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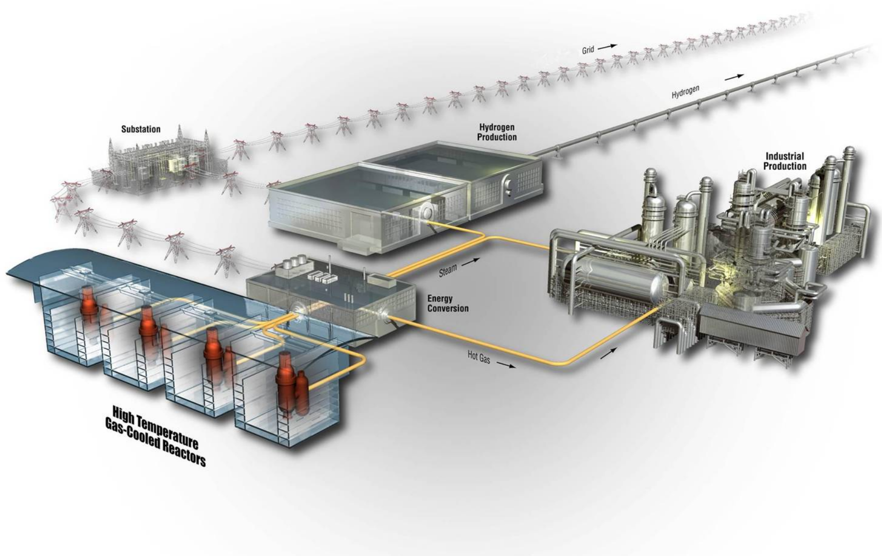
Project No. 23841

AGR-2 Irradiation Experiment Test Plan



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AGR-2 Irradiation Experiment Test Plan	Identifier: PLN-3798 Revision: 1 Effective Date: 10/05/2011 Page: i of viii

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REVISION LOG

Rev.	Date	Affected Pages	Revision Description
0	3/30/11	All	Initial issue of the AGR-2 Irradiation Experiment Test Plan
1	10/05/2011	iii, 2, 4, 6, 11, 14, 28, 29, 30, 31, 33	Grammatical and minor corrections
		3	Fuel design history
		7	Sphericity ratios (Table 3-4)
		7	Standard deviations and footnotes (Table 3-4)
		9	Contamination fraction (Table 3-5)
		13	TC / gas line positions on Figure 3-3
		14	Melt wires and TCs (Table 3-7)
		15	Capsule 6 thermal shielding and sweep gas flow
		16	Thru tubes description
		22	Figure 4-1 / Temperature in Capsule 6
		23	Figure 4-2 + new Figures 4-3 and 4-4
		28	Irradiation test acceleration
		29	Volume-average temperatures (Table 4-1)
		34	Irradiation / Post-irradiation
		39	Appendix

SUMMARY

This document presents the current state of planning for the AGR-2 irradiation experiment, the second of eight planned irradiations for the Advanced Gas Reactor (AGR) Fuel Development and Qualification Program. Funding for this program is provided by the U.S. Department of Energy (DOE) as part of the Next-Generation Nuclear Plant (NGNP) Project. The objectives of the AGR-2 experiment are:

1. Irradiate UCO and UO_2 fuel produced in a large coater. Fuel attributes are based on results obtained to date from the AGR-1 test and other project activities.

2. Provide irradiated fuel samples for post-irradiation examination and safety testing.

3. Support the development of an understanding of the relationship between fuel fabrication processes, fuel product properties, and irradiation performance.

In order to achieve the test objectives, the AGR-2 experiment will be irradiated in the B-12 position of the Advanced Test Reactor at Idaho National Laboratory. The test contains six independently controlled and monitored capsules. Each capsule contains a single type (UCO or UO_2) of the AGR coated fuel. The irradiation is planned for 600 effective full power days (approximately 2.75 calendar years) with a time-average, peak temperature ranging between approximately 1140 and 1400 °C depending on the specific capsule. Average fuel burnup, for the entire test, will be greater than 10% FIMA, and the fuel will experience fast neutron fluences between approximately 2.2 and 4.0×10^{25} n/m^2 (E>0.18 MeV).

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ACRONYMS

AGR	Advanced Gas Reactor
AGR-1	first irradiation test of the AGR program
AGR-2	second irradiation test of the AGR program
ASME	American Society of Mechanical Engineer
ATR	Advanced Test Reactor (INL)
AWS	American Welding Society
BAF	Bacon Anisotropy Factor
BOL	Beginning Of Life
BWXT	BWX Technologies
DOE	Department of Energy (U.S.)
EFPD	Effective Full Power Days
FIMA	Fissions per Initial heavy Metal Atom
FPMS	Fission Product Monitoring System
GT-MHR	Gas-Turbine Modular Helium Reactor
HPGe	Hyper Pure Germanium
INL	Idaho National Laboratory
IPyC	Inner Pyrolytic Carbon
LEU	Low Enriched Uranium
NDMAS	NGNP Data Management and Analysis System
NGNP	Next Generation Nuclear Plant
OPyC	Outer Pyrolytic Carbon
ORNL	Oak Ridge National Laboratory
PALM	Powered Axial Locator Mechanism
PIE	Post-Irradiation Examination
R/B	Release rate to Birth rate ratio
RMS	Root Mean Square
SiC	Silicon Carbide
TAVA	Time-Average Volume-Average
TC	ThermoCouple
TRISO	tristructural-isotropic
UCO	uranium oxycarbide

UO_2	uranium dioxide
VHTR	Very-High-Temperature gas-cooled Reactor

1. Introduction

Several fuel and material irradiation experiments are planned for the Advanced Gas Reactor (AGR) Fuel Development and Qualification Program which supports the development of the Very-High-Temperature gas-cooled Reactor (VHTR) under the Next-Generation Nuclear Plant (NGNP) Project. The goals of these experiments (Simonds, 2010) are to provide irradiation performance data to support fuel process development, qualify fuel for normal operating conditions, support development and validation of fuel performance and fission product transport models and codes, and provide irradiated fuel and materials for post-irradiation examination (PIE) and safety testing. AGR-2 is the second in this series of planned experiments to test tristuctural-isotropic (TRISO)-coated, low enriched uranium (LEU) oxycarbide fuel. This experiment is intended to demonstrate the performance of UCO (uranium oxycarbide) and UO_2 (uranium dioxide) fuel produced in a large coater.

This document presents the conceptual planning to implement requirements from the Technical Program Plan (Simonds, 2010) and the Irradiation Test Specification (Maki, 2010) for the AGR-2 experiment. Following this introduction, the test objectives and experimental approach are outlined in Section 2; descriptions of the test articles, test train, and the fission product monitoring system are presented in Section 3; anticipated irradiation conditions including temperature, burnup, and fast neutron fluence are presented in Section 4; measurements associated with test conduct are described in Section 5; significant operational procedures that apply to AGR-2 are briefly described in Section 6; safety and quality assurance issues are outlined in Section 7; program constraints and test schedule are listed in Section 8; and references are presented in Section 9. Requirements and planning associated with PIE and safety testing of the AGR-2 test articles will be presented elsewhere.

2. BACKGROUND

2.1 Test Objectives

As defined in the AGR Technical Program Plan (Simonds, 2010), the objectives of the AGR-2 experiment are to:

1. Irradiate UCO and UO_2 fuel produced in a large coater. Fuel attributes are based on results obtained from the AGR-1 test and other project activities.

2. Provide irradiated fuel samples for PIE and safety testing.

3. Support the development of an understanding of the relationship between fuel fabrication processes, fuel product properties, and irradiation performance.

The primary objective of the test is to irradiate both UCO and UO_2 TRISO fuel produced from prototypic scale equipment to obtain normal operation and accident condition fuel performance data. The UCO compacts will be subjected to a range of burnups and temperatures typical of anticipated prismatic reactor service conditions in three capsules. The test train also includes compacts containing UO_2 particles produced independently by the United States, South Africa, and France in three separate capsules. The range of burnups and temperatures in these capsules will be typical of anticipated pebble bed reactor service conditions. The test objectives listed here pertain only to U.S. produced fuel.

This test train will provide irradiated fuel performance data and irradiated fuel samples for safety testing and PIE to support the development of a fundamental understanding of the relationship between the fuel fabrication process, as-fabricated fuel properties, and normal operation and accident condition performance.

2.2 Experiment Approach

To achieve the test objectives outlined above, AGR-2 will be irradiated in the B-12 position of the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL). A core cross section indicating this location is displayed in Figure 2-1. Preliminary physics calculations (Chang, 2002) have shown that anticipated VHTR end of irradiation conditions (burnup up to about 20% FIMA and maximum fast neutron fluence of 5×10^{25} n/m^2, E>0.18 MeV) are best matched by the conditions obtained from irradiation in these large B positions, such as B-12, of the ATR. In addition, the rate of burnup and fast fluence accumulation, or acceleration, in these positions is less than three times that expected in the VHTR. Past U.S. and German experience indicates that by keeping the acceleration factor under three, an irradiation test is more prototypic of an actual reactor irradiation (Petti et al., 2002).

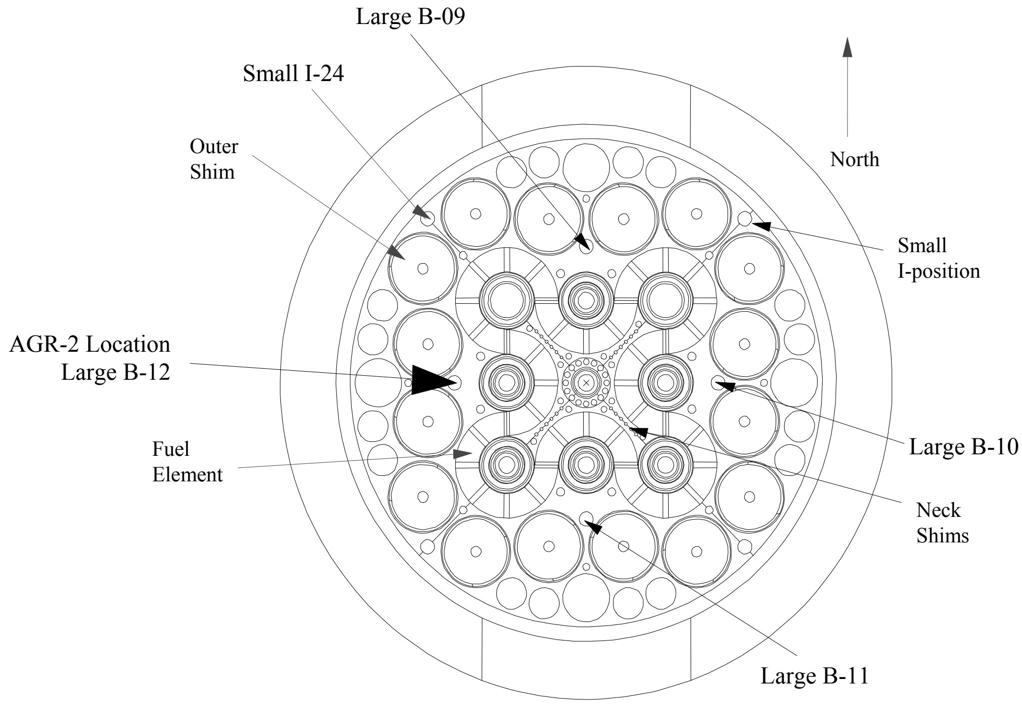

Figure 2-1. ATR core cross section displaying the B-12 position.

The test train planned for AGR-2 is based on the experience gained from previous irradiations in the ATR, using instrumented lead experiments. Instrumented lead experiments are used for irradiations requiring a controlled environment and monitored parameters. The experiment test train positions the fuel within the test location and contains sweep gas lines and thermocouple wiring that is routed through access ports to external support systems.

AGR fuel was originally based on the Gas-Turbine Modular Helium Reactor (GT-MHR) fissile particle design. This fuel form (UCO) was used as a starting point in defining a baseline fuel for AGR-1 (Barnes, 2006). Further development for fuel to be produced in a large coater has resulted in the AGR-2 fuel specification (Barnes, 2009). Similarly to AGR-1, AGR-2 uses UCO kernels with slightly increased diameters. AGR-2 fuel also contains UO_2 kernels typical of the fuel from German and South African pebble bed designs. Consequently fuels containing UCO and UO_2 will be irradiated in the AGR-2 experiment. Both types are included as one variant in separate capsules. In addition to U.S. produced fuel, UO_2 fuels produced in France and South Africa are included in AGR-2, but these foreign fuels are not discussed in this document.

3. EXPERIMENT DESCRIPTION

3.1 Fuel Particles

Fuel for AGR-2 consists of TRISO–coated particles that are slightly less than 1 mm in diameter. Each particle has a central kernel containing the fuel material, a porous carbon buffer layer, an inner pyrolytic carbon (IPyC) layer, a silicon carbide (SiC) barrier coating, and an outer pyrolytic carbon (OPyC) layer. This fuel design is illustrated in Figure 3-1. The functions of each coating layer are listed in Table 3-1.

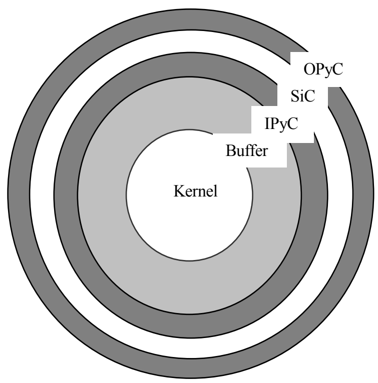

Figure 3-1. Schematic of a typical TRISO-coated fuel particle.

Table 3-1. Primary functions of particle fuel components.

Component	Primary function
Kernel	Contains fissile/fertile fuel
Buffer	Provides void space for fission product gases and accommodates differential changes in dimensions between coating layers and kernel
IPyC	Structural layer which also protects the kernel during SiC deposition
SiC	Primary structural layer and primary fission product barrier
OPyC	Structural layer which also permits bonding to carbonaceous matrix material

Kernels for AGR-2 consist of LEU UCO and LEU UO_2 fuels. The U.S. kernels were fabricated by BWX Technologies (BWXT) in accordance with the AGR-2 Fuel Product Specification (Barnes, 2009). For each type of fuel, several production batches were combined into a single composite: Lot G73I-14-69307 for UCO kernels and Lot G73AA-10-69308 for UO_2 kernels. Complete characterization data for these kernel lots are compiled in their respective Data Certification Package (BWXT, 07/2008 & 12/2008). Selected kernel composites properties and corresponding fuel product specifications are listed in Table 3-2.

Table 3-2. Selected properties for kernel Lot G73I-14-69307 (UCO) and Lot G73AA-10-69308 (UO_2).

