposed on the V-notch by the thicker regions of the specimen adjacent to the notch tip. These thicker regions impede the deformation of the specimen, and as a result, though stress is more concentrated at the V-notch, failure occurs in the straight gauge, where no such obstacle to deformation exists.
Figure 31. Graph showing local misorientation with respect to distance from either the edge of the specimen or the tip of the V-notch.

**Status of U-notch Testing**

U-notch testing has begun this quarter, with the first tests started at 800°C with an initial stress of 80 MPa, and the second set of tests at 750°C and an initial stress of 145 MPa. The data from these tests is shown in Figure 32; however, all of the U-notch tests are ongoing. The current levels of creep displacement are compared to the V-notch tests in Figure 32. The tests are in the early stages for the 750°C tests, so it is difficult to draw any conclusions yet; however, it is likely the small radius U-notch specimen will have a much longer creep rupture life than the straight gauge specimens at the same conditions, based on the low levels of creep it has experienced so far and the long creep life of the same specimen type at 800°C. It appears that there are notch-strengthening effects occurring for the small radius U-notch specimens.

Figure 32. U-notch creep displacement vs. time compared to V-notch measurements at the same condition – 750°C 145 MPa (left) and 800°C 80 MPa (right).
2.4 Graphite Development and Qualification

2.4.1 Materials—Graphite

2.4.1.1 Oxidation Effects on Graphite Performance. A small scoping study to determine the mechanical strength of graphite after oxidation was expanded to include not just mechanical strength changes but other material properties as well. Two grades of graphite, a small-grain and a large-grain grade, are being oxidized to 5 and 10% mass loss at three temperature levels (550, 650, and 750°C) to determine the changes in material properties in the two different grades. These data will be instrumental in predicting the behavior of graphite core components during both chronic and acute oxidation events.

Generally, oxidation reduces all material properties through the removal of material creating a lower density material. However, an interesting observation from these initial studies indicate that, similar to the rate of oxidation, the material property change is accelerated for small grained grades as seen in Figure 33 through Figure 35. As stated, these are initial results and a more thorough analysis in future studies will elucidate further results.

![Figure 33](image-url). Changes to stiffness modulus (Young’s Modulus) as a function of weight loss. Note accelerated changes to the fine grained IG-110.

![Figure 34](image-url). Changes to thermal diffusivity as a function of weight loss at different oxidizing temperatures.
2.4.1.2 **AGC Experimental Uncertainty.** Verification of INL’s MCNP Code responsible for calculating AGC dose levels continues. In order to determine the irradiation dose uncertainties for the AGC irradiations the potential errors and uncertainties propagated with the MCNP model must be understood. To ascertain these uncertainties, the ART graphite program will compare the activities calculated with an MCNP of a known reactor configuration and the actual measured activation values of a variety of flux wires. In order to perform this comparison between calculated and measured values an experiment must be performed within a reactor with a well-characterized core.

The MURR has agreed to collaborate with the INL to perform this comparison experiment. The MURR core is well characterized and is extremely similar to one of the fuel lobes in the ATR, providing a neutron condition similar to what the AGC capsules experience in the flux trap. MURR has agreed to provide their MCNP codes for their reactor and to irradiate multiple flux-wire configurations. Details of the experiment and design will be established during a visit to MURR planned for January 2017.

2.4.1.3 **Baseline Material Property Testing.** Baseline material property testing activities are focused on completing the remaining material property testing for five partial billets. This is approximately 75% complete. Completion of this milestone will result in capturing and qualifying the remaining mechanical and physical properties from five individual billets of graphite into the Next Generation Nuclear Plant (NGNP) Data Management and Analysis System (NDMAS) vault. Figure 36 shows typical flexural strength data for the five billets. The data will be verified by running suites of tests to identify errors and anomalies. These tests included conformance with specifications, correct use of data in calculations, and anomalous variation among measurements.
Collaborative Fundamental Irradiation Damage Studies. Irradiation of the SAM-1 capsules was completed and disassembly occurred at International Isotopes, Inc. (I3), in Idaho Falls, ID. Graphite samples within the SAM-1 capsule are intended for low-dose fundamental studies in collaboration with national and international research laboratories. Both polycrystalline and HOPG specimens were irradiated with the HOPG specimens being high purity TEM-sized specimens to assist in the characterization of irradiation damage. Polycrystalline graphite samples were composed of 4 different ART graphite grades; PCEA, IG-110, 2114, and NBG-18. These very-low-dose specimens will assist in determining the onset of material-property changes resulting from irradiation damage without significant microstructural change.

