GAF 000013

PC-000384/2

HTGR/RERTR FUEL MATERIALS CHARACTERIZATION AND PACKAGING REPORT

prepared for

GA HOT CELL D&D PROJECT
CONTRACT NO. DE-AC03-95SF20798
PROJECT NO. 7340

APRIL 2002

GENERAL ATOMICS

GENERAL ATOMICS

GA 2175 (REV. 9/89)

PROJECT CONTROL ISSUE SUMMARY

	DOC. CODE	PROJECT	DOCUMENT NO.	REV.
	RGN	7340	PC-000384	2

TITLE:

HTGR/RERTR Fuel Materials Characterization and Packaging Report

CM APPROVAL/ DATE	REV	PREPARED BY	APPROVAL(S) RESOURCE/ SUPPORT	APPROVAL(S) PROJECT	REVISION DESCRIPTION/ W. O. NO.
ISSUED 2 JAN 12 1995	0	S. Su 1/10/95 J. S. Greenwood 10JAN95	V. Nicolayeff 1/10/95	G. C. Bramblett 1-11-95	Initial Issue W. O. 7310.012.101
4 RELEASED AUG 14 1997	1	J.S.Greenwood 13AUG97	V.Nicolayeff 8/13/97	G.C.Bramblett for 8/14/97	Complete revision to incorporate fuel packaging specifications W.O. #07340.800.20000
ISSUED 2 APR 23 2002	2	J. Greenwood 4/22/02 V. Barbat 4/22/02	V. Nicolayeff 4/22/02	R. DeVelasco for 4/22/02	General revisions to address INEEL comments W. O. #07340.800.20000

CONTINUE ON GA FORM 2175-1

* See List of Effective Pages

☐ GA PROPRIETARY INFORMATION

☒ NO GA PROPRIETARY INFORMATION

LIST OF EFFECTIVE PAGES

TABLE OF CONTENTS

LIST OF FIGURES

LIST OF TABLES

1.0 INTRODUCTION

1.1 Purpose of IFM Characterization

In support of ongoing Hot Cell Decommissioning Project efforts by General Atomics (GA), it has been necessary to fully characterize the DOE/NE Irradiated Fuel Materials (IFM), which were remotely packaged and removed from the Hot Cell Facility (HCF) Bldg. 23, and subsequently transported to temporary storage in Bldg. 30, in December 1995 then to Bldg. 31 in January 2000. The removal of the IFM from the HCF was necessary to allow the Hot Cell Decommissioning Project efforts to proceed in accordance with planned schedules. The objective of this report is to definitively quantify, describe, and document the pertinent characteristics of the stored IFM and associated packaging, in order to provide sufficient supporting data to meet the receipt criteria documentation requirements of the Idaho National Engineering and Environmental Laboratory (INEEL).

1.2 Contractual Requirements

GA Hot Cell Decommissioning Project work is being performed to meet the requirements set forth in Ref. 1-1. The specific requirement of IFM characterization, the results of which are documented herein, is outlined in Task 16 of Ref. 1-1.

1.3 Consolidation and Packaging of IFM

In November 1995, the subject IFM was remotely inspected, inventoried, and packaged in the GA Hot Cell Facility for the purpose of physical removal of the material from that building and the temporary storage on the GA site. As part of this process, the IFM was separated by fuel type into two packaging groups: one IFM group composed of the High-Temperature Gas-cooled Reactor (HTGR) type fuel entities (designated as HTGR IFM), and the other IFM group composed of the Reduced-Enrichment Research and Test Reactor (RERTR) type fuel, designated as RERTR IFM. The HTGR IFM packaging involved the physical consolidation of discrete HTGR entities, each of which had originated from specific DOE-sponsored fuel test irradiation programs conducted by GA. These HTGR IFM entities, held at GA under the HTGR Advanced Fuel Base Program (DOE Project No. L-AF-2050-100), had been separately controlled, inventoried, and retained at the GA Hot Cell Facility in shielded, retrievable storage. However, to facilitate handling and disposal of these items, GA obtained permission from the DOE to physically consolidate all stored HTGR IFM. This authorization was granted by DOE/OAK in July 1992. The packaging of each of the two IFM groups involved the remote handling, collection, loading, and weld-encapsulation of the IFM into 304 Stainless Steel Primary and Secondary Enclosures, the design of which allowed for the subsequent installation of the packaged IFM groups into separate, shielded storage casks. All IFM packaging steps were implemented in accordance with a GA Hot Cell operating procedure (Ref. 1-3).