UCO Kernel Property	Specified Range for Mean Value	Actual Mean Value ± Population Standard Deviation
Diameter (μm)	425 ± 10	426.7 ± 8.8
Density (Mg/m^3)	≥ 10.4	10.966 ± 0.033
U-235 enrichment (wt %)	14.0 ± 0.10	14.029 ± 0.026
Carbon/uranium (atomic ratio)	0.40 ± 0.10	0.392 ± 0.002
Oxygen/uranium (atomic ratio)	1.50 ± 0.20	1.428 ± 0.005
[Carbon + oxygen]/uranium (atomic ratio)	≤ 2.0	1.818 ± 0.005
Total uranium (wt %)	≥ 88.5	89.463 ± 0.051
Sulfur impurity (ppm – wt)	≤ 1500	365 ± 12
Phosphorus impurity(ppm – wt)	≤ 1500	≤ 50
All other impurities	various	Below minimum detection limits and within specification
UO_2 Kernel Property	Specified Range for Mean Value	Actual Mean Value ± Population Standard Deviation
Diameter (μm)	500 ± 10	507.7 ± 11.9
Density (Mg/m^3)	≥ 10.4	10.858 ± 0.082
U-235 enrichment (wt %)	9.6 ± 0.10	9.600 ± 0.010
Oxygen/uranium (atomic ratio)	≥ 1.98 & ≤ 2.1	2.003 ± 0.005
Phosphorus, Sulfur impurities (ppm – wt)	≤ 1500	≤ 50
All other impurities	various	Below minimum detection limits and within specification

The UCO and UO_2 kernels were coated and characterized by BWXT (BWXT, 09/2008 & 2009). In addition, ORNL characterized composite particles with anisotropy, sphericity, mass and diameter measurements (Hunn, 2008, 07/2010, 02/2010 & 03/2010). Coating was performed in accordance with the AGR-2 Fuel Product Specification (Barnes, 2009). Two particle composite lots comprise the fuel to be irradiated in AGR-2, one for each type of fuel: Lot G73J-14-93073A for UCO coated particles and Lot G73H-10-93085B for UO_2 coated particles.

Each AGR-2 capsule contains only one fuel type. U.S. UCO fuel will be irradiated in Capsules 2, 5 and 6, U.S. UO_2 fuel in Capsule 3, French UO_2 fuel in Capsule 1, and South African UO_2 fuel in Capsule 4. These assignments are listed in Table 3-3 where the capsules are numbered consecutively from the bottom (Capsule 1) to the top (Capsule 6).

A summary of selected properties, based on actual characterization data, for each of the two U.S. coated particle composites (UCO and UO_2) is listed in Table 3-4. Mean value specifications, where applicable, are also listed in Table 3-4 for comparison purposes.

Table 3-3. AGR-2 capsule contents.

Location	Coated Particle Composite	Fuel Designation
Capsule 6	G73J-14-93073A	UCO
Capsule 5	G73J-14-93073A	UCO
Capsule 4	-	South African UO_2
Capsule 3	G73H-10-93085B	UO_2
Capsule 2	G73J-14-93073A	UCO
Capsule 1	-	French UO_2

Table 3-4. Selected properties for AGR-2 coated particle composites.

Property	Specified Range for Mean Value	Actual Mean Value ± Population Standard Deviation	
		UCO	UO_2
Buffer thickness (μm)	100 ± 15	98.9 ± 8.4	97.7 ± 9.9
IPyC thickness (μm)	40 ± 4	40.4 ± 2.5	41.9 ± 3.2
SiC thickness (μm)	35 ± 3	35.2 ± 1.2	37.5 ± 1.2
OPyC thickness (μm)	40 ± 4	43.4 ± 2.9[(a)]	45.6 ± 2.4[(a)]
Buffer density (Mg/m^3)	1.05 ± 0.10	not measured[(b, d)]	0.99[(c)]
IPyC density (Mg/m^3)	1.90 ± 0.05	1.890 ± 0.011	not measured[(b, d)]
SiC density (Mg/m^3)	≥ 3.19	3.197 ± 0.004	3.199[(e)]
OPyC density (Mg/m^3)	1.90 ± 0.05	1.907 ± 0.007	1.884 ± 0.004
IPyC anisotropy (BAF)	≤ 1.045	1.0349 ± 0.0012	1.0334 ± 0.0027
OPyC anisotropy (BAF)	≤ 1.035	1.0263 ± 0.0011	1.0219 ± 0.0012
IPyC anisotropy post compact anneal (BAF)	Not specified	1.0465 ± 0.0049	1.0471 ± 0.0036
OPyC anisotropy post compact anneal (BAF)	Not specified	1.0429 ± 0.0019	1.0365 ± 0.0016
SiC sphericity (aspect ratio)	Mean not specified [(f)]	1.037 ± 0.011	1.034 ± 0.010
OPyC sphericity (aspect ratio)	Not specified	1.052	1.052
Particle diameter [(g)] (μm)	Mean not specified	873.2 ± 23	953.0 ± 28
Particle mass (mg)	Mean not specified	1.032 ± 0.003	1.462 ± 0.005

Notes: (a) 95% upper confidence thickness exceeds specifications. Justification of acceptance: OPyC thickness does not affect the compacting process or the fuel performance during irradiation (BWXT, 09/2008 & 2009).
(b) B&W's hot sampling system does not allow both buffer and IPyC density measurements (BWXT, 2009).
(c) Single determination, no statistical confidence available (BWXT, 2009).
(d) Similar samples showed measurement results within specifications (BWXT, 09/2008 &2009).
(e) Lower confidence level.
(f) Critical region is specified such that ≤ 1 % of the particles shall have an aspect ratio ≥ 1.14 for UCO fuel and ≥ 1.10 for UO_2 fuel.
(g) Based on mean average particle measurements, not sums of mean layer thicknesses.

3.2 Fuel Compacts

After coating, AGR-2 fuel was formed into right cylindrical compacts. The compact matrix material is composed of a thermosetting carbonaceous material. Prior to compacting, the fuel particles were overcoated with approximately 215 and 390 μm thick layers of the compact matrix material for UCO and UO_2 fuels respectively. This overcoat is intended to prevent particle-to-particle contact and help achieve the desired packing fraction of fuel particles.

AGR-2 compacts are nominally 25.1 mm in length and 12.3 mm in diameter. The compacts are fabricated with fuel free end caps of matrix material less than 0.5 mm thick. These end caps ensure smooth, protected surfaces that help to prevent fuel particle damage during handling.

The same compacting process was used for both UCO and UO_2 fuels. A summary of selected properties, based on actual characterization data (Hunn, 02/2010 & 03/2010) and derived from these data, for each fuel type is listed in Table 3-5. Mean value specifications, where applicable, are also listed in Table 3-5 for comparison purposes. Data for compact mass, diameter and length are based on averages of those compacts sent to INL. For traceability, Table 3-6 lists the compacts sent to INL.

Table 3-5. Selected properties for AGR-2 compacts.

Property	Specified Range for Mean Value	Actual Mean Value ± Population Standard Deviation	
		UCO	UO_2
Compact mass (g)	Not specified	6.294 ± 0.011	6.103 ± 0.015
Mean uranium loading (g U/compact)	1.265 ± 0.07 (UCO) 1.00 ± 0.05 (UO_2)	1.257 ± 0.03	0.993 ± 0.006
Diameter [(b)] (mm)	12.22 – 12.46	12.286 ± 0.005	12.269 ± 0.007
Length [(b)] (mm)	25.02 – 25.40	25.141 ± 0.017	25.135 ± 0.018
Number of particles per compact [(a)]	Not specified	3176	1543
Particle volume packing fraction (%)	Not specified	37	23
Effective overall compact density[(a)] (Mg/m^3)	Not specified	2.11	2.05
Compact matrix density (Mg/m^3)	≥ 1.45	1.589 ± 0.005	1.680 ± 0.008
Compact weight % U [(a)]	Not specified	19.97	16.27
Compact weight % O [(a)]	Not specified	1.92	2.19
Compact weight % Si [(a)]	Not specified	6.85	4.54
Compact weight % C [(a)]	Not specified	71.26	77.00
Iron content (µg Fe outside of SiC/compact)	≤ 25	4.04	2.75
Chromium content (µg Cr outside of SiC/compact)	≤ 50	0.61	0.48
Manganese content (µg Mn outside of SiC/compact)	≤ 50	0.136	0.133
Cobalt content (µg Co outside of SiC/compact)	≤ 50	1.115	0.113
Nickel content (µg Ni outside of SiC/compact)	≤ 50	0.96	0.59
Calcium content (µg Ca outside of SiC/compact)	≤ 50	39.34	35.16
Aluminum content (µg Al outside of SiC/compact)	≤ 50	29.60	42.69
Titanium content (µg Ti outside of SiC/compact)	Note (c)	2.81	3.31
Vanadium content (µg V outside of SiC/compact)	Note (c)	17.09	15.41
U contamination fraction [(d)] (g exposed U / g U in compact)	$\leq 2.0 \times 10^{-5}$	$\leq 2.5 \times 10^{-5}$ [(e)]	$\leq 3.2 \times 10^{-5}$ [(e)]
U contamination fraction w/o exposed kernels (g leached U / g U in compact)	Not specified	1.59×10^{-6}	1.57×10^{-6}
Defective SiC coating fraction [(d)]	$\leq 1.0 \times 10^{-4}$	$\leq 1.2 \times 10^{-5}$	$\leq 2.5 \times 10^{-5}$
Defective IPyC coating fraction [(d)]	$\leq 1.0 \times 10^{-4}$	$\leq 4.8 \times 10^{-5}$	$\leq 7.7 \times 10^{-5}$
Defective OPyC coating fraction [(d)]	$\leq 1.0 \times 10^{-2}$	$\leq 9.5 \times 10^{-4}$	$\leq 2.0 \times 10^{-3}$

Notes: (a) Approximate calculated value derived from other characterized properties.
(b) Allowable range corresponding to upper and lower critical limits specified with no compacts exceeding the limits, which require 100 % inspection of all compacts.

(c) Mean value specification of ≤ 240 µg Ti+V outside of SiC per compact.
(d) 95% confidence defect fraction.
(e) Values exceed specifications: the non-conformances are documented in NCR-44791 with a disposition of use as is.

Table 3-6. AGR-2 compacts sent to INL.

UCO Lot LEU09-OP2-Z		UCO Lot LEU09-OP2-Z		UO_2 Lot LEU11-OP2-Z	
Compact ID	Assigned Position[(a)]	Compact ID	Assigned Position[(a)]	Compact ID	Assigned Position[(a)]
Z016	6-4-1	Z105	2-1-3	Z001	3-2-3
Z017	6-3-1	Z112	5-1-3	Z029	3-1-1
Z018	5-4-3	Z119	2-4-3	Z032	3-2-1
Z022	6-2-1	Z125	2-3-1	Z034	3-3-2
Z026	6-1-1	Z126	2-2-1	Z036	3-2-2
Z028	5-4-1	Z128	5-2-2	Z098	3-1-2
Z040	5-3-3	Z131	6-1-3	Z106	3-3-1
Z043	2-4-2	Z132	5-3-1	Z127	3-4-3
Z049	6-4-2	Z135	5-2-1	Z150	3-4-2
Z053	6-3-2	Z141	5-1-1	Z180	3-1-3
Z056	6-2-2	Z154	5-1-2	Z188	3-4-1
Z059	5-4-2	Z156	2-1-1	Z193	3-3-3
Z062	5-2-3	Z014	spare	Z065	spare
Z063	6-1-2	Z015	spare	Z072	spare
Z066	2-3-2	Z100	spare	Z075	spare
Z075	2-2-2	Z103	spare	Z078	spare
Z077	5-3-2	Z114	spare	Z079	spare
Z079	2-1-2	Z124	spare	Z089	spare
Z082	6-4-3	Z127	spare	Z101	spare
Z083	2-4-1	Z129	spare	Z133	spare
Z085	6-3-3	Z134	spare	Z140	spare
Z088	2-3-3	Z142	spare	Z181	spare
Z092	2-2-3	Z153	spare	Z183	spare
Z104	6-2-3	Z167	spare	Z197	spare

Notes: (a) Sequence is capsule number – level number – stack number where capsules are numbered sequentially from bottom (Capsule 1) to top (Capsule 6), levels are also numbered sequentially within a capsule from bottom (Level 1) to top (Level 4), and stacks are sequentially numbered clockwise from the northwest (Stack 1) to the southwest (Stack 3).

3.3 Test Train

As required by the Test Specification (Maki, 2010), the AGR-2 test train is a multi-capsule, instrumented lead experiment, designed for irradiation in the 38.1 mm (1.5 inch) diameter B-12 position of the ATR. The test train contains six capsules, each independently controlled for temperature and independently monitored for fission product gas release. An axial view of the test train is illustrated in Figure 3-2. Each AGR-2 capsule is 152.4 mm (6 inches) long. Capsules hosting U.S. and South African fuels contain twelve fuel compacts arranged in three vertical stacks with each stack containing four compacts. The capsule hosting the French fuel contains six fuel compacts arranged in three vertical stacks with each stack containing two compacts. Figure 3-3 illustrates a radial view of a capsule. Significant features of the test train are described below and further details are presented in the Technical and Functional Requirements documents (TFR-248, 2005 & TFR-559, 2010).

Thermocouples

The type and size of thermocouples (TCs) used for AGR-2 were based on experience with the TCs used for AGR-1. Niobium sheathed TCs were selected to avoid possible migration of nickel, iron, or chromium from Inconel TC sheathes through the graphitic sample holders and compact matrix and potentially attack the SiC layer of the fuel. Commercial Type N TCs are used in AGR-2 because the INL-developed molybdenum-niobium (Mo-Nb) TCs used in AGR-1 did not show a better survivability.

It was also decided to use two TCs in Capsules 5 through 1 in order to use larger diameter TCs than were used in AGR-1. An outside diameter of 0.80 inches was determined by selecting the largest diameter possible that would still allow all of the TCs and gas lines to fit inside the thru tubes. The larger diameter TCs allow the use of larger thermoconductors. It is hoped that larger thermoconductors will help to mitigate premature failure of the TCs because of the repeated thermocycling that occurs as the reactor starts up and shuts down.

Six test TCs were manufactured using the same specifications as for the TCs used in AGR-2, except for a shorter length. These test TCs underwent 20 thermocycles (22 °C – 1105 °C) at the vendor's laboratory. Five of the test TCs were still operational after 20 cycles. The sixth TC failed after 13 cycles.

Each capsule contains two TCs except for Capsule 6, which contains five TCs. The TCs are placed in holes drilled in the graphite sample holders. All of the capsules have two TCs located around the periphery, on the side of the graphite holder facing away from core center. Avoidance of the hottest regions of the sample holders is expected to extend TC life. Capsule 6 has an additional three TCs. Two of these TCs are placed around the periphery on the side of the graphite holder facing toward the core center. The remaining TC is located in the center of the graphite holder.

A summary of TC type, sheath and insulation materials, and placement within the test train is provided in Table 3-7.