The three expected dose levels for all samples are 0.001, 0.004, and 0.02 dpa and were irradiated over temperature ranges of 600, 900, 1200°C. INL expects to receive the capsules by January 2017 and will begin irradiated-material analysis at that time. The specimens will be shared with researchers from the Universities of Manchester and Surrey (U.K.), Boise State University (BSU), and ORNL.

2.4.1.4 Chronic Oxidation Studies and Model Development. Chronic oxidation by moisture studies continues with new Mersen 2114 specimens. All Mersen 2114 specimens have been machined and are ready for testing. A total number of 250-300 data points will be collected for statistically significant results from nonlinear data analysis. Each data point is an oxidation-rate value measured at certain conditions (water vapor pressure, temperature, and added hydrogen partial pressure). The tests started with measurement in helium containing water vapor only and the first results were already obtained for values between 25 and 500 Pa water partial pressure, and temperatures between 800 and 1100°C. A continuous run takes about 30 hours, with constant values for $P_{H_2O}$ and $P_{H_2}$, and incremental increase of temperature in steps of 50 degrees. Additionally, the machining of IG-110 and Mersen 2114 specimens specifically for effective diffusivity measurements of water vapor in helium has been completed.

The weight losses caused by oxidation are measured continuously, and oxidation rates are calculated from the slope of weight loss versus time, on constant temperature segments of 3 hours each. The rates are normalized to the initial weight of the specimen at the beginning of each constant temperature segment. In parallel, the gas composition in the microbalance is monitored by the mass spectrometer. Figure 37 shows an example of experimental data corresponding to slow (chronic) oxidation of specimen M-2 (Mersen) in helium with 200 Pa of water partial pressure ($P_{H_2O}$).
Figure 37. Example of analytical data obtained from thermogravimetric and mass spectrometric analysis during accelerated oxidation tests of graphite in helium with low water concentration. A Mersen 2114 specimen 25 × 5 mm ø; initial weight 901.27 mg) was oxidized in 1.5 L/min He containing 200 Pa H2O. After outgassing at 1200°C in pure He, the temperature (red line) was varied from 800 to 1100°C in He + H2O mixture and the oxidation rates were calculated from the slope of weight trace (green line). Simultaneously, the mass spectrometer signals for H2O (mass 18), OH (mass 17), CO (mass 28), H2 (mass 2), O2 (mass 32) and CO2 (mass 44) were recorded. The integrated areas of CO and H2 signals were also calculated (yellow color). Note the logarithmic scale for mass spectrometer signals.

2.4.1.5 Image Analysis Development for Microstructure Evolution. Non-destructive imaging tools to analyze the internal graphite microstructure continue to be developed. These imaging tools and investigative capability have contributed significantly to our ability to determine changes to the internal microstructure during oxidation, creep, and irradiation. The intent of the program is to assist in the development of predictive models for all material properties that rely upon the microstructure changes such as strength, oxidation rate, fracture behavior, irradiation creep, irradiation dimensional change, and pore development.

Recent developments include new skeletonization-image-analysis techniques that can identify major pore structures within the microstructure. These linked pore structures, or influence zones, are assumed to be responsible for many of the unique behaviors of graphite from fracture behavior to gas diffusion to pore generation during irradiation. Examples of the usefulness of this technique are presented in Figure 39.
through Figure 41. This new analysis tool will allow for quantitative inferences about the pore structures of graphite.

Figure 38. Skeleton image of IG-110 porosity split into different regions by “influence zone.”

Figure 39. View of the black region from previous figure. The red spots are places where the skeleton is connected to the previous color regions.
Figure 40. Primary, secondary, tertiary branches highlighted within the black region influence zone skeleton. The remaining black branches are much smaller and have fewer connections with other branches throughout the zone.