1.4 Planned Disposition of IFM

Work to develop a strategy for the disposition of the subject IFM was initiated by General Atomics in early 1992, as a conjunctive effort with DOE/OAK. In 1993, DOE/OAK identified the Oak Ridge National Laboratory (ORNL), Oak Ridge, TN as the probable selected facility to receive and store the GA Hot Cell Facility IFM and directed GA to coordinate implementation of the transfer with ORNL. Accordingly, GA opened dialogue with ORNL at that time to define the scope of work involved in the proposed IFM transfer and to obtain the spent fuel receipt criteria and IFM packaging requirements for the ORNL spent fuel dry storage facility. In late 1995, DOE/OAK notified GA of a decision[1] by the National Spent Fuel Program, to designate the Idaho National Engineering and Environmental Laboratory Irradiated Fuel Storage Facility (INEEL IFSF), as the recipient for the GA Hot Cell Facility IFM. DOE/OAK subsequently directed GA to initiate dialogue with INEEL to coordinate transfer of all GA Irradiated Fuel Materials and Spent Fuel to that facility. Revision 1 of this report was prepared to reflect the programmatic change of recipient facility designation (from ORNL to INEEL), and included INEEL receipt requirements, set forth in Ref. 1-4. The current revision, (Rev. 2) reflects the comments received from INEEL on Rev. 1.

1.5 Temporary Storage of IFM at GA

Information related to the temporary storage of the subject IFM at GA has been previously documented and provided to the DOE in Ref. 1-5.

[1] The decision to direct the GA IFM to INEEL was made to comply with a Consent Order/Settlement Agreement on Spent Fuel and Nuclear Waste, reached 17 October 95 between the DOE, the U. S. Navy, and the State of Idaho, (i.e., Batt Agreement). In accordance with the terms of this agreement, shipments of IFM from GA to INEEL cannot start before 1 January 01.

2.0 SCOPE

The scope of this characterization effort, and the subject of this analytical report, encompasses the entire quantity of previously-irradiated, highly-radioactive nuclear fuel test specimens, which had been previously stored in the GA HCF. These IFM samples had been collected and retained in the HCF archival fuel storage inventory during a succession of Hot Cell Post-Irradiation Examination (PIE) projects, conducted by GA in support of various DOE-sponsored fuel development programs over the 30+ year operating history of the HCF. A preliminary inventory report on the subject IFM was prepared by GA and submitted to the U.S. Department of Energy in 1993, in fulfillment of a contract deliverable requirement (Ref. 2-1).

The subject IFM samples are associated with two DOE-sponsored reactor fuel development programs conducted by GA between ~1967 and ~1987. Each of these two programs involved the in-pile reactor exposure of discrete fuel material type groups, which were irradiated as either fuel test capsules or fuel test elements. The two fuel material types serve as a logical basis for the categorization of the subject IFM throughout this report, under the following designations:

- HTGR IFM (High-Temperature Gas-cooled Reactor Irradiated Fuel Materials), and
- RERTR IFM (Reduced-Enrichment Research & Test Reactor Irradiated Fuel Materials)

A summary of the individual HTGR IFMs constituent design data is presented in Table 2.1. The consolidated data for all of the individual HTGR IFMs is presented in Table 2.2.

A summary of the constituent design data of the different RERTR IFM is presented in Table 2.3. The consolidated data for all of the individual RERTR IFMs is presented in Table 2.4. The gravimetric composition of the cladding is presented in Table 2.5.

The purpose of this report is to provide sufficient data to allow INEEL to approve shipment of the subject IFMs. Based on the data herein and approval of INEEL to allow GA to ship the fuel, GA can then enter into contract negotiations with various NRC licensed Type B shipping cask suppliers. Cask suppliers will only negotiate with GA once an agreement to ship is reached. Once a contract with a cask supplier is in place, GA shall then complete remaining items in the Fuel Required Shippers Data Form (RSD, INEEL Form 434.28) and then complete the Packaging Required Shippers Data Form (RSD, INEEL Form 434.29) and provide this data to INEEL prior to the actual shipment. Completed forms are included in an attachment to this report and are completed based on the data provided herein. The description of the cask will be completed when a supplier is named and the activation levels of the IFM will be calculated to be current as of the time of shipment.