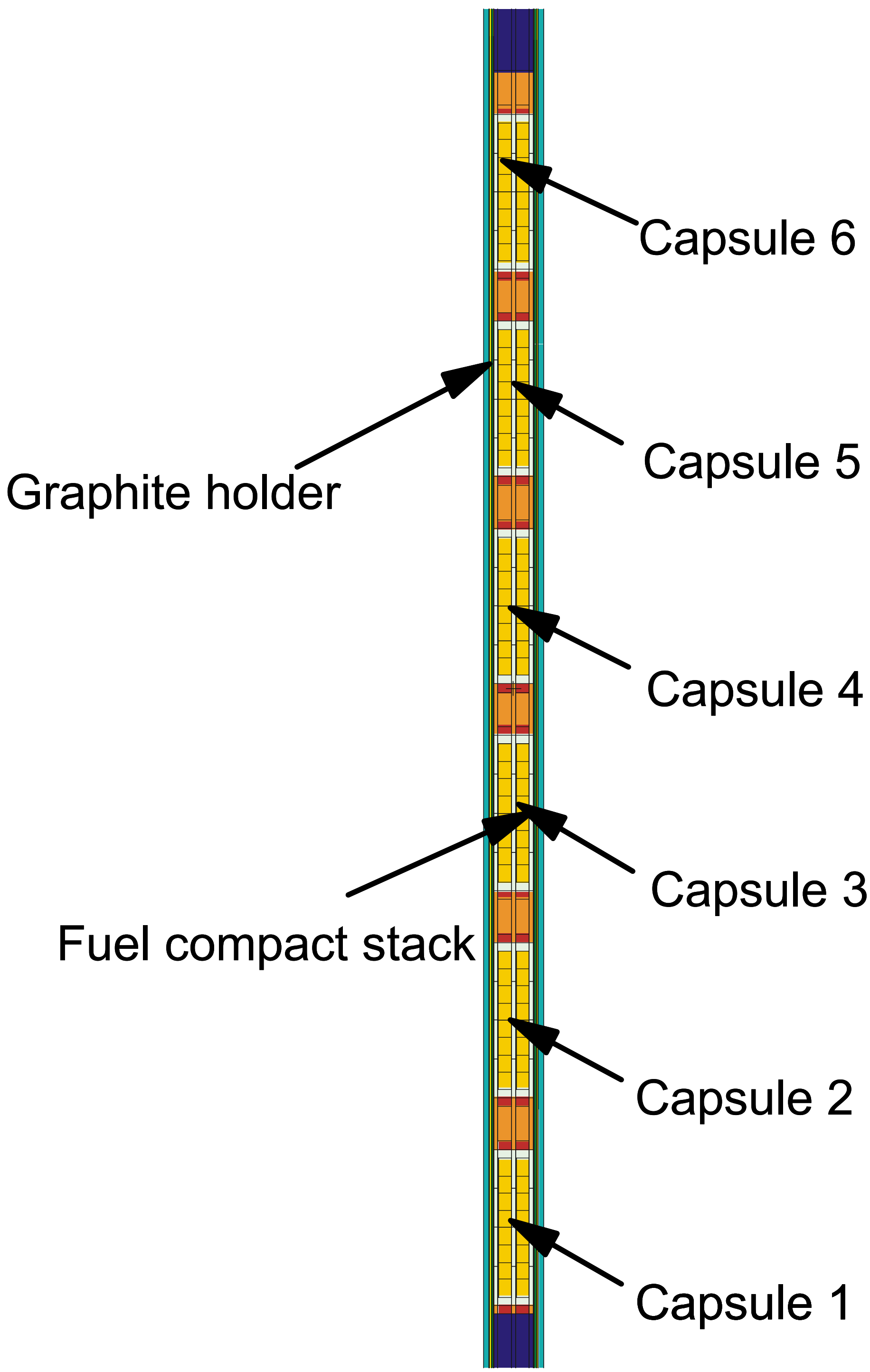

Figure 3-2. Axial schematic of the AGR-2 capsules.

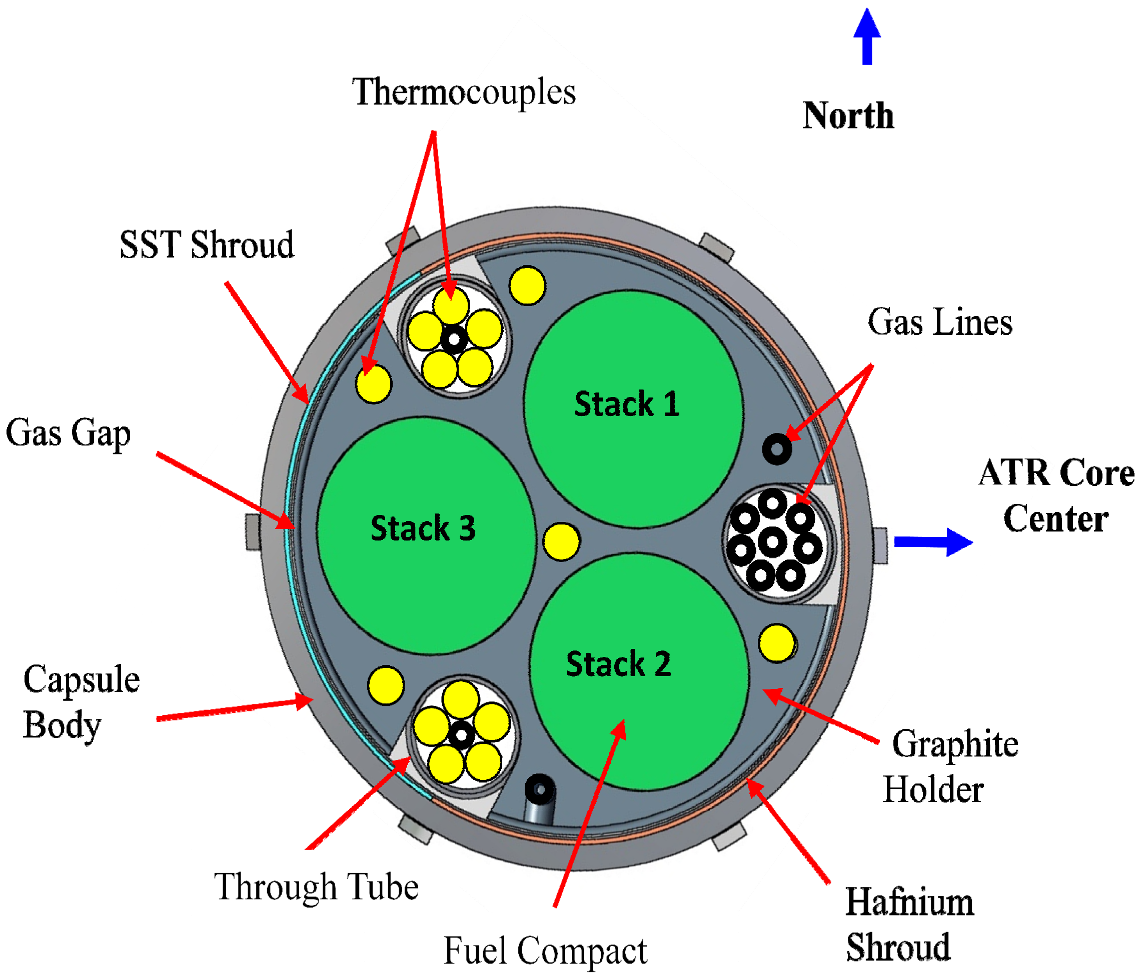

Figure 3-3. Radial schematic of an AGR-2 capsule.

Table 3-7. AGR-2 thermocouple assignments.

Capsule	Location[a]	Thermocouple Type [b]	Sheath / Insulation
6	NW	Type N	Nb/MgO
6	SW	Type N	Nb/MgO
6	C	Type N	Nb/MgO
6	N	Type N	Nb/MgO
6	SE	Type N	Nb/MgO
5	NW	Type N	Nb/MgO
5	SW	Type N	Nb/MgO
4	NW	Type N	Nb/MgO
4	SW	Type N	Nb/MgO
3	NW	Type N	Nb/MgO
3	SW	Type N	Nb/MgO
2	NW	Type N	Nb/MgO
2	SW	Type N	Nb/MgO
1	NW	Type N	Nb/MgO
1	SW	Type N	Nb/MgO

Notes: (a) Northwestern (NW) TCs are located in the outer portion of the graphite sample holder, north of the sample holder center, and away from the reactor center. Southwestern (SW) TCs are located in the outer portion of the graphite sample holder, south of the sample holder center, and away from the reactor center. North (N) TC is located in the outer portion of the graphite sample holder, directly north of the sample holder center (found only in Capsule 6). Southeastern (SE) TC is located in the outer portion of the graphite sample holder, south of the sample holder center, and toward the reactor center (found only in Capsule 6). Center (C) TC is located in the center of the graphite sample holder (found only in Capsule 6).
(b) All TCs are 2.03 mm (0.080 inch) in diameter.

Melt Wires

Capsules 1, 3, 4, 5, and 6 each contain two beryllium melt wires approximately 0.25 mm in diameter. Capsule 2 contains a nickel melt wire approximately 0.5 mm in diameter. For each capsule, both melt wires (and the single melt wire for Capsule 2) are encapsulated in one single pure vanadium tube. The encapsulation is about 7.9 mm long with an outer diameter of approximately 1.4 mm and is engraved with a unique identification number. The vanadium tubes are placed within holes drilled near the center of each capsule and about midway up from the bottom of each graphite sample holder. Post irradiation examination of the beryllium melt wires will indicate if Capsules 1, 3, 4, 5, or 6 experienced temperatures in excess of 1278 °C. Examination of the nickel melt wire will indicate if Capsule 2 experienced temperatures in excess of 1400 °C.

Neutron Monitors

In order to measure both thermal and fast neutron fluences, flux wires are placed in each capsule. After irradiation, the induced activity of the wires will be converted to fluences with the appropriate neutron energy range and will also be used as a benchmark for physics analyses. Three materials are used for the wires, pure iron (Fe), vanadium (V) + 0.1% Cobalt (Co), and pure niobium (Nb). Each wire is encapsulated in a pure vanadium tube with an outer diameter of about 1.4 mm. The lengths of the encapsulations are about 7.1 mm for the iron wire, 4.8 mm for the V + 0.1% Co wire, and 8.6 mm for the Nb wire. A unique identification number is engraved on each encapsulation. These encapsulated neutron monitors are placed in holes drilled into the bottom and around the periphery of the graphite sample holders. Characteristics of the flux wires are listed in Table 3-8.

Table 3-8. Characteristics of AGR-2 flux wires.

Material	Reaction	Reaction Product Half-Life	Neutron Activation Energy Range
V + 0.1% Co	Co-59 (n,γ) Co-60	5.3 years	thermal
Fe	Fe-54 (n,p) Mn-54	312 days	1 MeV threshold
Nb	Nb-93 (n,n') Nb-93m	16 years	0.18 MeV threshold

Sweep Gas and Temperature Control

Independent gas lines will route a mixture of helium and neon gases through each of the six capsules to provide temperature control and to sweep released fission product gases to the fission product monitoring system (FPMS). Temperature control is based on temperature feedback from the TCs in each capsule and by varying the sweep gas composition (between 100 % helium for high conductivity and 100% neon for low conductivity). This blending of sweep gases will be accomplished by a computerized mass flow controller before the gas enters the test train. Gas flow will be ≤ 50 sccm (standard cubic centimeters per minute) at a pressure of about 1–3 psig (pound per square inch – gauge) or 7–21 kPa-gauge.

The specific location of Capsule 6 at the top of the test train makes it less exposed to neutron flux. With the standard capsule design, the sweep gas could not maintain the temperature in Capsule 6 within the range specified by the test specifications. To prevent the temperature drop, a 0.002 inch thick stainless steel thermal radiation shield is installed in the gas gap between the graphite holder and the stainless steel and hafnium shrouds. This thermal radiation shield limits heat transfer and helps keep the temperature inside Capsule 6 within specifications.

Sweep gas flow, originating from gas supply bottles, is routed to the mass flow controller cabinet where the helium and neon gases are blended for each capsule. When a new bottle is connected to the system, a solenoid valve is actuated and a sample of the gas from the new bottle is temporarily routed to the gas verification panel where thermal conductivity and moisture measurements are performed for both the helium and neon gas lines. After verification, the solenoid is again actuated and the gas flow bypasses the gas verification cabinet and is routed directly from the gas regulator panel to the mass flow control cabinet. Gas routed to the mass flow control cabinet is then routed on to the capsule inlet isolation panel, which can be used to isolate inlet gas flow to each capsule independently during reactor outages or in the event of a failure. Upon exiting the capsule and test train, the gas flows through the outlet isolation panel to another panel containing a particulate filter, a moisture detector, and a 3-way valve. The valve routes the gas either to the designated fission product monitor or the standby-backup fission product monitor. Another 3-way valve allows the gas to be routed to a manual grab sample line. After passing through the fission product monitor system, the gas lines combine into a common exhaust header that routes the gas through a silver-zeolite filter. The exhaust gas is finally routed to the ATR stack. A schematic of this gas flow is presented in Figure 3-4.

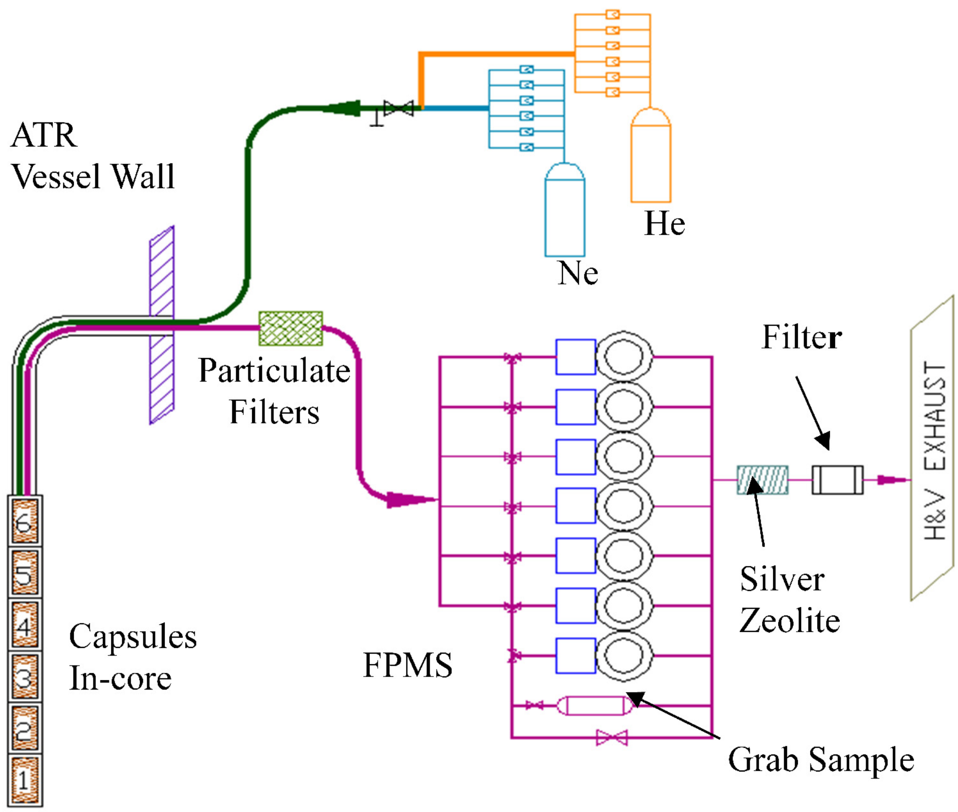

Figure 3-4. Simplified flow path for AGR-2 sweep gas.