Figure 41. Highlights and displays for only the primary, secondary, and tertiary branches of the black influence zone region.
2.4.1.6 **Graphite Sample Preparation Glovebox.** Fabrication of the graphite sample preparation glove box (Figure 42) is complete. Delivery of the system, which will contain a diamond slow-speed wet saw, polishing wheel, and hole saw for preparation of irradiated graphite specimens to the INL Carbon Characterization Lab, is expected in early January 2017.

![Figure 42. Factory testing of graphite sample preparation glovebox. Includes helium leak testing, fire suppression system, and other operational features of the glovebox.](image)

2.4.1.7 **Graphite Sample Preparation Glovebox.** ORNL continues irradiation damage studies utilizing XRD and SANS techniques to ascertain the changes to graphite crystal structure changes. HOPG and polycrystalline specimens from AGC-1 irradiation are being analyzed (Table 1). An electronic letter report, “X-ray Diffraction (XRD) and Angle Neutron Scattering (SANS) annealing studies,” was submitted to INL to fulfill a milestone for ORNL FY16 MPO (SOW-12688).

Table 8. List of HOPG samples and the various annealing treatments and XRD scans that have been performed to each sample.

<table>
<thead>
<tr>
<th>Sample</th>
<th>$T_{irr}$ (°C)</th>
<th>Dose (dpa)</th>
<th>After Irradiation</th>
<th>1st Annealing @ 700°C</th>
<th>2nd Annealing @ 900°C</th>
<th>3rd Annealing @ 1200°C</th>
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</thead>
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<tr>
<td>HOPG-unir radiated</td>
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<td>—</td>
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<td>✓</td>
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<td>✓</td>
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<tr>
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2.4.2 **Advanced Graphite Creep Irradiations**
2.4.2.1 AGC-3 PIE Testing. AGC-3 PIE testing is 85% complete. AGC-3 samples were irradiated at a temperature of 800°C and over an approximate dose range of 1 – 3.5 dpa.

2.4.2.2 AGC Experimental Uncertainty. Verification of INL’s MCNP Code responsible for calculating AGC dose levels continues. In order to determine the irradiation dose uncertainties for the AGC irradiations the potential errors and uncertainties propagated with the MCNP model must be understood. To ascertain these uncertainties, the ART graphite program will compare the activities calculated with an MCNP of a known reactor configuration and the actual measured activation values of a variety of flux wires. In order to perform this comparison between calculated and measured values, an experiment must be performed within a reactor with a well-characterized core.

The MURR has agreed to collaborate with the INL to perform this comparison experiment. The MURR core is well characterized and is extremely similar to one of the fuel lobes in the ATR, providing a neutron condition similar to what the AGC capsules experience in the flux trap. MURR has agreed to provide the MCNP codes for their reactor and to irradiate multiple flux wire configurations. Details of the experiment and design will be established during a visit to MURR planned for January 2017.

2.5 Methods

2.5.1 Experiments and Computational Fluid Dynamics Validation

2.5.1.1 High-Temperature Test Facility. More water was discovered when heating up the HTTF after re-installing the circulator. The source of the water is believed to be the core ceramic. The ceramic is hygroscopic, holding up to 3% water by weight (estimated to be as much as 60 gallons), and the system was open to atmosphere for five months. The two potential sources of water leakage into the primary coolant system are the steam generator secondary (through tube leakage) and the cooling water to the gas circulator. The steam generator secondary has been dry throughout this time, and subsequent pressure testing of the circulator cooling system revealed no leakage. When the core ceramic was heated, water was forced out as steam, which then condensed on the cooler metal walls in the primary coolant system. The pressure vessel lower head has the lowest temperature, and the circulator is at the low point in the system, where water condensed in the cold leg portion of the system would collect. The water in the lower head shorted out the heaters again. After drying, the heaters are working better than ever. Heatup procedures are being modified to minimize condensation in the piping. It is also now known how the instrumentation responds to the steaming, so that the problems can be identified should it occur again. Drain valves on the vessel lower head and circulator housing are available to remove any collected liquid.