A summary of the DOE-sponsored irradiated fuel test element programs that generated the IFM samples is presented in Table 2.6.

Table 2.1. HTGR Fuel Constituent Design Data

ITEM	CHEMICAL/PHYSICAL FORM (as manufactured)	DIMENSIONAL/GRAVIMETRIC DATA (as manufactured)
FUEL PARTICLES (Kernels)	Solid, spheridized, high-temperature sintered fully-densified, ceramic kernel substrate, composed of: UC_2, UCO, UO_2, $(Th,U)C_2$, or $(Th,U)O_2$. In general, HTGR fuel kernels were manufactured by the GA-developed VSM inert-gas high-temperature drop furnace method. Note: These kernels were used in the Fuel Particles (Coatings). There are no uncoated particles in this shipment.	Spherical geometry; ~200 → 1000 µm kernel dia. ~720 µm, dia, (typ).
FUEL PARTICLES (Coatings)	Solid, spherical, isotropic, discrete multi-layered fuel particle coatings, successively applied in a gaseous, high-temperature chemical-vapor-deposition fluidized-bed system over fuel kernel substrate. The chemical composition of the coatings includes pyrolytic-carbon (PyC) & silicon carbide (SiC). Note: ~33 vol-% of the HTGR IFM is composed of loose fuel particles.	Spherical geometry; ~250 → 1000 µm particle dia. ~20 → 80 µm coating thickness. 3 → 5 coatings/particle (typ).
FUEL COMPACTS (Rods)	Multi-coated, ceramic fuel particles, bound in solid, cylindrical, injection-molded, high-temperature heat-treated compacts. The fuel compact matrix is composed of carbonized graphite shim, coke, & graphite powder. Note: ~65 vol-% of the HTGR IFM is composed of fuel compacts.	Right-circular cylindrical geometry; ~0.490" (1.25cm) compact dia. ~1.94" (4.93cm) compact ℓ.
FUEL PEBBLES (AVR only)	Multi-coated, ceramic fuel particles, bound in solid, spherical injection-molded, high-temperature heat-treated pebbles. The fully-cured binding matrix is composed of carbonized graphite shim, coke, and graphite powder. Note: There is a total of 2 ea. AVR fuel pebbles contained in the HTGR IFM or ~2 vol-%.	Spherical geometry; ~2.36" (6.00cm) sphere dia. ~20000 fuel particles/sphere.

Table 2.2. Consolidated HTGR Fuel Data

Fuel Material Volume:		~9556 cm^3 (~584 in^3)	
Fuel Material Mass:		10.668 kg.	
Overall Elemental Constituents for as-manufactured Fuel Material:	C	66.33 wt-%;	7075.55 g
	Th	18.34 wt-%;	1956.87 g
	Si	13.20 wt-%;	1408.37 g
	U	1.92 wt-%;	204.81 g
	O	0.21 wt-%;	22.40 g