Thru Tubes

Thru tubes route TCs and gas lines through each of the six capsules. Niobium thru tubes were selected for AGR-2 to avoid possible migration of Ni, Fe, or Cr through the graphitic sample holders and compact matrix and potentially attack the SiC layer of the fuel.

In order to maintain gas tightness in each capsule, the original test train design concept had the thru tubes brazed at the top and bottom of each capsule. However, because of high axial and azimuthal thermal gradients in the thru tubes, high stresses would result with tube bowing and contact with the graphite sample holders. To alleviate the stresses and tube bowing, the AGR-1 and AGR-2 designs have the thru tubes brazed only at the top of each capsule but a close slip fit at the bottom. Neolube was applied around the tube at the bottom of the capsules to aid in assembly and further reduce the clearance between the tubes and mating holes. To further prevent capsule-to-capsule cross gas leakage, a nominal helium or neon flow of 18 sccm at about 1 psig (6.9 kPa-gauge) above the capsule pressure is provided into the leadout cavity (and into the test train common plenum or void volume) via a mass flow controller. This small gas flow provides an inward flow through the space between the capsule bottom heads and around the thru tubes. Experimental validations were conducted prior to start of irradiation to confirm that ingress gas flow and tube clearances are sufficient to prevent gas leakage from capsule to capsule.

Power Shaping

Two techniques are used to adjust the neutron flux incident upon the AGR-2 test articles to shape the temporal and spatial fuel power distribution. These techniques include the placement of a neutron absorbing shroud (hafnium) around the fuel, and placing a burnable absorber (B_4C) near the fuel.

Since compact fuel Stacks 1 and 2 directly face ATR's core center and shield Stack 3, fuel in Stacks 1 and 2 have higher neutron exposure and hence, higher rates of burnup and fluence. To achieve more uniform neutron exposure, a hafnium shroud is placed next to the inner wall of the capsules. The 0.25 mm (0.010 inch) thick hafnium shroud extends 240° around the circumference, centered between Stacks 1 and 2. The hafnium with a high neutron absorption cross section thus reduces the neutron exposure to Stacks 1 and 2 (in greater proportion than to Stack 3) resulting in lower neutron exposure gradients across the capsules. To maintain a uniform control gas gap, the remaining 120° circumference is filled with 0.25 mm thick stainless steel. This configuration is illustrated in Figure 3-5.

Another power shaping technique used in AGR-2 is the addition of B_4C to the graphite sample holders. Without this burnable absorber, fuel power, or heat generation rate, would be highest at beginning of life (BOL) and drop exponentially as the fissile fuel content is consumed. For AGR-1 test fuel, calculations showed that the range of heat generation rates spans about 125 W/cm^3 from beginning of life to essentially, full burnup. Unfortunately, temperature control of the test fuel can only be maintained within a limited range of heat generation rates (about half of the maximum heat rate i.e. 50-60 W/cm^3 at most) for a given control gas gap width and with a varying mixture of helium and neon sweep gas. To reduce the range of test fuel heat generation rates, B_4C is added to the graphite sample holders. At beginning of life, the boron absorbs neutrons, reducing neutron exposure to the fuel and hence, reducing fuel power. As the irradiation continues, boron is depleted, neutron exposure to the fuel increases which increases fuel power. Eventually, with the boron consumed and the fissile fuel content diminished, fuel power decreases with continued irradiation. This temporal effect is illustrated in Figure 3-6.

The same behavior is expected on the AGR-2 test train and B_4C was also added to the graphite sample holders. Due to various types of fuels and enrichments, and to axial reductions in neutron flux at the top and bottom of the test train, the capsules require different amounts of B_4C. Design calculations have determined that AGR-2 graphite sample holders would contain 4.92 wt% B_4C in Capsules 3 and 4, and 5.7 wt% B_4C in Capsules 2 and 5. The bottom capsule, Capsule 1, contains 4.25 wt% B_4C, and the top capsule , Capsule 6, contains 4.83 wt% B_4C.

The control gas gap of each AGR-2 capsule is sized so that thermal control may be maintained during the peak in test fuel heat generation (depending on capsule, occurring at about 200 to 350 effective full power days [EFPDs]) under normal reactor power. This allows for thermal control from BOL through approximately 500 EFPD. However, during a high power reactor cycle, or PALM cycle, reactor power is increased by as much as 43%, resulting in a corresponding increase in test fuel power. Should this occur during the time the fuel would be normally operating above about 90 W/cm^3 (between about 50 and 375 EFPD), the resulting increase in fuel power would exceed the power range that permits thermal control. To avoid undesirable fuel power increase during a PALM cycle, the test train will be moved from its B-12 position to either the ATR canal or the I-24 position of the ATR (see Figure 2-1).

Relocating the experiment from the west large B position (B-12) to the northwest small I position (I-24) during a PALM cycle would require minimal handling and allow it to operate normally and be monitored as thermocouples and gas lines would also be relocated during the outages before and after the PALM cycle. Temperature control and fission product release monitoring through FPMS would therefore be carried out similarly in both ATR positions. The current ATR planning includes two PALM cycles during the span of the AGR-2 irradiation, after about 240 and 450 EFPD of irradiation. The expected peak heat rate of 124 W/cm^3 is expected around 200 EFPD, which would be decreased to about 59 W/cm^3 with the test train relocated in the I-24 position.

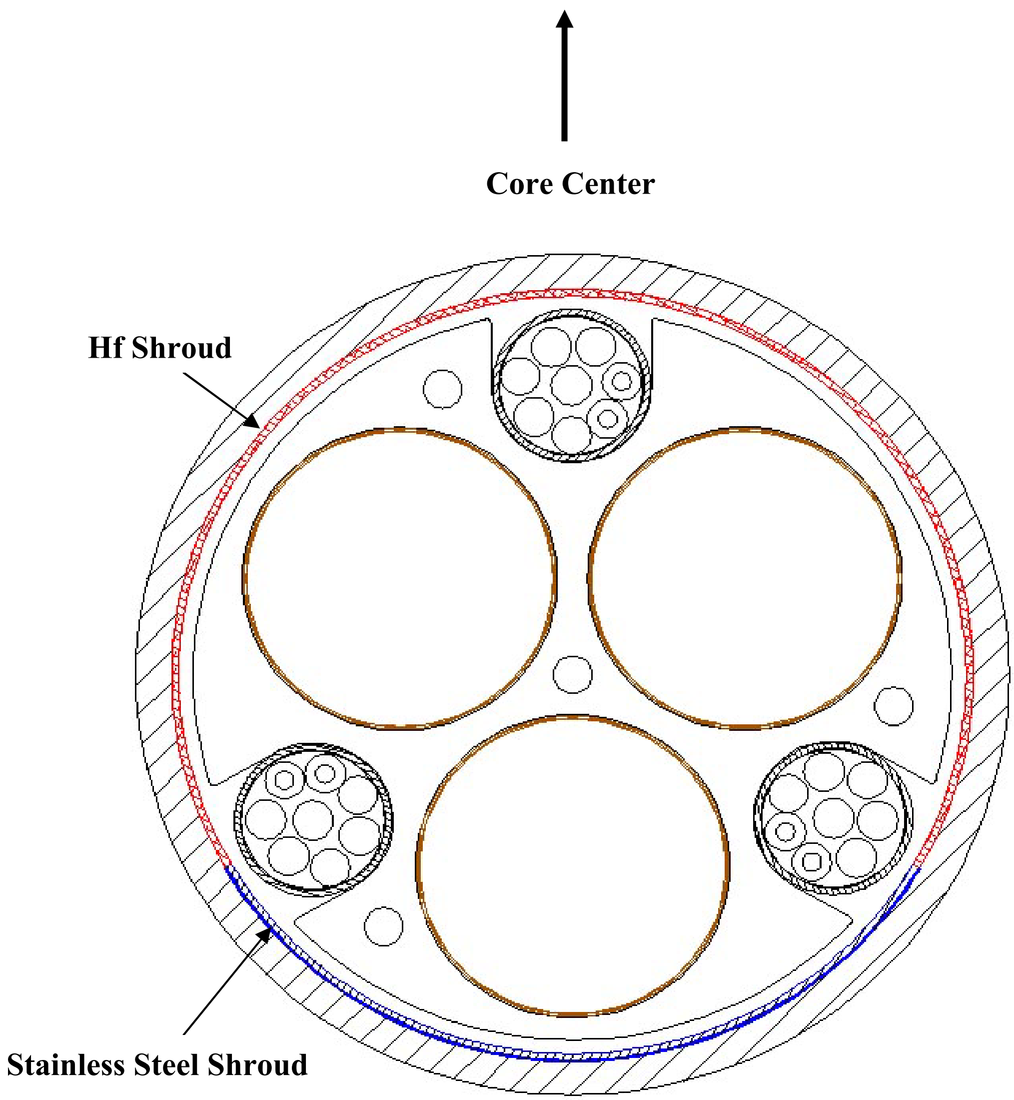

Figure 3-5. Placement of hafnium shroud in AGR-2.

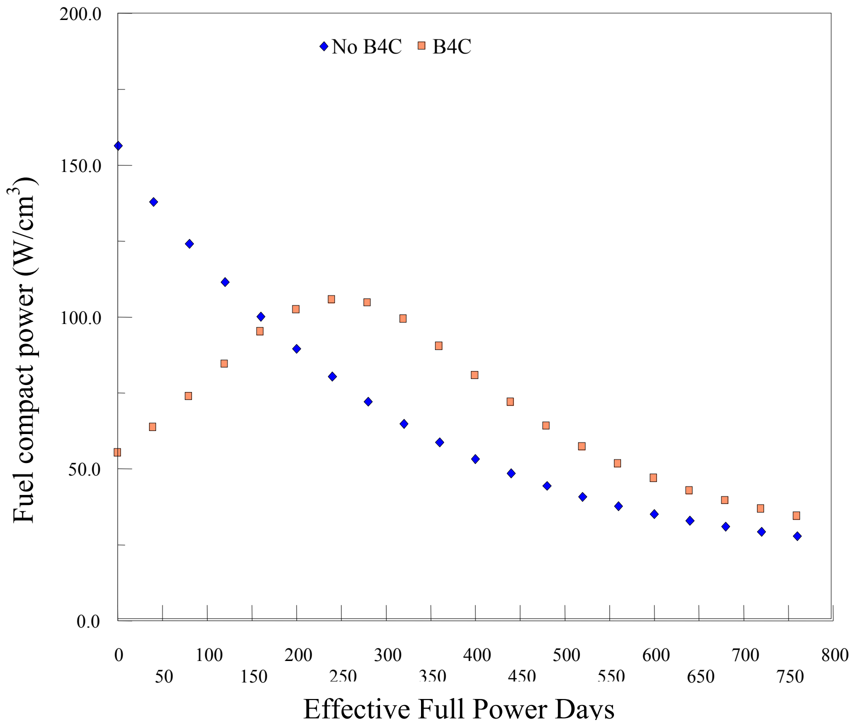

Figure 3-6. Illustration of the effect of B_4C on fuel power.

3.4 Fission Product Monitoring System

Each AGR-2 capsule will be continuously monitored for fission product gas release by the fission product monitoring system (FPMS). The FPMS consists of seven sets of gross radiation monitor and spectrometer detector pairs. One detector set is designated for each of the six capsules, while the seventh detector set serves as a backup spare. A detector set is illustrated in Figure 3-7.

Sweep gas carries released fission product gases from the capsules to the detector system under normal conditions with a transit time of about 150 seconds. The sweep gas passes in front of the gross radiation monitor, which uses a NaI(Tl) detector to detect each fuel particle failure up to the first 250 failures. Flow continues on to the spectrometer system, which uses an hyper pure germanium (HPGe) detector. The spectrometer system measures radionuclide concentrations that are used to determine release-to-birth ratios (R/Bs). Under normal operation, computerized data acquisition, analysis and storage occur continuously without operator intervention.

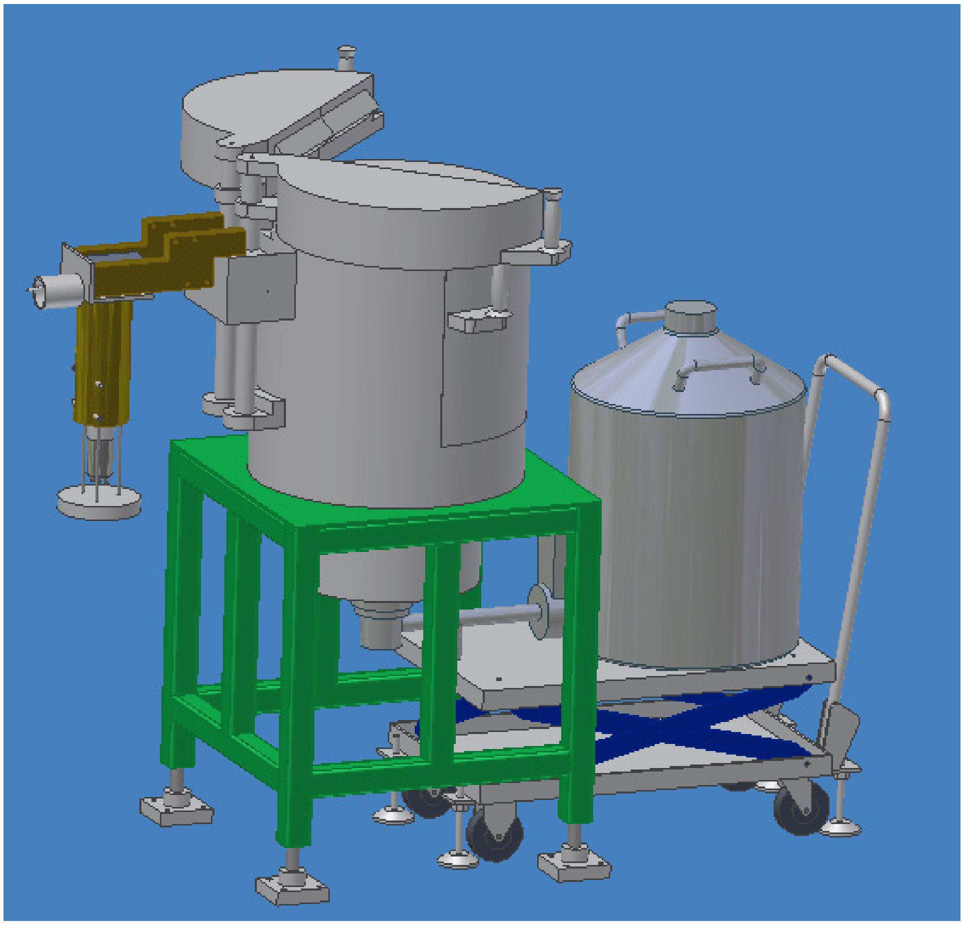

Figure 3-7. Gross radiation monitor and spectrometer detector for one AGR-2 sweep gas line.

4. TEST CONDITION REQUIREMENTS

This section presents the irradiation conditions expected for the AGR-2 experiment. These calculated conditions were derived from the latest physics and thermal analyses.