2.5.2 Physics Methods

Accurate, high-resolution full core model of prismatic high temperature reactors for burnup and transient analysis remains a challenge for both reactor analysts and computers. The cores are spatially large, but some features are quite small (e.g., TRISO particles, burnable poison pellets). The treatment of both neutron scattering in graphite and resonance capture are complex and not adequately captured using the methods traditionally used in HTGRs and commonly used in light water reactor (LWR)s. For transient analysis, temperature feedback is a function of the TRISO fuel form, but full core models cannot resolve phenomena at this scale without careful averaging over space and energy. For burnup calculations, a second (pin) level of heterogeneity must be resolved to capture the local effects of burnable poisons and fuel pins while accurately propagating their effects through and between blocks.

The Organization for Economic Cooperation and Development (OECD) Modular High Temperature Gas Cooled Reactor (MHTGR) Benchmark is being used at INL to drive the development of full core and transient models and for code-to-code verification against other methods. The IAEA CRP on HTGR Uncertainty Analysis in Modeling (UAM) is likewise being developed to compare the propagation of uncertainties in lattice and full-core modeling. The tools developed in these activities are being used in related HTGR-support work—for example, in the joint modeling of the High Temperature Test Reactor
(HTTR) with Japan Atomic Energy Agency (JAEA) under the ART Civil Nuclear Energy Research and Development Working Group funding, as well the design of the Advanced High Temperature Test Reactor. Existing lattice tools (e.g., DRAGON, HELIOIS, SERPENT, Standardized Computer Analyses for Licensing Evaluation [SCALE]) have different strengths and weaknesses, and concurrent modeling with these tools helps to identify code and model discrepancies, sensitivities, and best practices.

2.5.2.1 **OECD/NEA MHTGR-350 Benchmark.** INL is leading an international prismatic HTGR benchmark for comparing and evaluating lattice and core analysis codes, based on the 350 MW MHTGR design information obtained from General Atomics. Similar to the PBMR-400 Transient Benchmark sponsored by the OECD’s Nuclear Energy Agency, this OECD/NEA benchmark is a multiyear project (2013-2017) that will yield a set of reference steady-state and lattice problems that can be used by DOE, NRC, and vendors to assess their codes. The OECD/NEA sponsorship of the prismatic benchmark is valuable because it attracted international participation and leverages benchmark expertise and publishing apparatus already used for other LWR, sodium fast reactor, and high temperature reactor development programs.

This benchmark activity is now nearing completion, with the data submitted for Phases I and III currently being collected and compared. A comparison report on Exercise I of Phase I was released to the participants for review at the end of September 2016, and will be released as an OECD/NEA publication in 2018. The comparisons of Exercises 2 and 3 of Phase I, as well as the results for Phase III, will be completed in 2017 and released to the OECD and DOE as an L3 milestone report by the end of September. The Phase II transient exercises were calculated by the INL team and reported at the end of FY-15 in INL/EXT-15-36307. Since no other results are expected for Phase II of the benchmark, an OECD comparison report will not be created for these exercises.

2.5.2.2 **IAEA CRP on HTGR Uncertainties.** Best-estimate-plus-uncertainty (BEPU) analysis of reactors is today not only broadly accepted, but in many cases is replacing the traditional conservative (stacked uncertainty) method for safety and licensing analysis. The use of a more fundamental methodology is also consistent with reliable high-fidelity physics models and robust, efficient, and accurate codes available today. To facilitate uncertainty-analysis applications, a comprehensive approach and methodology must be developed and applied. HTGRs have their own peculiarities, including coated-particle design, large graphite quantities, different materials, and high temperatures that require simulation techniques not utilized in LWR analysis. The IAEA launched the CRP on the HTGR UAM in 2013 to study uncertainty propagation in the HTGR-analysis chain. Two benchmark problems are defined, with the prismatic design represented by the General Atomics (GA) MHTGR-350 and a 250 MW modular pebble-bed design similar to the HTR-PM (INET, China). INL is leading the prismatic-reactor problem specification of Phases I-III. The IAEA recently approved an extension of this CRP to 2019. It currently involves 1 FTE of INL support and a 2017 sub-contract with NCSU for $50,000.

The 2017 Research Coordination Meeting (RCM) will be held in Vienna at the IAEA Headquarters on May 22-25. Fifteen participants from the USA, China, South Korea, South Africa, and Germany are currently scheduled to attend the RCM to present their latest results for Phase I of the pebble-bed and prismatic benchmarks. In the first quarter of FY-17, the focus at INL was on the definition of the depletion exercises for Phase II. The benchmark specifications for these exercises will be released to the participant group by the end of March 2017. A few of the preliminary depletion results obtained by the NCSU/INL team up to the end of December are discussed in the section below.