Table 2.3. RERTR Fuel Constituent Design Data

ITEM	CHEMICAL/PHYSICAL FORM (as manufactured)		DIMENSIONAL/GRAVIMETRIC DATA (as manufactured)	
FUEL	20 ea. irradiated fuel elements; active fueled portion of elements are solid, cylindrical casting composed of homogeneous alloyed metal hydride as $Er(U,Zr)H_{1.65}$; 5 ea. @ 20 wt-% U, 78.6 wt-% Zr, 0.00 wt-% Er, 1.35 wt-% H, 0.00 wt-% C; 6 ea. @ 30 wt-% U, 68.0 wt-% Zr, 0.51 wt-% Er, 1.15 wt-% H, 0.33 wt-% C; 9 ea. @ 45 wt-% U, 53.1 wt-% Zr, 0.89 wt-% Er, 0.92 wt-% H, 0.10 wt-% C; The nominal value for ^{235}U enrichment for all RERTR fuel is 19.7%. Actual enrichments may vary by up to ± 0.1%.		Fuel:	0.512" (1.30 cm) OD x 22.05" (56.0 cm) length.
			Fuel Volume/ Element :	74.4 cm^3.
			Fuel Mass/ Element:	~473 g for 20 wt-% U elements. ~513 g for 30 wt-% U elements. ~619 g for 45 wt-% U elements.
ELEMENT COMPONENTS	Cladding:	Incoloy 800H extruded seamless tubing.	Cladding:	0.543" (1.38 cm) OD x 0.016" (0.041 cm) wall thickness.
	Top/Bottom Fixtures:	304SS casting.		
	Compression Spring:	Inconel 600.		
ELEMENT	20 ea. solid cylindrical welded-assembly irradiated fuel elements, of which 13 ea. are intact assemblies; the remaining 7 ea. elements have been previously sectioned for examination purposes; the component segments of each sectioned element were collected into separate aluminum tubes and the ends crimped.		Element:	0.543" (1.38 cm) OD x 29.92" (76.0 cm) length overall.
			Volume/Element:	133.6 cm^3
			Mass/Element:	665 g for 20 wt-% U 705 g for 30 wt-% U 811 g for 45 wt-% U

Table 2.4. Consolidated RERTR Fuel Data

Fuel Material Volume:		~7141 cm^3 (~436 in^3)
Fuel Material Mass:		10766.54 g
Overall Elemental Constituents in Fuel Material:	Zr U H Er C	62.42 wt-%; 6721.10 g 35.77 wt-%; 3850.66 g 1.08 wt-%; 116.02 g 0.59 wt-%; 63.32 g 0.14 wt-% C 15.44 g
neutron poisons		Er/U = 16.44 $g\text{-}kg^{-1}$

Table 2.5. Gravimetric Composition of Cladding & Non-fuel Component Materials in RERTR IFM

Cladding & Non-Fuel Component/ Material Composition	Component Mass, ($g\text{-}FHU^{-1}$)	Fe		Ni		Cr		Mn		Mo	
		wt-%	wt. (g)	wt-%	wt. (g)	wt-%	wt. (g)	wt-%	wt. (g)	wt-%	wt. (g)
Tubing/ Incoloy 800H	1607.1	40.0	642.8	35.0	562.5	25.0	401.8	0.0	0.0	0	0.0
Compression Spring/ Inconel 600	270.3	0.0	0.0	75.0	202.7	25.0	67.6	0.0	0.0	0	0.0
End Fittings/ 304SS	1538.7	69.0	1061.7	10.0	153.9	19.0	292.3	2.0	30.8	0	0.0
Spacer/ Molybdenum	17.3	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0	100.0	17.3
TOTALS, ($g\text{-}FHU^{-1}$)	**3433.4**		1704.5		919.1		761.7		30.8		17.3

PC-000384/2

Table 2.6: Summary of DOE-Sponsored General Atomics Fuel Test Element Programs Related to the HTGR & RERTR Irradiated Fuel Materials (IFM)