4.1 Particle Power

Fuel power is restricted by specification (Maki, 2010) and by an operational need to control test temperature (which is defined as the ability to adjust and maintain fuel temperatures within a prescribed range). The instantaneous peak power per particle specification of ≤ 400 mW/particle is intended to limit peak kernel temperatures and temperature gradients across the particle, which reduces fission product diffusion and potential fission product/SiC interactions, and may affect possible amobea effect in UO_2 fuels. Temperature control is achieved by varying the composition of the sweep gas (between 100% helium for high conductivity and 100% neon for low conductivity) within the control gas gap surrounding the fuel. For a given gas gap width, this control can be maintained within a range, or window, of fuel heat generation rates.

In order to extend the time that thermal control can be maintained, boron carbide (B_4C) is added to the graphite sample holders. This power shaping measure lowers the fuel heat generation rate early in life by absorbing neutrons and hence lowering the thermal neutron flux incident on the fuel. As the boron burns out later in life, the incident thermal neutron flux increases and fuel heat generation rates are dependent upon the remaining fissile fuel content. Since the AGR-2 test train extends axially beyond, both above and below, the flat portion of the ATR flux profile, the B_4C content is reduced in the top and bottom capsule sample holders to compensate for the lower neutron flux.

Based on projected nominal ATR power cycles, the maximum and minimum compact average heat generation rates (Chang and Parry, 2010) for AGR-2 are presented in Figure 4-1. A peak compact average power of 161 mW/particle is reached after about 200 EFPD. Considering that a conservative upper bound for compact peak-to-average power ratio is 1.1, the peak particle power is well within the specification limit of ≤ 400 mW/particle.

The effect of the boron carbide power shaping is clearly evident from Figure 4-1. Without B_4C, peak power would occur at beginning of life (BOL) when the fuel fissile content is highest and the power would thereafter continuously decrease. With B_4C, the BOL power is reduced by about a factor of three (see Figure 3-6), and as the boron burns out, power increases to its peak after about 200 EFPD and then decreases. This power shaping thus extends duration of thermal controllability.

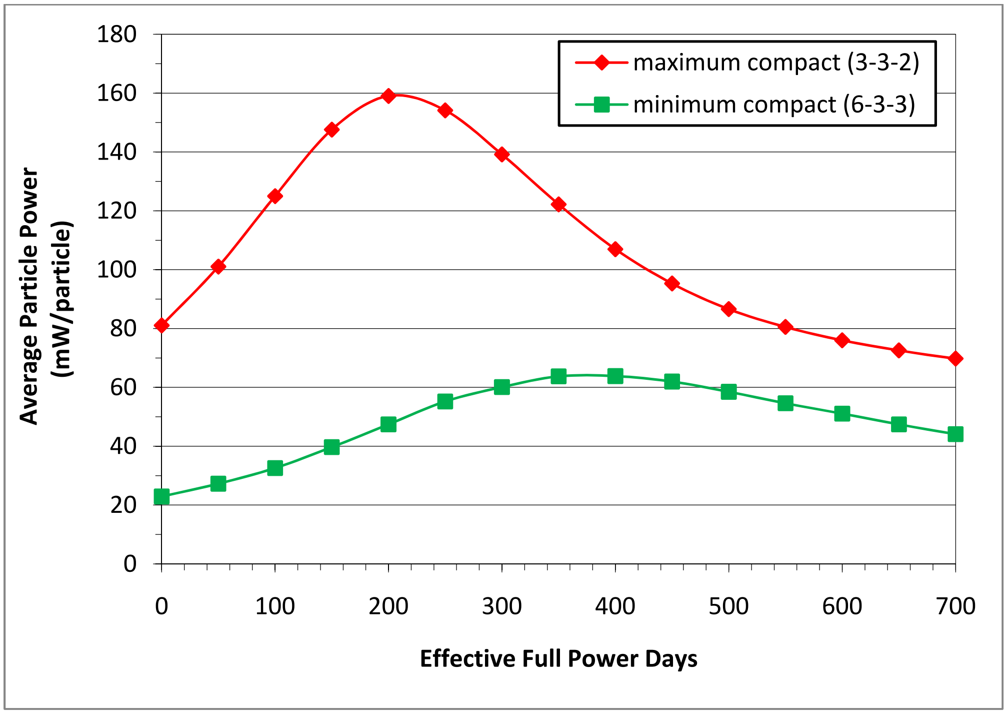

Figure 4-1. Average particle power for the maximum compact (Capsule 3 – Level 3 – Stack 2) and minimum compact (Capsule 6 – Level 3 – Stack 3).

4.2 Temperature

Three-dimensional, finite element, thermal calculations were performed at various irradiation times from BOL to 700 EFPD for the AGR-2 experiment. These calculations were performed as described in Reference (Hawkes, 2011) with the heat generation rates (Chang and Parry, 2010) described above with optimized control gas gap widths and varying sweep gas compositions.

The control gas gap width in each capsule was optimized, as stated above, such that:

- The instantaneous peak temperature for each capsule shall be ≤ 1800 °C.
- The time-average, peak fuel temperature shall be ≤ 1400 °C for one capsule containing UCO fuel, ≤ 1250 °C for each remaining capsule containing UCO fuel, and ≤ 1150 °C for the capsule containing UO_2 fuel.
- The time-average, volume-average (TAVA) temperature goal should be ≥ 1150 °C for the highest temperature capsule containing UCO fuel, ≥ 1000 °C for each remaining capsule containing UCO fuel, and ≥ 900 °C for each capsule containing UO_2 fuel.

For AGR-2, a time-average, peak temperature of ≤ 1250 °C duplicates the expected maximum prismatic NGNP temperature, while a time-average, peak temperature of ≤ 1150 °C duplicates the expected maximum pebble bed NGNP temperature. A time-average, peak temperature of ≤ 1400 °C allows for the study of UCO fuel at high irradiation temperatures. An instantaneous peak temperature specification of ≤ 1800 °C provides an operational limit to minimize over heating of the test fuel. The time-average, volume-average temperature goals ensure a sufficiently high temperature without exceeding the time-average peak temperature and the instantaneous peak temperature limits. Also, the time-average, volume-average temperature goals limit the extent of possible palladium attack of the fuel's silicon carbide (SiC) layer that could be significant, for durations of two years or more, at temperatures near or above 1200 °C (Petti and Maki, 2004).

Illustrative examples of these calculations are presented in Figures 4-2, 4-3, and 4-4 which respectively display the instantaneous peak temperatures, the time-average, peak temperatures and the time-average, volume-average (TAVA) temperatures as a function of effective full power days (EFPDs) for capsules loaded with U.S. fuel. As shown, temperatures are well controlled under their maximum limits by adjusting the gas mixture. The instantaneous peak temperature (Figure 4-2) is well under its limit of 1800 °C and it is requested to remain ≤ 1400 °C under normal operational conditions. In addition, the time-average, peak temperature (Figure 4-3) in Capsule 6 is kept steady by the use of a thermal radiation shield. For the other capsules, the drop in time-average, peak temperature expected after around 500-550 EFPD can be mitigated with an adequate raise of the power level by rotation of shim cylinders, or removal of neck shims, or a combination of both. A request will be made to ATR to that effect, as was done in AGR-1.

In addition to these plots, detailed temperature profiles are provided for each capsule in the Appendix.

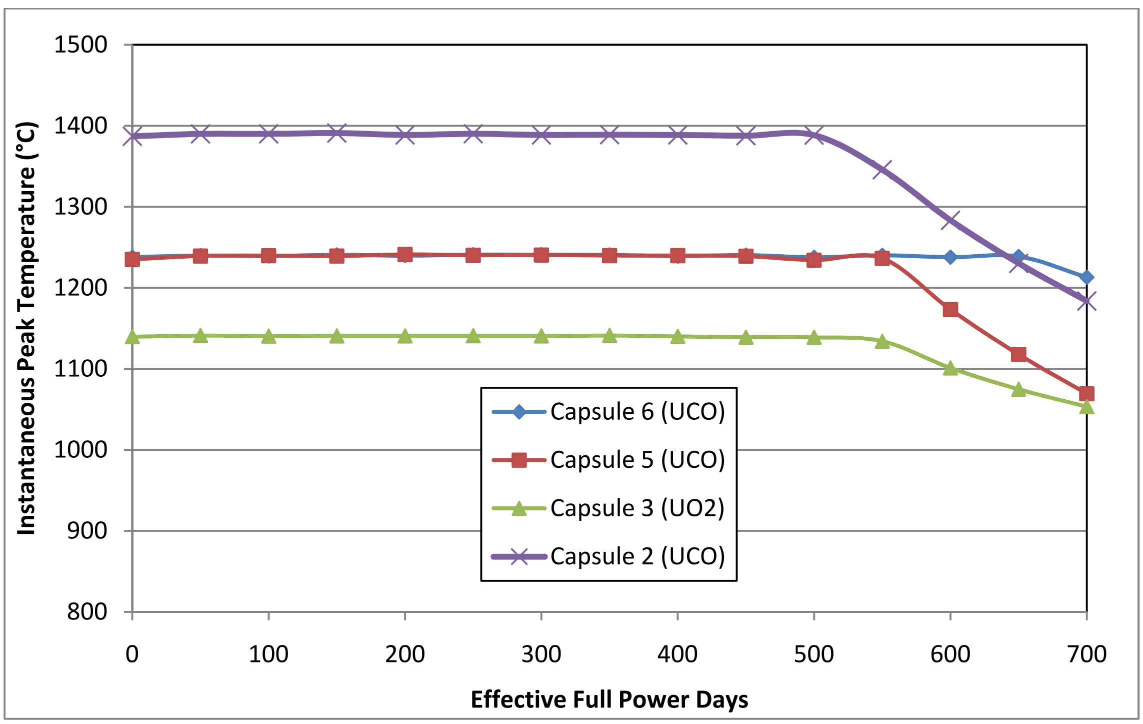

Figure 4-2. Instantaneous, peak temperature as a function of effective full power days (EFPD).

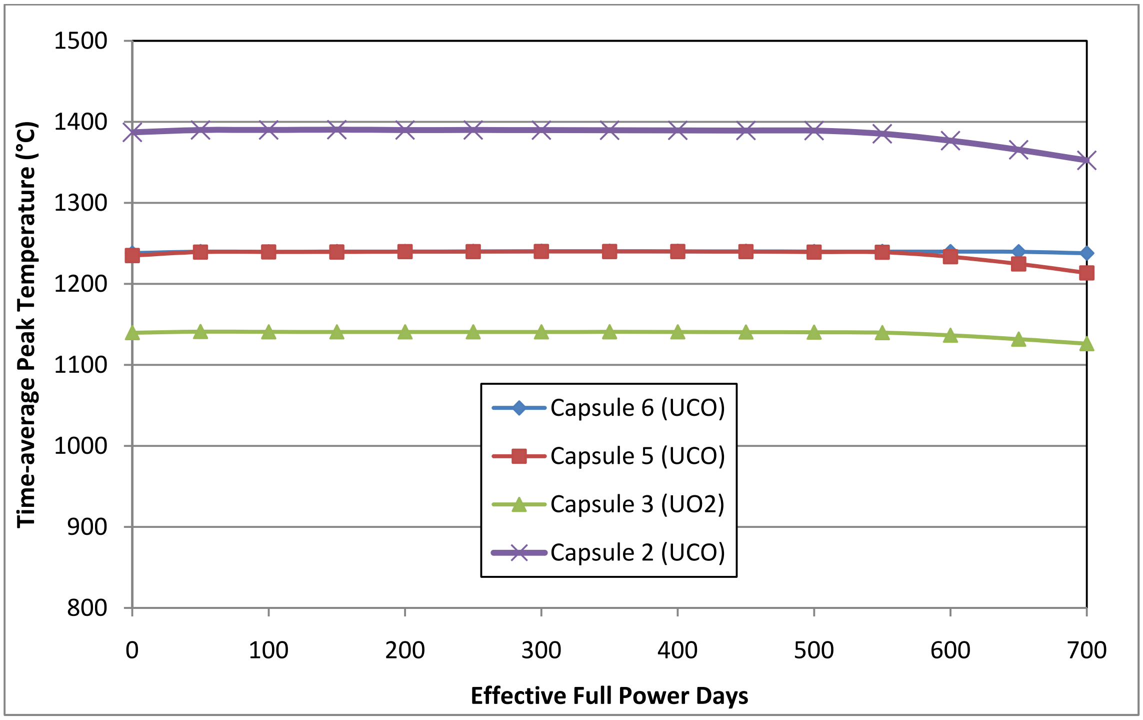

Figure 4-3. Time-average, peak temperature as a function of effective full power days (EFPD).

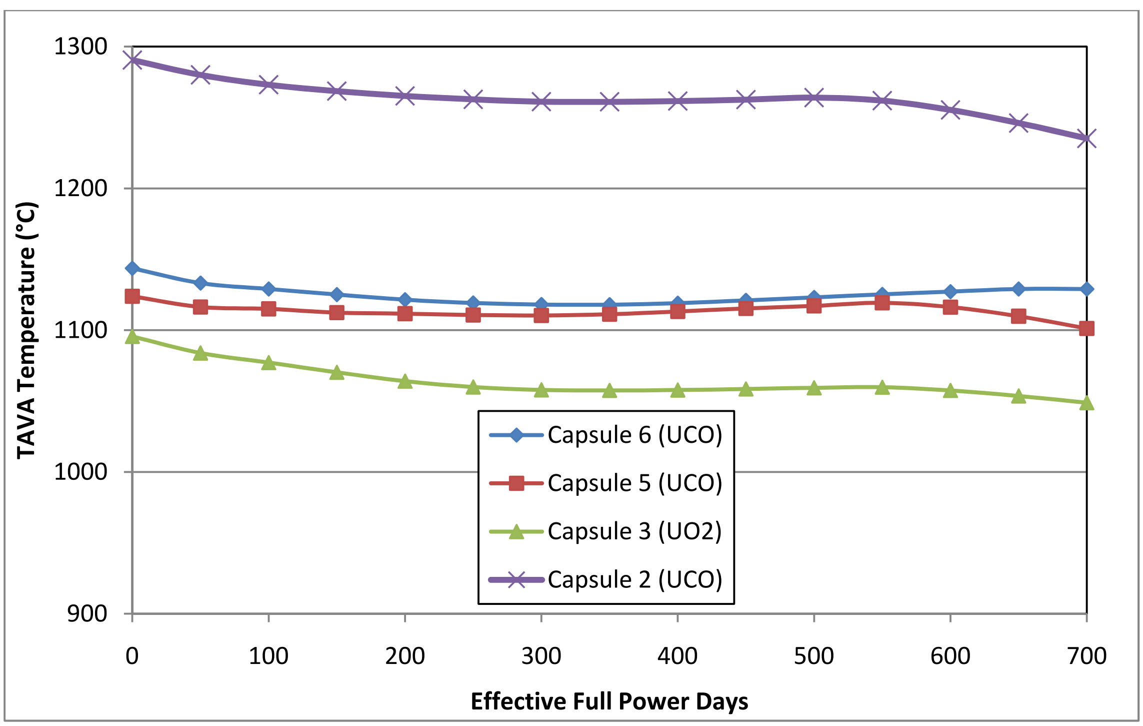

Figure 4-4. Time-average, volume-average (TAVA) temperature as a function of effective full power days (EFPD).