The main purpose of the lattice problems defined as Exercises I-2a and I-2b (fresh and depleted single blocks) and Exercise I-2c (supercell) is the development of cross-section libraries for use in Phase II depletion and core steady-state calculations and the coupled propagation of cross-section and manufacturing uncertainties from Phase I to II. As a preliminary study, SCALE6.2/TRITON depletion calculations were run without the SCALE6.2/SAMPLER perturbation sequence to establish the approach that needs to be taken in the depletion sequence. An analysis of the statistical convergence of the three

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depletion models (a single fresh block, a supercell consisting of 7 blocks and a 1/3rd core section) is presented here. An investigation of the various depletion options available in SCALE 6.2 is also included.

The single block, supercell, and core SCALE/KENO-VI models are shown in Figure 43, Figure 44 and Figure 45. The homogenized burnable-poison compacts (shown as purple dots in Figure 43) include burnable-poison particles smeared out with block graphite material. The homogenized-fuel-compacts (blue dots in Figure 43) region is obtained by homogenizing the TRISO fuel particles with the fuel graphite matrix material.

Figure 43. KENO-VI two-dimensional plot of the single block model.

Figure 44. KENO-VI two-dimensional plot of the super cell model.
The first parameter evaluated is the statistical convergence achieved as a function of the number of neutrons sampled. The total number of neutrons considered in the KENO-VI Monte Carlo simulation is defined as the number of neutrons per generation multiplied by the number of neutron generations. The default value in KENO-VI is 203,000 neutrons (203 generations, 1,000 neutrons per generation and 3 skipped generations for statistical reasons). For this study, the number of generations skipped was increased to 20 to improve the statistical error. The eigenvalue standard deviation (in pcm) was evaluated at the beginning and end of the depletion sequence for each of the three models (listed as beginning of cycle [BOC] and end of cycle [EOC] in Figure 46 and Figure 47), varying the total number of neutrons from 100,000 to 2,000,000 neutrons. The end of the depletion sequence is chosen to represent a burnup equal to 107.5 GWd/MTHM (corresponding to the MHTGR-350 operating for 4.5 years at 350 MW). The standard deviation of the eigenvalue as a function of the total number of neutrons is shown in Figure 46. The simulation execution time, as a function of the total number of neutrons sampled for the single block, supercell and core models, is presented in Figure 47. These calculations were performed on the INL Falcon machine, using only 1 processor because the SCALE/TRITION depletion sequence has not yet been parallelized by the SCALE developer team. This sensitivity study shows that for all three models, a standard deviation of less than 75 pcm can be obtained using a total population of more than ~1,000,000 neutrons. This choice will allow the simulations to complete in less than 12 hours. It is furthermore interesting to note in Figure 47 that the supercell model is almost as expensive as the 1/3 core model in terms of simulation time, in spite of the latter containing 22 fuel blocks compared to the 7 fuel blocks of the supercell model.
In the second sensitivity study, the two different types of depletion options available in SCALE/TRITON offers were evaluated in terms of the effect this choice has on the fuel and burnable poison (BP) nuclide inventories and the eigenvalue.

- The first depletion option allows the user to normalize the power to the specific power defined in one or more of the model materials. For example, in this MHTGR-350 case the user can set the specific power equal to 65.9 MW/MTHM in the homogenized fuel material. If this option is selected, no power is generated in any other material, implying that $n-\gamma$ reactions are neglected in the BP compacts, for example.
• The second depletion option allows the user to choose between deplete-by-power or deplete-by-flux. (The SCALE 6.2 User Manual note that most applications would perform best using the depletion-by-flux option, but also to test these other options). If the user selects deplete-by-power, SCALE/ORIGEN uses the specific power input by the user and defines the appropriate flux at each time step to match the user-input specific power. This option is more suitable for the fuel regions because the neutron flux can vary over a time step and, hence account for isotope inventory changes as a function of time. The deplete-by-flux option is more suitable in regions where the isotope inventory changes sufficiently to induce significant changes in the local flux levels, such as BP regions.