Fuel Test Capsule/ Fuel Test Element ID	Physical Form of Fuel Material (as manufactured)	Irrad. Reactor	Irrad. Start Date	Irrad. End Date	Report Ref. No.
FSV FTE-2	(Th,U)C_2, ThC_2, & ThO_2[TRISO] coated particles/compacts	FSV	~3/79	~1/84	5-1,5-2,5-3
FSV SURV	(3.6:1 Th:U)C_2 & ThC_2[TRISO] coated particles/compacts	"	7/3/76	2/1/79	5-1,5-4,5-5
PB FTE-3	UC_2, UO_2, (2.75:1 Th:U)C_2, ThC_2[TRISO], & ThC_2, ThO_2[BISO] coated particles/compacts	PB	7/11/71	1/7/72	5-6,5-11
PB FTE-4	UC_2, UO_2, (2.75:1 Th:U)C_2, ThC_2[TRISO], & ThC_2, ThO_2[BISO] coated particles/compacts	"	7/11/71	9/14/73	5-7,5-11
PB FTE-6	UC_2, UO_2, (2.75:1 Th:U)C_2, ThC_2[TRISO], & ThC_2, ThO_2[BISO] coated particles/compacts	"	7/11/71	11/30/74	5-8,5-11
PB FTE-14	UC_2, UO_2, (1:1 Th:U)O_2, ThC_2, ThO_2[TRISO], & ThC_2, ThO_2[BISO] coated particles/compacts	"	6/18/72	9/14/73	5-9,5-11
PB FTE-15	UC_2, UO_2, (1:1 Th:U)O_2, ThC_2, ThO_2[TRISO], & ThC_2, ThO_2[BISO] coated particles/compacts	"	7/6/72	10/31/74	5-9,5-11
PB FTE-16	(1:1 Th:U)C_2, ThC_2[TRISO] coated particles/compacts	"	7/6/72	10/31/74	5-10,5-11
AVR	(Th,U)C_2[TRISO] & (Th,U)O_2[BISO] coated particles encapsulated in isostatically injection-molded graphite 6 cm dia. spherical pebbles; 2 ea	AVR	~12/67	~6/68	5-12,5-13
P13P	UC_2, UO_2, (Th,U)O_2, ThC_2, ThO_2[TRISO] & ThC_2, ThO_2[BISO] coated particles/compacts	ETR	4/15/72	4/30/73	5-14
P13Q	UC_2[TRISO] & ThO_2[BISO] coated particles/compacts	ORR	8/22/73	2/23/75	5-15,5-16
P13R/S	UC_2[TRISO] & ThO_2[BISO] coated particles/compacts	GETR	12/10/73	10/31/74	5-17,5-18
P13T	UC_2, UCO[TRISO] & ThO_2[BISO] coated particles/compacts	ORR	5/10/75	7/5/76	5-19
P13V	UCO[TRISO] & ThO_2[BISO] coated particles/compacts	GETR	3/1/76	11/11/76	5-20,5-21
HB-2	UCO(WAR)[TRISO], ThO_2[Si/BISO], ThO_2[ZrC/BISO] coated particles/compacts	"	9/7/75	1/8/76	5-22
HRB-14	(Th,U)O_2, UO_2, ThO_2[TRISO] coated particles/compacts	HFIR	5/20/78	1/4/79	5-23
HRB-15A	(Th,U)O_2, UO_2, ThO_2[TRISO] coated particles/compacts	"	7/26/80	1/29/81	5-24
GF-3	(Th,U)O_2, ThO_2[TRISO], & ThO_2[BISO] coated particles/compacts	SILOE	1/31/74	7/25/75	5-25
DR-GB2	UC_2[TRISO] & ThO_2[BISO] coated particles/compacts	DRAGON	~3/75	~12/75	5-26,5-27
RERTR	Er(U,Zr)$H_{1.65}$ solid metal alloy w/Incoloy 800 cladding, 20 ea. fuel element assemblies	ORR	~12/79	~8/84	5-28

3.0 CHARACTERIZATION METHOD

The methodology employed for the characterization of the IFM is described in Ref. 3-1. The specific characterization parameters specified herein have been addressed, in order to meet the radiological characterization requirements identified in Ref. 1-4.

In preparing this report, existing, previously-published GA records, technical reports, test plans, program summaries, post-irradiation examination (PIE) reports, and other pertinent experimental data were examined and analyzed for applicability and incorporation herein. These documents, which are individually referenced in the following sections, collectively represent a comprehensive information database to support the characterization analyses herein. All data presented in this report which relate to the physical/chemical characteristics and irradiation history of the IFM, have been drawn from the aforementioned reference documents.

3.1 Methodology Utilized for Radiological Characterization of HTGR IFM

Radiological characterization of the HTGR IFM fuel capsules and fuel test elements was performed with the GARGOYLE computer code. The GARGOYLE computer code was chosen for the HTGR IFM because it is geared towards the HTGR fuel cycle, and has been used at GA and other DOE sites for HTGR fuel cycle analysis for the past 30 years. GARGOYLE is a zero-dimensional multi-group diffusion-theory burn-up and depletion code. The original GARGOYLE user's manual was issued in 1969 as a GA report. The code was later modified to include decay heat calculations using a large library of nuclear data, which includes cross-sections, fission-product yields, decay chains, decay energy data, etc. (Ref. 3-2).