4.3 Fuel Burnup

The intent of the test objectives and test specifications is for the fuel to obtain a substantial fraction of burnup within a reasonable amount of time (on the order of 2.5 to 3 calendar years). As such, the test specification lists a burnup goal, not a requirement, for the majority of the fuel compacts (UCO and UO_2) to have a compact average burnup > 10% FIMA. However, a minimum compact average burnup is specified to be > 7% FIMA.

Figure 4-5 presents the currently calculated capsule average burnups and Figure 4-6 displays the maximum and minimum compact average burnups. These results indicate that after 600 EFPD of irradiation, the majority of the compacts will have reached the goal burnup of 10% FIMA, and that none of the compacts will be under the minimum specified burnup of 7% FIMA (at 600 EFPD, 36 of the 48 compacts have burnups > 10% FIMA, 45 compacts have burnups > 9% FIMA, 46 compacts have burnups > 8% FIMA and all compacts have burnups > 7% FIMA). Furthermore 41 of the 48 compacts have burnups > 10% FIMA at 650 EFPD and 45 of the 48 compacts have burnups > 10% FIMA at 700 EFPD.

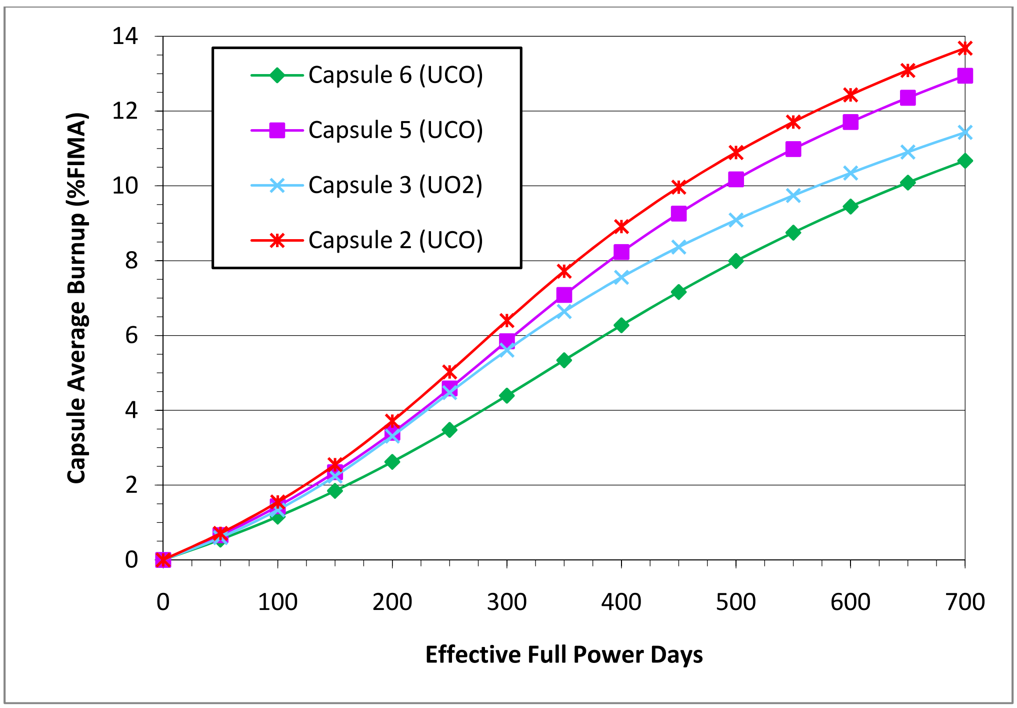

Figure 4-5. Capsule average burnups for AGR-2.

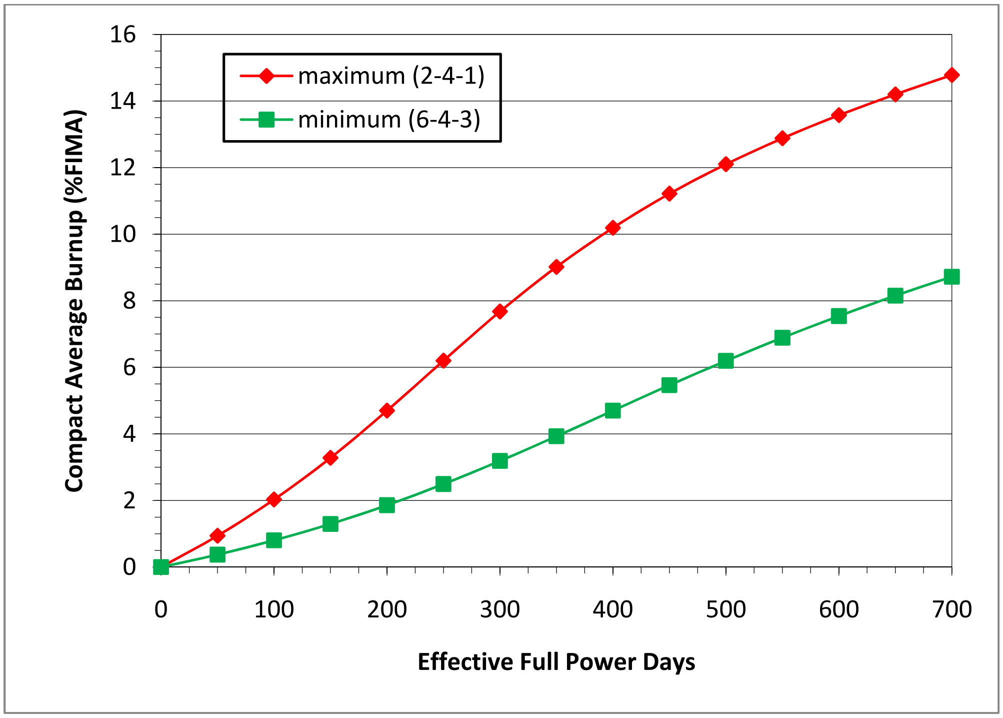

Figure 4-6. Compact average burnup for the maximum compact (Capsule 2 – Level 4 – Stack 1) and minimum compact (Capsule 6 – Level 4 – Stack 3).

4.4 Fast Neutron Fluence

The fast neutron fluence for each fuel compact is restricted by specification (Maki, 2010) to be > 1.5 × 10^{25} and < 5 × 10^{25} n/m^2 for E > 0.18 MeV. The upper limit is intended to bound expected VHTR service conditions while the lower limit is intended to ensure that the fuel pyrocarbon experiences the transition from creep-dominated strain to swelling-dominated strain.

Projections (Chang and Parry, 2010) for capsule average fast neutron fluences are presented in Figure 4-7 and fluences for the maximum and minimum compacts are presented in Figure 4-8. The data indicate that the minimum specified fluence is reached for all compacts slightly after 400 EFPD, and that the maximum specified fluence is not reached during an irradiation of 700 EFPD (the maximum reached is 4.0×10^{25} n/m^2 at 600 EFPD and 4.6×10^{25} n/m^2 at 700 EFPD for Compact 3-3-1).

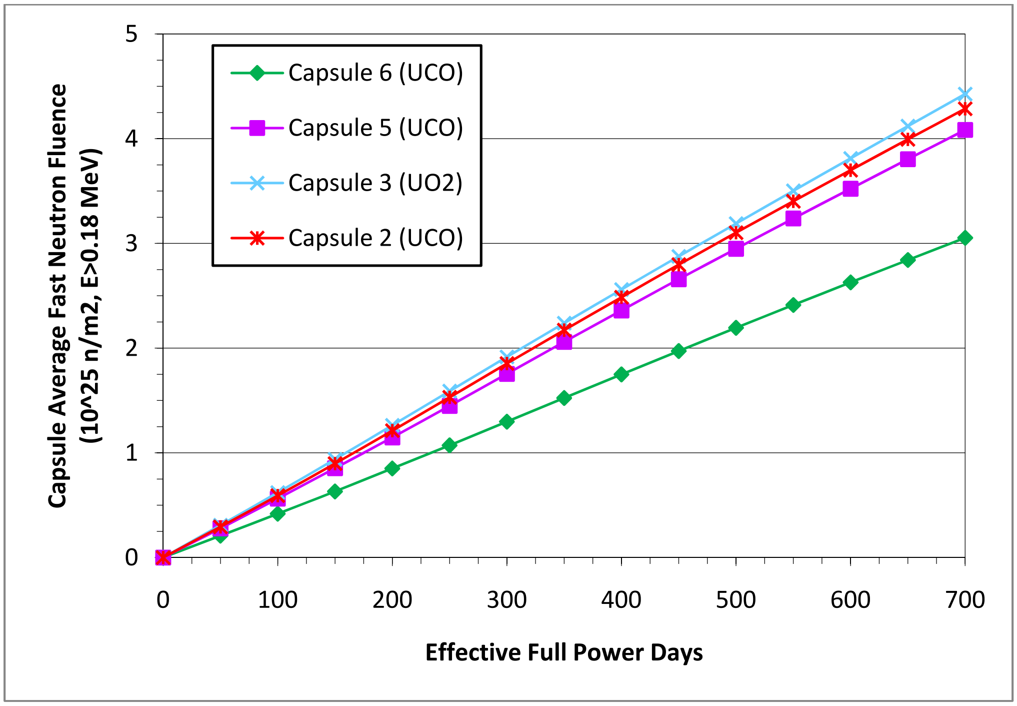

Figure 4-7. Capsule average fast neutron fluences for AGR-2.

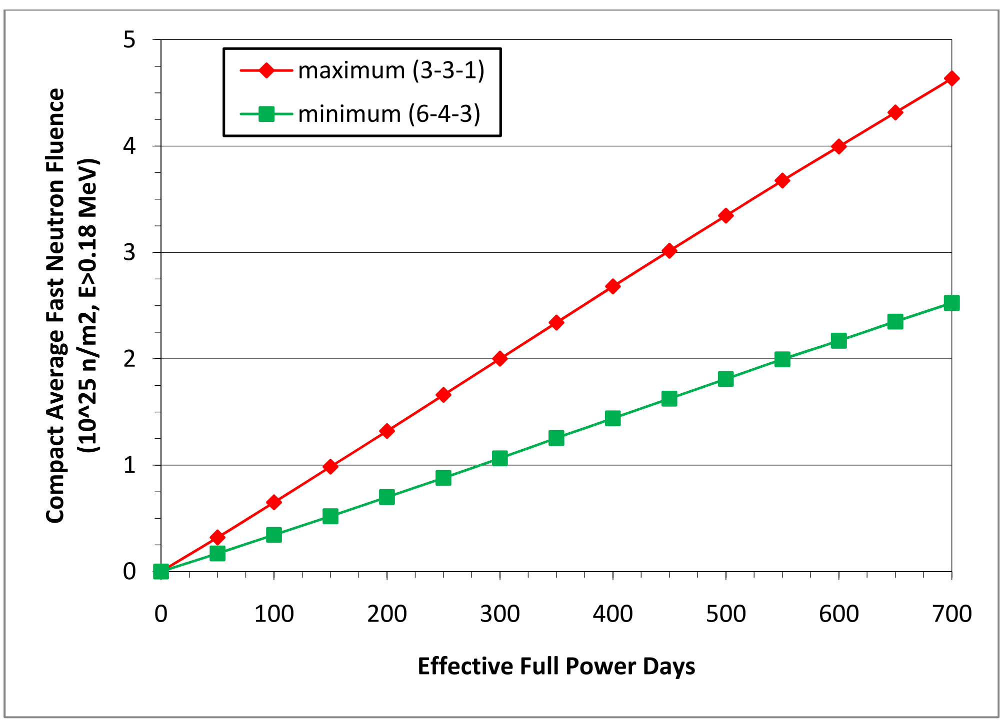

Figure 4-8. Compact average fast neutron fluence for the maximum compact (Capsule 3 – Level 3 – Stack 1) and minimum compact (Capsule 6 – Level 4 – Stack 3).

4.5 Irradiation Duration

The AGR-2 irradiation duration is scheduled to be 600 EFPD to come close to the AGR-1 duration of 620 EFPD, and to match the scheduling dictated by the Technical Program Plan (Simonds, 2010).

It is also constrained by the Technical Program Plan assumption to limit the irradiation test acceleration to under three times that expected in a real-time VHTR irradiation, and by the test specifications (Maki, 2010) for ancillary irradiation conditions. Since irradiating in a large B position in the ATR assures test acceleration is well under a factor of three, test duration is determined by evaluating the attributes of temperature, fast neutron fluence, and burnup. This approach must balance increasing duration with decreasing temperatures and increasing burnup and fast fluence.

A summary of selected AGR-2 irradiation conditions and associated test specifications are presented in Table 4-1. As evident from the table, and discussed in Section 4.2, AGR-2 will have achieved the minimum specified burnup level of 7% FIMA well within 2.75 calendar years. Irradiation could be extended to 650 or 700 EFPD to increase the number of compacts reaching 10% FIMA if such an extension does not result in the capsule temperatures ceasing to meet the test specifications and if it does not impact the overall AGR program schedule.

Table 4-1. Summary of AGR-2 irradiation conditions.

Parameter	Test Specification	Irradiation Duration[a] (EFPD)		
		600	650	700
Calendar years[b]	Not specified	2.75	3.0	3.25
Time-average, peak temperature[c] (°C)	≤ 1400 (UCO[d]) ≤ 1250 (UCO) ≤ 1150 (UO_2)	1377 1240 1137	1366 1240 1132	1353 1238 1126
Time-average, volume-average temperature[c] (°C)	≥ 1150 (UCO[d]) ≥ 1000 (UCO) ≥ 900 (UO_2)	1255 1127 1058	1246 1129 1054	1235 1129 1049
Fast fluence range[e] (10^{25} n/m^2, E>0.18 MeV)	1.5 – 5.0	2.2 – 4.0	2.4 – 4.3	2.5 – 4.6
Number of compacts with burnup < 7% FIMA	0	0	0	0
Number of compacts with burnup ≥ 10% FIMA	Goal of > 24	36	41	45
Test train average burnup (% FIMA)	Not specified	11.0	11.6	12.2

Notes: (a) Bold-italic entries do not meet test specifications.
(b) Assumes 220 EFPD per calendar year to account for ATR outages.
(c) Range is on a per capsule basis.
(d) Highest temperature capsule containing UCO.
(e) Range is on a per compact basis.

5. MEASUREMENT REQUIREMENTS

Several measurements are needed to demonstrate that the AGR-2 has reached the irradiation condition goals and test specifications. These conditions include time-average peak temperature, time-average volume-average temperature, fuel burnup, fast neutron fluence, and fission gas release. Because the fuel compacts cannot be directly instrumented (which may induce particle failures), burnup, neutron fluence, and fuel temperature will be determined by calculations that require supporting measurement data. Each of these measurement categories are discussed below.

5.1 Neutron Dosimetry

Both thermal and fast neutron fluence measurements will be made for the AGR-2 experiment. The purpose of these measurements is to provide neutron exposure data that will support the calculations of the average burnup, fast neutron fluence and fission product inventory of each compact. This support may consist of a set of point values used to normalize physics calculations.