The cases investigated here are numbered as follow:

• Case 1: The system-averaged power matches the specific power input by the user. Both the fuel and BP materials are depleted by power.

• Case 2: The system-averaged power matches the specific power input by the user only in the fuel material, and both fuel and BP materials are depleted by power.

• Case 3: The system-averaged power matches the specific power input by the user. The fuel material is depleted by power, and the BP material is depleted by flux.

• Case 4: The system-averaged power matches the specific power input by the user only in the fuel material. The fuel material is depleted by power, and the BP material is depleted by flux.

All these depletion options are sensitive to the refinement of the burn step time step length. The effect of choosing large burn steps (as given in Table 9) are shown for the single block, super cell and core model for the B-10 (Figure 48) and U-235 (Figure 49) nuclide number densities and the eigenvalue (Figure 50). It can be seen that the boron depletion is slower if the depletion-by-flux is turned on, but the U-235 depletion does not change significantly as a function of the depletion options.

Table 9. Broad burn steps.

<table>
<thead>
<tr>
<th>Burn time step (d)</th>
<th>Cumulated time (d)</th>
<th>Cumulative burn-up (GWd/MTHM)</th>
</tr>
</thead>
<tbody>
<tr>
<td>1.5</td>
<td>1.5</td>
<td>100</td>
</tr>
<tr>
<td>0.8</td>
<td>2.3</td>
<td>150</td>
</tr>
<tr>
<td>2.3</td>
<td>4.6</td>
<td>301</td>
</tr>
<tr>
<td>10.6</td>
<td>15.2</td>
<td>1,000</td>
</tr>
<tr>
<td>15.2</td>
<td>30.4</td>
<td>2,000</td>
</tr>
<tr>
<td>45.5</td>
<td>75.9</td>
<td>5,000</td>
</tr>
<tr>
<td>75.9</td>
<td>151.7</td>
<td>10,000</td>
</tr>
<tr>
<td>151.8</td>
<td>303.5</td>
<td>20,000</td>
</tr>
<tr>
<td>151.8</td>
<td>455.2</td>
<td>30,000</td>
</tr>
<tr>
<td>151.8</td>
<td>607.0</td>
<td>40,001</td>
</tr>
<tr>
<td>75.9</td>
<td>682.9</td>
<td>45,000</td>
</tr>
<tr>
<td>379.4</td>
<td>1,062.2</td>
<td>70,001</td>
</tr>
<tr>
<td>379.4</td>
<td>1,441.6</td>
<td>95,001</td>
</tr>
</tbody>
</table>
Figure 48. B-10 number density for the four depletion options selected and using broad burn steps.
Figure 49. U-235 number density for the four depletion options selected and using broad burn steps.
The infinite multiplication factor shows significant differences: up to 5,000 pcm depending on the depletion options. The choice of power normalization based on the fuel material or the entire set of materials does not seem to affect the number densities and k-infinity values (comparison of Case 1 and Case 2), but for these large burn-up steps, the selection of the depletion-by-flux or depletion-by-power has a significant impact.

If the burnup steps are refined, the B-10 number densities (Figure 51) show much better agreement for all 3 models, but the infinite multiplication factor (Figure 52) still shows a disagreement, depending on the depletion method selected. The mean difference between case 1 and 3, along with the standard deviation of the differences, is presented in Table 10. This table shows that, on average over the cycle, the supercell model produced larger discrepancies than the single-block model, depending on the type of KENO-VI depletion option, with an average difference of 466 pcm and a large standard deviation (325.5 pcm). At certain burn steps, the difference goes up to 1,000 pcm if the mean plus two standard deviations is taken into account. These results can be explained by the way the flux is treated over the burn steps in non-fissile and fissile regions. As the flux is more thermalized in the supercell model, the thermal neutron capture in the BP compacts occurs faster than in the harder spectrum of the single-block model. The flux treatment, therefore, represents an important parameter in these HTGR problems, leading to the conclusion that the depletion-by-flux option represents the best SCALE/TRITION approach for the depletion simulations.
Figure 51. B-10 number density for the four depletion options selected and using refined burn steps.
Figure 52. $k$-infinity for the four depletion options selected and using refined burn steps.