The diffusion and depletion calculations in GARGOYLE are based on a homogeneous target composition. Therefore, the effect of flux self-shielding on the nuclide cross-sections in the actual heterogeneous fuel element geometry must be considered using self-shielding factors. For the HTGR IFM, the self-shielding factors have been generated for ^{232}Th and ^{238}U, and are contained in the GARGOYLE data library. Although these factors were developed for a specific HTGR fuel, they are applicable to the characterization study by adjusting other parameters, such as thermal power generated.

The GARGOYLE computer code can treat multiple geometric regions. For this study, however, a single homogeneous core region was used in calculating the multi-group flux spectra. The composition contained initial fuel loadings and moderator atom densities. The fuel loadings corresponded to the loadings in a given HTGR IFM capsule or test element. Typical moderator density (graphite moderator assumed in all cases) was used for spectrum calculations.

Input data required for GARGOYLE computer code calculations include irradiation history (i.e., EFPD and decay time), initial fuel loadings (both Th and U), core volume corresponding to the fuel loadings, and assumed core thermal power. The thermal power input was adjusted as appropriate in order to obtain the results consistent with the reported %FIMA or activity levels.

Conformation of the accuracy of the GARGOYLE code is based on the following:

- It was used to design the Fort St. Vrain core and accurately depicted the core. It was accepted by the NRC for use in calculating the core configuration.

- The data generated by GARGOYLE were compared with the output of ORIGEN, a program used for LWR fuel. Results were very favorable as reported in Refs. 3-4 and 3-5.

Results generated from GARGOYLE calculations for each fuel sample include:

- Target Material Actinide Content (in g),
- Target Material Fission Product Activity Levels (in Ci), and
- Target Material Decay Heat Content (in W).

3.2 Methodology Utilized for Radiological Characterization of RERTR IFM

For the RERTR IFM, similar characterization methods were used as those used for the HTGR IFM. The actinide content of the RERTR IFM however is well documented, and this information is used for characterization purposes for the fuel. The fission product and activation product inventory were determined using the ORIGEN2 code (Ref. 3-3). ORIGEN2 was chosen for the RERTR IFM analysis because it is a widely recognized code for fissile burn-up and depletion calculations. ORIGEN2 is a point depletion code, made available through ORNL, and is used primarily for LWR fuel burn-up analysis. The initial fuel material inventory, including cladding and the average irradiation time, was input into ORIGEN2. The RERTR IFM power level was chosen to produce the documented ^{235}U content at the end of the irradiation time given in the referenced reports. The actinide content resulting from ORIGEN2 calculations was not used because of the differences between LWR and ZrH_X spectra. While the spectral effect is important to the actinide content of the RERTR IFM, the effects are minimal for the fission products and activation products.

Calculated ORIGEN2 output includes results for:

- Target Material Fission Product Activity Levels (in Ci),
- Target Material Activation Product Activity Levels (in Ci), and
- Target Material Decay Heat Content (in W).

4.0 ASSUMPTIONS

The following assumptions were used in this Irradiated Fuel Materials (IFM) radiological characterization study:

- The IFM target entities were exposed to a continuous, straight burn for the reported Effective Full-Power Days (EFPD) duration, followed by decay from the end of irradiation to the Reference Decay Date of 1 January 96.

- Irradiation reactor neutron flux moderation was provided by graphite (of ~1.3 g/cm^3 specific gravity) for all HTGR IFM target entities.

- Irradiation reactor neutron flux moderation was provided by water for the RERTR IFM target entities.

- The applied self-shielding factors utilized for ^{232}Th and ^{238}U were taken from existing reference data, which was available in both the GARGOYLE and ORIGEN2 computer code libraries.

- For HTGR IFM target entities, one homogeneous active core region was assumed for calculation input. The smeared composition of the irradiation environment model incorporated reported target fuel loadings and graphite density, thereby simulating the fuel burn-up and depletion in a driver reactor.

- The peak fissile and fertile burn-up levels (i.e., Peak FIMA), which were previously documented in the corresponding engineering report references for the individual fuel target entities, were utilized for all calculations.