Following irradiation and test train disassembly, the encapsulated flux wires, described in Section 3.3, will be removed from each capsule. After removal from the encapsulation, the flux wires will be prepared and counted for their neutron induced radionuclide activities. Counting uncertainties will stay within specified limits of ± 10%. Data collected from the neutron monitors will be corrected for decay according to standard procedures. Derived fast neutron fluence data will be further corrected to energies greater than 0.18 MeV. At all times, identification information (monitor type, serial number or similar code, original test train location) will remain with each neutron monitor.

5.2 ATR Parameters

ATR data that describes the core neutronic and thermal-hydraulic environment will be required to assist physics analysis (to calculate fuel burnup, heat generation rates and fast neutron fluences), assist thermal analysis (to calculate compact temperatures), and support temperature control.

The ATR is a light water moderated 93% enriched uranium fueled test reactor. As shown in Figure 2-1, the fueled core is arranged in a four-lobe clover leaf configuration. Each of the four corner lobes can be controlled at different powers to match the requirements of various in-pile experiments. ATR is rated at a total thermal power of 250 MW, but the reactor is normally operated in the range of 105 to 115 MW to meet most experiment needs.

ATR data that will be provided include individual lobe powers, shim cylinder positions, and core inlet temperatures. These data are recorded, and backed up on a separate storage device, once every minute.

5.3 Temperature Measurements

Temperature measurements will be performed by TCs terminating within the graphite sample holders of each capsule. These measurements will support thermal analyses of the test train, which will ultimately determine fuel temperatures and support temperature control of the experiment. For this function, one TC per capsule is designated as the control TC. Measurements from the control TCs provide feedback to the automated sweep gas control system, which adjusts gas blend to maintain reference temperatures.

AGR-2 TCs have at least an-installed accuracy of ± 2% of reading as required by the test specification. During normal and abnormal operation, TC data are recorded, and backed up on a separate storage device, once every minute.

5.4 Sweep Gas Parameters

In addition to the TC measurements, several sweep gas parameters are required for thermal analyses and temperature control. These include pressure, mass flow rates of each constituent gas, and moisture content. Sweep gas pressures and constituent mass flow rates that determine gas mixture ratios will be used in physics and thermal analyses of the test train. Moisture content measurements (measured on the outlet side of the capsule and compared to the gas supply verification measurement) provide indicators of capsule integrity.

Capsule inlet pressure is measured to within ± 0.007 MPa (± 1 psi) with constituent mass flow rates measured within 1% RMS (root mean square). Moisture data are converted to ppm-vol (parts per million by volume) relative to 15 psi. These data are recorded, and backed up on a separate storage device, once every minute.

5.5 Fission Gas Release Monitoring

Fission gas release measurements provide indicators of fuel irradiation performance. Gross radiation monitors continuously measure the sweep gas from each capsule to indicate fuel particle failures. Spectrometer detectors measure radionuclide concentrations to determine release rate to birth rate (R/B) ratios of selected nuclides. R/B values provide indicators of initial fuel quality and also provide indications of fuel failure.

The gross radiation monitors have sufficient sensitivity to detect every fuel particle failure, up to and including the first 250 failures, from each capsule. These fuel particle failures are indicated by a rapid rise and drop, or spike, in the measured count rate. Such spikes are a result of a sudden release of stored fission product inventory. Measured spectra are automatically stored and backed up.

The spectrometer detector systems measure the concentrations of various krypton and xenon isotopes in the sweep gas from each capsule. During normal operation, 8-hour counting intervals are used to measure the concentrations of Kr-85m, Kr-87, Kr-88, Kr-89, Kr-90, Xe-131m, Xe-133, Xe-135, Xe-135m, Xe-137, Xe-138, and Xe-139. These concentrations are converted to fuel release rates, which are used with calculated birth rates to determine R/B ratios. Measured spectra are automatically stored and backed up.

During reactor outages, the capsules are swept with pure helium-4. This sweep gas is then analyzed for Xe-133 and Xe-135. These xenon concentrations are used to calculate concentrations of their parent iodine isotopes. The presence of fission product iodine is also an indicator of fuel performance.

5.6 Data Validation and Qualification

Measured data are evaluated for validation and then qualified for use. The NGNP Data Management and Analysis System (NDMAS) processes the data for this purpose. The parameters are captured and processed by NDMAS are: fuel irradiation data (TC readings, sweep gas compositions, flow rates and pressures, and moisture monitor readings), FPMS data (isotopic release data and gross gamma counts), and ATR operating conditions data (lobe powers, control cylinder positions, neck shim positions, and control rod positions).

6. OPERATIONAL REQUIREMENTS

All operational activities associated with the AGR-2 experiment comply with all applicable INL and ATR standard procedures. These activities also comply with all safety and quality assurance requirements outlined in Section 7. Activities requiring special or unique consideration are identified below.

6.1 Pre-irradiation

Prior to final test train assembly, confirmatory physics and thermal analyses will be made using actual fuel characterization data, expected ATR operating conditions, and material properties for the borated graphite sample holders. Based on these results, the control gas gap width may be re-optimized and stress analysis of the test train may be re-evaluated.

Following receipt and inspection of the AGR-2 fuel compacts, UCO and UO_2 compacts are selected for irradiation based upon integrity and dimensions. The sample holders in each capsule are specifically bored to accept these compacts based upon compact diameters.

After assembly, test train and fission product monitor components and sub-systems undergo inspection, testing, and calibration as-needed. This includes, but is not limited to, leak testing of all pressure boundaries and gas lines and continuity checks of all TCs. Following these activities, a review is conducted and any findings corrected.

Following successful completion of the review and obtaining all appropriate ATR approvals, the AGR-2 test train is inserted into the B-12 position of the ATR. Air within the lead and gas lines is purged and the final component inspections performed.

6.2 Irradiation

During irradiation, temperature control is automatically maintained by the gas control system. This system requires temperature feedback from a control TC within each capsule. Should a control TC fail, a previously selected back-up TC within the same capsule will be used as the control TC and the reference control temperature reset based on thermal analysis calculations. Should all TCs fail within a capsule, results from physics and thermal analyses supported by the operating history of an adjacent capsule will be used to manually set the gas blends of the affected capsule. Ultimately, should all TCs fail within the test, temperature control may be based on predictive thermal analyses, augmented by analyses of fission product gas release which is sensitive to temperature.

Thermocouple drift will be monitored by analyses. With actual gas mixes and predicted heat generation rates from physics analyses, the thermal model will be adjusted and calibrated to TC readings during the start of the first irradiation cycle (about two days after reactor start-up so that xenon equilibrium is reached). Thereafter, thermal model results will be compared to the TC readings. Should the difference between model predictions and actual readings of a control TC differ by more than 50°C (about 4 to 5% of reading), control set points for the gas mix system will be adjusted to compensate for the TC drift.

The current ATR planning includes two PALM cycles during the span of the AGR-2 irradiation. These PALM cycles are scheduled to last about 14 EFPD and are planned after about 240 and 450 EFPD of irradiation. Prior to the PALM cycles the AGR-2 test train will be removed from its B-12 position to limit the higher power irradiation. It will then be relocated either to the I-24 position with the same control and monitoring capabilities or in the ATR canal. It is expected that the power will be decreased by a factor of about two in the I-24 position compared to the B-12 position, limiting higher power irradiation within a fairly easy-to-handle relocation.

During any switch in control gas (to pure helium-4, to a neon helium mix, etc.), flows of the common plenum gas and each control gas will be appropriately adjusted to ensure that continuity is maintained in the pressure differential between the common plenum and each capsule. This ensures that cross flow between capsules is avoided. After each ATR shutdown and during the entire outage, the control gas will be switched to pure helium-4 for each capsule, and the helium will continuously flow through each capsule at the nominal operating flow rate. The plenum flow will also be maintained at its nominal operating flow rate.

Should a capsule experience excessive fuel particle failures, on the order of 5% or approximately 2,000 UCO particles or 1,000 UO_2 particles, sweep gas to the capsule will be set to consist of 100% helium-4. The helium sweep gas will be maintained at the nominal operating flow rate until the end of the irradiation.

Indicators of moisture ingress (sweep gas outlet moisture content higher than inlet content) will be closely monitored. Past experience has shown that once the presence of moisture is detected, the content rapidly increases. Should a rapid increase in moisture be observed in a capsule, the test train may be removed from the reactor at the next scheduled reactor outage to avoid significant water-graphite interactions possibly compromising other capsules via gaps that may form around the thru tubes (because of reactions between steam and Neolube).

Program participants may be able to view time-series data on-line via a secured site. Viewable data should include at least TC measurements, sweep gas parameters, and gross radiation monitor count rates. Content and format for this possible data presentation has not been fully defined.

As a result of cycle-to-cycle variations in ATR lobe powers, accumulated burnup and fast neutron fluence for the AGR-2 test articles must be periodically updated based on as-run data. These as-run physics data reports will be issued after the end of each reactor cycle to the test completion.

6.3 Post-irradiation

The AGR-2 test train will be removed from the reactor after completion of the irradiation. For removal, the TCs and gas lines will be disconnected at the reactor vessel penetration flange (where the leadout passes through the reactor wall). The gas lines will then be capped and a cover installed on the test train leadout flange. The entire test train will then lifted from the B-12 test position and passed through the transfer chute to the ATR canal.

After completion of the irradiation, the test train will cool in the canal for about three months before being transferred to a hot cell for disassembly. Preliminary PIE will be conducted during and immediately after disassembly. Plans for follow-on detailed PIE have not yet been finalized but they should roughly be similar to that proposed for the AGR-1 experiment (Demkowicz, 2010).

Within a year of test completion, a Final Irradiation Test Results report will be issued. Results from PIE and safety testing will be documented separately after the completion of those activities.

7. SAFETY AND QUALITY ASSURANCE

7.1 Safety

The design, fabrication, installation, operation, and disassembly activities of the AGR-2 experiment comply with all applicable health, safety, and environmental requirements. These activities and their corresponding requirement directives are listed in Table 7-1.

Table 7-1. AGR-2 safety requirements.

Activity / Component	Requirements
Design, installation and operation of test lead	ATR Technical Safety Requirements Upgraded Final Safety Analysis Report
Capsule containment tube	ASME Boiler and Pressure Vessel Code
Mechanical design	Applicable sections of ASME and AWS Codes
Nuclear materials accountability	Applicable DOE orders
Radioactive material shipments	Applicable DOE orders

7.2 Quality Assurance

Quality assurance activities associated with the AGR-2 experiment comply with all applicable requirements set forth in:

- INL Quality Assurance Program based on ASME NQA-1 2000
- AGR Quality Program Plan, PLN-1468
- Reactor Technology Complex (INL) Site Specific quality assurance Implementation Procedures and Forms.

Activities requiring specific quality requirements include, but are not limited to the following:

- Capsule design review
- Capsule fabrication
- Component and system operational testing
- Test calibration
- Operational procedures
- Computer software control

- Irradiation data collection
- Neutron monitor analysis
- Fission product gas analysis
- Data management
- Data validation
- Reporting.

8. PROGRAM CONSTRAINTS AND SCHEDULE

Several possible programmatic constraints may affect the scheduling and accomplishment of significant activities presented in this test plan. Some of these constraints are discussed below.

The AGR-2 test train was ready to insert into ATR on June 16, 2010. Irradiation started on June 23, 2010 and is scheduled to run until March 26, 2013.

The irradiation duration is planned for approximately 2.75 calendar years. This duration may be shortened because of significant test train or fuel failures or lengthened to gain more fuel performance data with increased burnup. Duration to achieve targeted burnups depend on ATR operation where lobe powers are adjusted each cycle for the needs of various experiments, including PALM cycles.

Two of these PALM cycles are planned during the scheduled 600 EFPD irradiation of AGR-2, respectively after 240 and 450 EFPD of irradiation, and they are expected to last about 14 EFPD each.

A schedule indicating major activities for the AGR-2 irradiation test is shown in Figure 8-1.

	Start	Finish	06	07	08	09	10	11	12	13	14	15	16
Fuel Fabrication	10/2/06	4/9/10											
Design & Assembly	5/5/08	6/11/10											
Irradiation	6/23/10	3/26/13											
PIE & Safety Testing	3/27/13	4/19/16											
Data Analysis & Reporting	10/2/06	10/4/16											

Figure 8-1. Schedule for AGR-2 irradiation activities.