Table 10. Difference in the multiplication factor between depletion mode 1 and 3 in the SB, SC, and Core models.

<table>
<thead>
<tr>
<th>Model</th>
<th>Mean difference between depletion mode 1 and 3 (pcm)</th>
<th>Standard deviation of the difference between depletion mode 1 and 3 (pcm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Single Block</td>
<td>220</td>
<td>165</td>
</tr>
<tr>
<td>Super cell</td>
<td>466</td>
<td>326</td>
</tr>
</tbody>
</table>

### 3. 90-DAY LOOK AHEAD

#### 3.1 Important Activities

- The remaining two TRISO-particle coating batches will be completed, all five batches will be characterized, and a TRISO lot will be blended for the AGR-5/6/7 irradiation experiments.
- Compacts will be formed at 10% PF as a demonstration of capabilities and limitations for forming fuel compacts at low packing fractions.
- A readiness review will be conducted at BWXT to assess readiness to commence fuel compact fabrication.
• Fabrication of low-enriched uranium compacts at 40% and 25% packing fractions will be completed for the AGR-5/6/7 irradiation experiments.
• A criticality control area (CCA) will be established in the Test Train Assembly Facility (TTAF) to enable AGR-5/6/7 fuel compacts to be received from BWXT and stored at the INL.
• Re-irradiate in the Neutron Radiography Reactor (NRAD) and safety test in the fuel accident condition simulator (FACS) furnace at least three crushed AGR-2 Compact 6-4-1 particles.
• Complete qualification and installation of AGR-3/4 ring sampling equipment at HFEF.
• Initiate physical sampling of AGR-3/4 inner and outer rings at HFEF.
• Complete replacement of FACS furnace thermocouples.
• Complete AGR-3/4 fluence wire analysis at PNNL.
• Complete installation and testing of the out-of-cell gamma counting station in HFEF.
• Continue STEM analysis on particle AGR2-223-RS06 in January and February 2017
• Issue report INL/LTD-16-39740, by “T. M. Lillo, I. J. van Rooyen, H. Wen, J. A. Aguiar, Supplemental PED Advanced Microscopy Data from Microstructural Analyses of AGR-1 TRISO Particles.”

3.1.2 High Temperature Materials
• Staff from INL will participate in American Society of Mechanical Engineers (ASME) Code Week in Atlanta, GA in February 2017.

3.1.3 Graphite Development and Qualification
• Release and transfer raw data from the first billet of “new batch” PCEA to the Nuclear Data Management and Analysis System (NDMAS).
• Distributing specimens and custom fixtures to the participating laboratories that will be collecting round-robin data for the ASTM International test standard on Brazilian disc tensile splitting strength.
• Commence tensile testing on three grades of graphite with reduced gauge sections to evaluate possible size effects associated with fracture in reduced volumes under stress.
• Attend a review meeting at Boise State University January 2017, to discuss the SAM-1 irradiation specimens, fundamental studies on irradiation damage in graphite, and BSU’s new EPSCoR project on nuclear graphite.
• Take delivery of and commence installation sample preparation glovebox in the Carbon Characterization Laboratory in early January.
• Distribute Brazilian disc tensile splitting (split-disc) strength specimens and custom fixtures to the participating laboratories in support of the ASTM International round robin testing.
• Attend a kick-off meeting with MURR February 2017, to finalize the flux-wire irradiation experiment, irradiation schedule, and future collaborations related to this material irradiation experiment.
• Attend ASME Boiler Code week at Atlanta, GA, 12-17 February 2017.
• Attend the Materials Project Management Board (PMB) meeting for the Generation IV International Forum at JAEA in March 2017.
3.1.4 Methods

- Resume shakedown testing at the HTTF.
- Establish the HTTF baseline configuration and initiate matrix testing.
- Distribute the IAEA CRP on HTGR Uncertainties Phase II depletion specifications to the participant working group.
- Present the latest status of the modular high temperature gas-cooled reactor (MHTGR)-350 benchmark to the Organization for Economic Cooperation and Development (OECD)/Nuclear Energy Agency (NEA) Expert Group on Uncertainty Analysis in Modelling.

4. REFERENCES