5.0 ANALYSIS

5.1 General

For analysis purposes, the two types of Irradiated Fuel Materials considered (i.e., HTGR and RERTR) were further divided into five Subgroups, based on target fuel type, fuel elemental composition, and reactor irradiation exposure conditions. These IFM Subgroups are identified and defined below. Also provided below are notes pertaining to the nuclear analysis computer code calculations performed for each Subgroup.

Subgroup *(i)*: FSV FTE IFM:
HTGR-type fuel irradiated as part of Fuel Test Element programs in the Fort St. Vrain Nuclear Generating Station Reactor; (see Section 5.3).

Subgroup *(ii)*: PB FTE IFM:
HTGR-type fuel irradiated as part of Fuel Test Element programs in the Peach Bottom Atomic Power Station, Unit 1 Reactor; (see Section 5.4).

Subgroup *(iii)*: AVR IFM:
HTGR-type fuel material irradiated in the Arbeitsgemeinschaft Versuchs-Reaktor Pebble-Bed Reactor; (see Section 5.5).

Subgroup *(iv)*: OTHER HTGR IFM:
High-Temperature Gas-cooled Reactor fuel material, irradiated as part of Fuel Test Capsule programs in one of several different irradiation facilities, including the ETR, ORR, GETR, HFIR, SILOE, or DRAGON Reactors; (see Section 5.6).

Subgroup *(v)*: RERTR IFM:
TRIGA-type Reduced Enrichment Research & Test Reactor fuel element assemblies, irradiated in the Oak Ridge Reactor; (see Section 5.7).

5.2 Assumptions

For IFM Subgroups *(i)*, *(ii)*, & *(iii)*, each of which are single reactor-specific, data concerning the neutron flux spectra, to which the target fuel specimens were exposed during reactor irradiation, were available from previously published information regarding the FSV, PB, and AVR reactors, respectively. Fissile burn-up analyses for these Subgroups were obtained with the GARGOYLE computer code (Ref. 3-2).

IFM Subgroup *(iv)* is composed of various HTGR-type Fuel Test Capsules, each of which have been irradiated in non-HTGR reactors. The effective driver reactor neutron flux spectra to which these target HTGR fuel specimens were exposed was therefore influenced by the structural graphite incorporated into the Fuel Test Capsule design. In order to model these special irradiation conditions for nuclear analysis calculations, it was assumed that the HTGR fuel entities making up IFM Subgroup *(iv)* were irradiated under normal FSV operating conditions; appropriate adjustments were then applied to the thermal power input, to yield similar neutron fluence and/or fissile burn-up levels.

For the analysis of IFM Subgroup *(v)*, i.e., RERTR material, which is a Incoloy 800 clad metal alloy TRIGA-type fuel design, the effective neutron flux spectrum to which the target material was exposed was normalized for light water moderation. The fissile burn-up analyses for this subgroup was obtained with the ORIGEN2 computer code. (Ref. 3-3)

In general, a trial and error approach was employed to adjust the thermal power input values to the code, in order to obtain the reported fissile burn-up levels.

The following pages in this section present the actual computer code input data utilized to calculate the post-irradiation isotopic activities and decay heat content of the individual IFM entities characterized herein. Pertinent references relating to each IFM entity are cited on each analysis sheet.

The general format of the information presented in this section is as follows:

5.X IFM Subgroup Designation

5.X.X Fuel Test Capsule/Element Identity

Sample Designation	=	Identity of Archival IFM Sample Entity
V_{ELE}, (cm^3)	=	Volume of Fuel Test Element/Capsule
$^{235}U\ Wt_{ELE,EOL}$, (g/ELE)	=	Post-Irradiation ^{235}U in Entire Fuel Test Element/Capsule
$^{235}U\ Wt_{SX,EOL}$, (g/SX)	=	Post-Irradiation ^{235}U in IFM Archival Sample of Fuel Test Element/Capsule
Normalization Factor, K_N	=	Fraction of Fuel Test Element/Capsule ^{235}U in IFM Archival Sample
V_{SX}, (cm^3)	=	Volume of IFM Archival Sample

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
Nuclide Species (or Material) contained in IFM Sample	Reported As-Manufactured Nuclide Mass (in grams) contained in IFM Fuel Test Element/Capsule[2]	Calculated As-Manufactured Nuclide Mass (in grams) contained in IFM Sample	Calculated IFM Target Atom-Density (in atoms per barn-centimeter)