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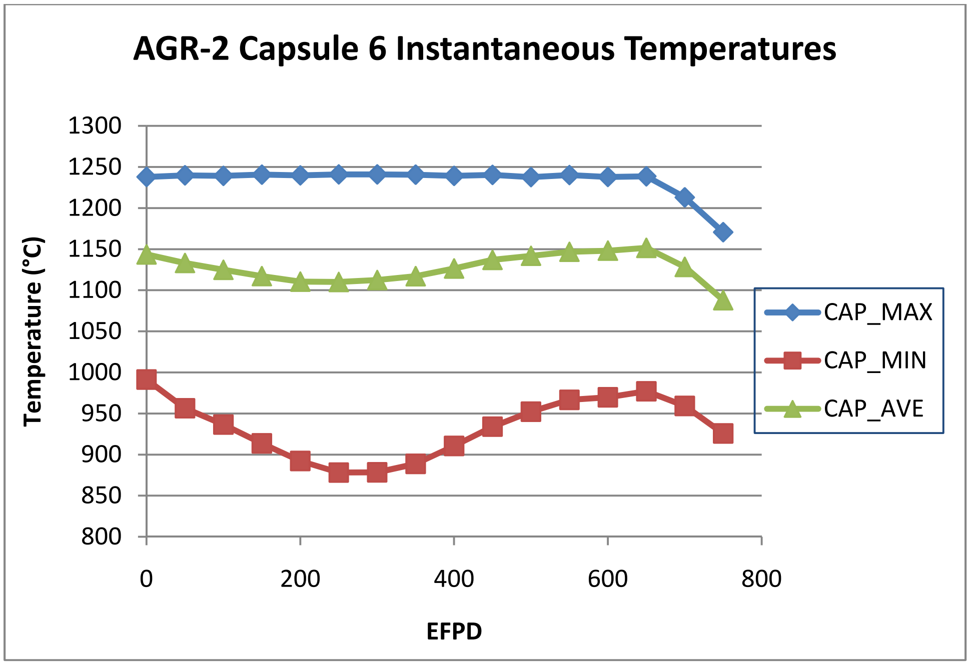
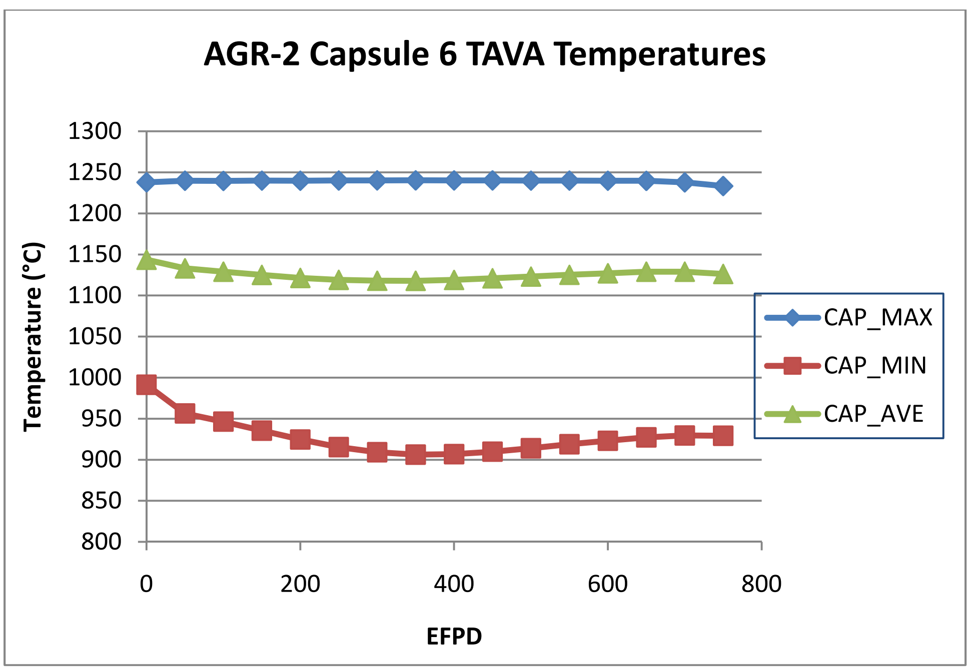
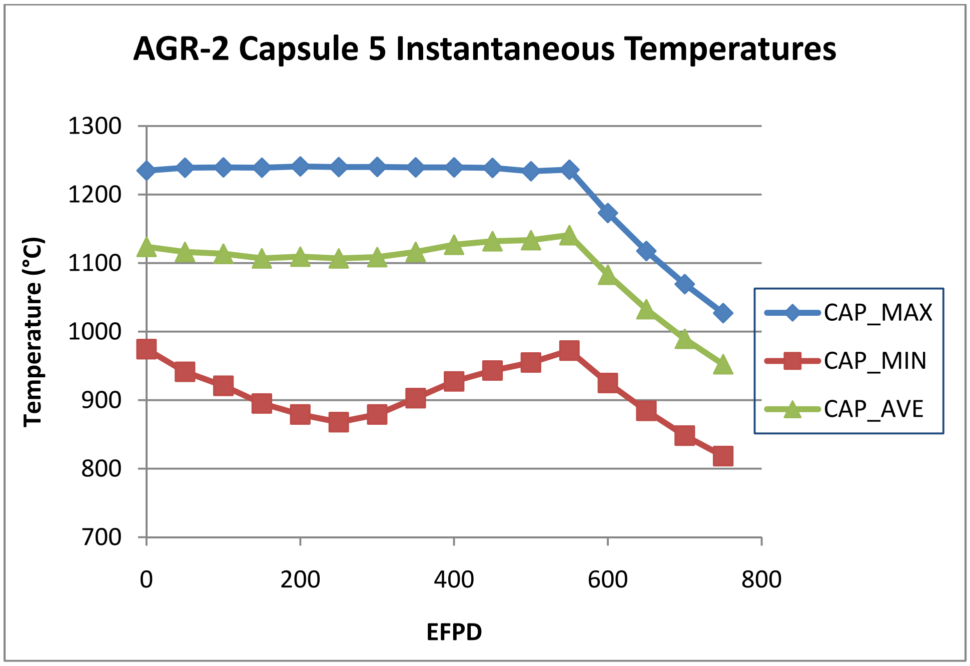
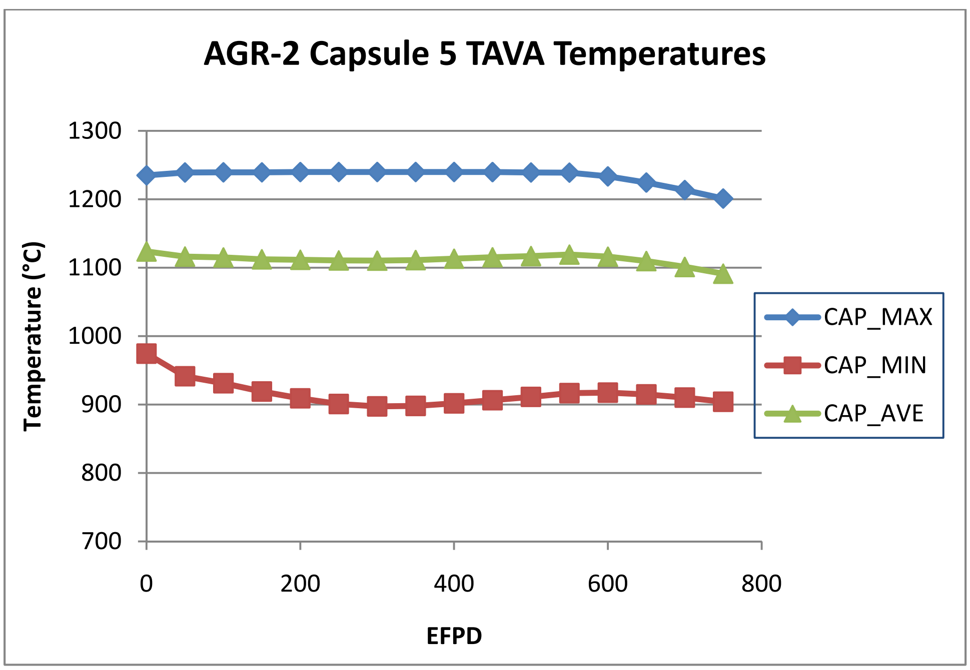
10. APPENDIX

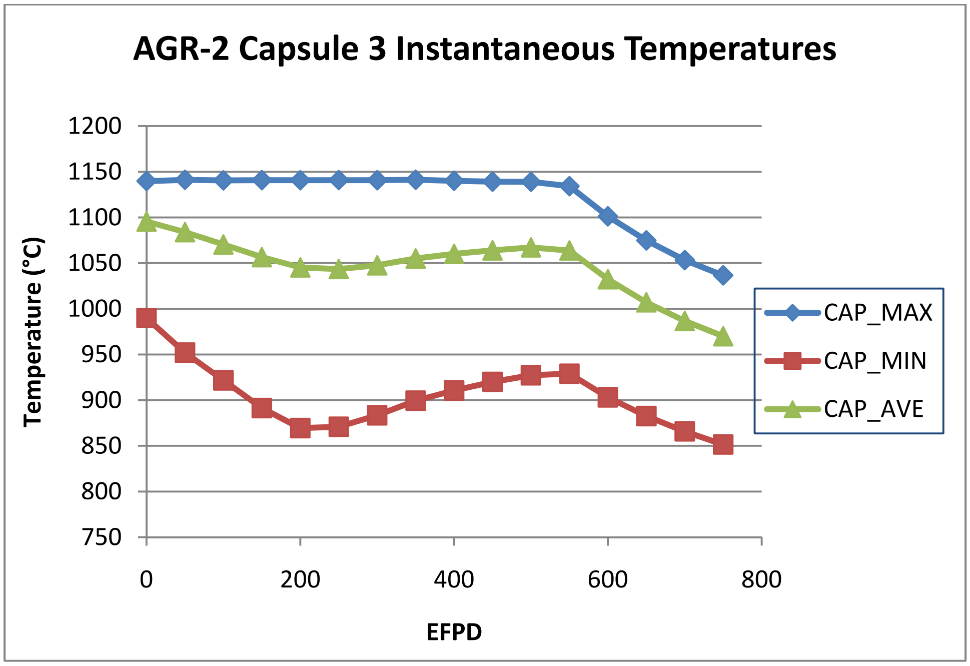
This Appendix presents instantaneous and time-average, volume-average (TAVA) temperature profiles throughout irradiation for each of the four capsules loaded with U.S. fuel. The maximum temperatures of the "Instantaneous Temperatures" profiles correspond to the instantaneous peak temperatures. Similarly, the maximum temperatures of the "TAVA profiles" correspond to the time-average, peak temperatures.

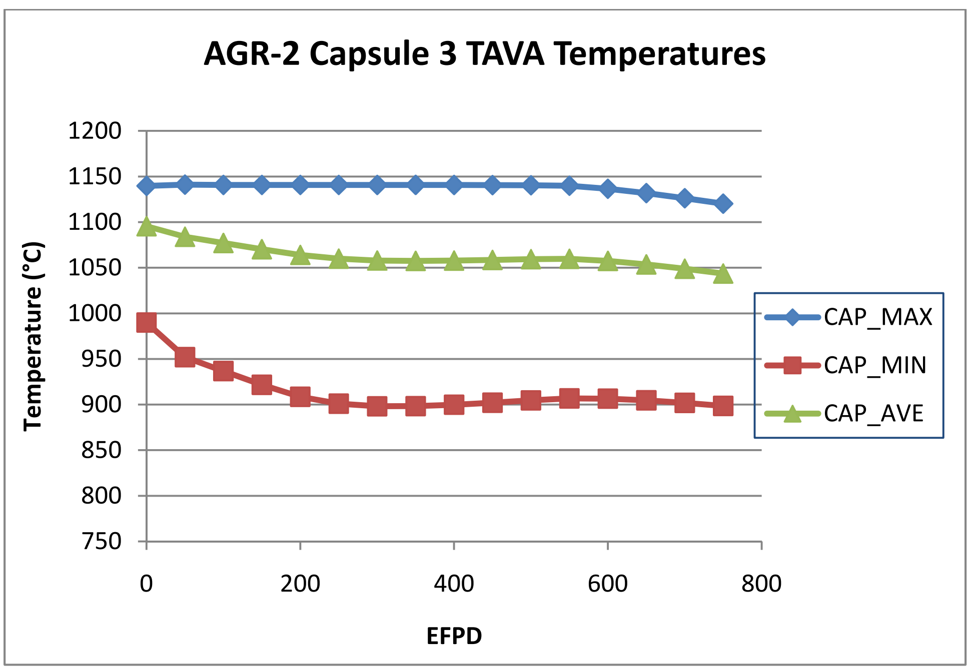


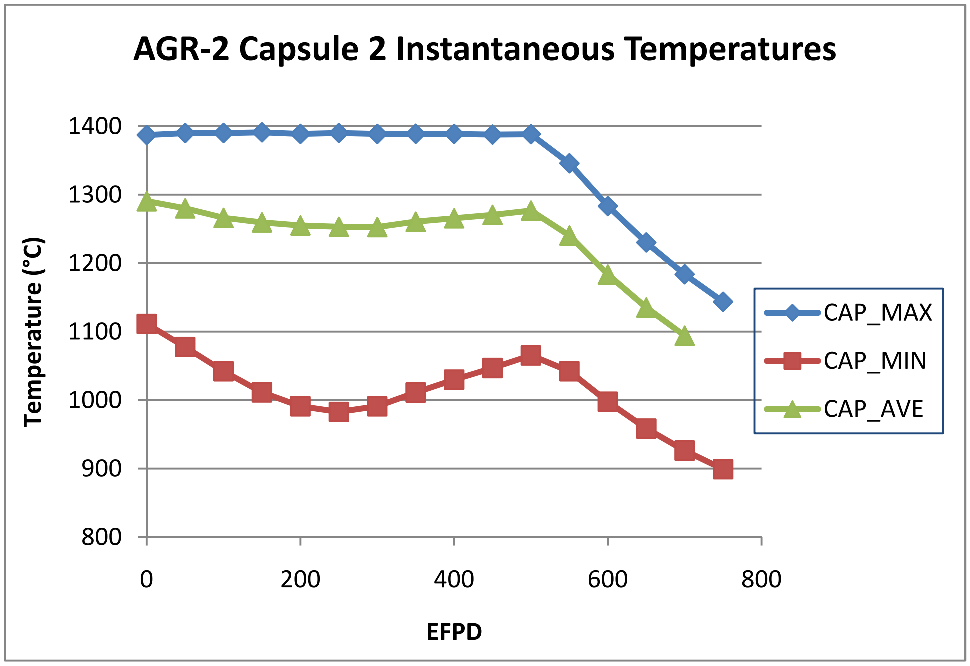


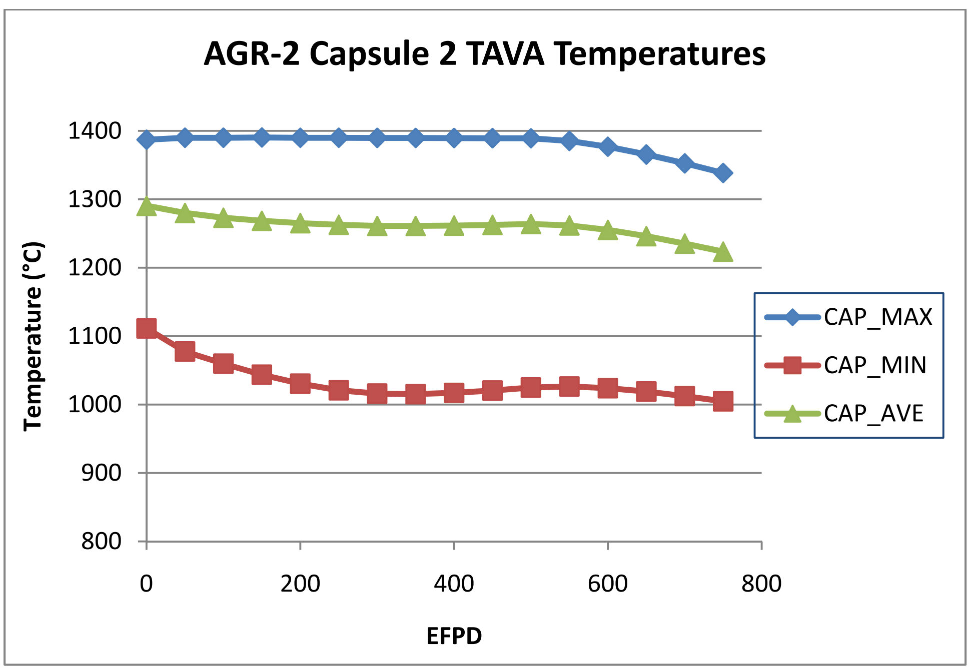

AGR-2 Capsule 5 Instantaneous Temperatures
Temperature (°C)
1300
1200
1100
1000
900
800
700
0
200
400
600
800
EFPD
CAP_MAX
CAP_MIN
CAP_AVE


AGR-2 Capsule 5 TAVA Temperatures
Temperature (°C)
1300
1200
1100
1000
900
800
700
0
200
400
600
800
EFPD
CAP_MAX
CAP_MIN
CAP_AVE


AGR-2 Capsule 3 Instantaneous Temperatures
Temperature (°C)
1200
1150
1100
1050
1000
950
900
850
800
750
0
200
400
600
800
EFPD
CAP_MAX
CAP_MIN
CAP_AVE


AGR-2 Capsule 3 TAVA Temperatures
Temperature (°C)
1200
1150
1100
1050
1000
950
900
850
800
750
0
200
400
600
800
EFPD
CAP_MAX
CAP_MIN
CAP_AVE


AGR-2 Capsule 2 Instantaneous Temperatures
Temperature (°C)
1400
1300
1200
1100
1000
900
800
0
200
400
600
800
EFPD
CAP_MAX
CAP_MIN
CAP_AVE


AGR-2 Capsule 2 TAVA Temperatures
Temperature (°C)
1400
1300
1200
1100
1000
900
800
0
200
400
600
800
EFPD
CAP_MAX
CAP_MIN
CAP_AVE