Reactor:		Irradiation Reactor Identity, Full name of reactor given in Section 8.
$Start_{Irradiation}$:		Reported Irradiation Start Date
$End_{Irradiation}$:		Reported Irradiation End Date
EFPD, (d)	=	Period of Irradiation, in Effective Full Power Days
Φ_{FAST}, (nvt)	=	Peak Fast Neutron Fluence in Irradiation Reactor
Peak $FIMA_{Fissile}$	=	Peak Percent Fissile Burn-up in Target
Peak $FIMA_{Fertile}$	=	Peak Percent Fertile Burn-up in Target
ρ_P, (W/cm^3)	=	Average Power Density of Irradiation Reactor
$P(t)_{SX}$, (W)	=	Peak Power Generated in Target during Irradiation
References:		Applicable Engineering Reports Utilized for Calculation Input Data

There was no Limit-of-Error information in any of the referenced reports, therefore there is no information regarding this value in the data presented.

[2] The data for the as-manufactured values of the individual fuel items was taken from the referenced reports. There is no Limit of Error information in these reports, therefore none is reported.

5.3 FSV FTE IFM

5.3.1 FSV Fuel Test Element FTE-2

Sample Designation = FSV/FTE-2

V_{ELE}, (cm^3) = 89000.

^{235}U $Wt_{ELE,EOL}$, (g/ELE) = 2.04E+02

^{235}U $Wt_{SX,EOL}$, (g/SX) = 5.80E-01

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = (0.58g)/(204.0g) = 2.84E-03

V_{SX}, (cm^3) = $(V_{ELE})(K_N)$ = 253

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	1.09E+04	3.12E+01	3.19E-04
^{234}U	5.31E+00	1.50E-02	1.50E-07
^{235}U	5.10E+02	1.45E+00	1.47E-05
^{236}U	2.30E+00	7.00E-03	7.00E-08
^{238}U	2.99E+01	8.50E-02	8.50E-07
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.1%

Reactor: FSV

$Start_{Irradiation}$: 03/79

$End_{Irradiation}$: 01/84

EFPD, (d) = 488.8

Φ_{FAST}, (nvt) = 1.90E+21

Peak $FIMA_{Fissile}$ = 45.0 %

Peak $FIMA_{Fertile}$ = 1.0 %

ρ_P, (W/cm^3) = $(P_{AVG})(P)$ = (6.3)(1.17) = 7.37

$P(t)_{SX}$, (W) = $(V_{SX})(\rho_P)$ = (253)(7.37) = 1867

Refs: 5-1, 5-2, 5-3

5.3.2 FSV Surveillance Element

Sample Designation = FSV/SURV

V_{ELE}, (cm^3) = 89000.

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = 2.64E+02

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 6.80E+00

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE}) = (6.80g)/(263.76g) = 2.58E\text{-}02$

V_{SX}, (cm^3) = $(V_{ELE})(K_N) = 2295$

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	1.95E+03	NR	5.69E-05
$^{232}Th_{Fertile}$	1.08E+04	NR	3.16E-04
$^{232}Th_{Total}$	1.28E+04	3.30E+02	3.73E-04
^{234}U	3.45E+00	8.70E-02	9.75E-08
^{235}U	4.33E+02	1.12E+01	1.25E-05
^{236}U	1.32E+00	3.60E-02	4.02E-08
^{238}U	2.71E+01	6.99E-01	7.71E-07
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.1%

Reactor: FSV

$Start_{Irradiation}$: 07/03/76

$End_{Irradiation}$: 02/01/79

EFPD, (d) = 174.0

Φ_{FAST}, (nvt) = 1.0E+21

Peak $FIMA_{Fissile}$ = NR

Peak $FIMA_{Fertile}$ = NR

ρ_P, (W/cm^3) = $(P_{AVG})(P) = (6.3)(1.55) = 9.77$

$P(t)_{SX}$, (W) = $(V_{SX})(\rho_P) = (2295)(9.77) = 22340.$

Refs: 5-1, 5-4, 5-5

5.4 PB FTE IFM

5.4.1 PB Fuel Test Element FTE-3

Sample Designation = PB/FTE-3

V_{ELE}, (cm^3) = 14076.

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = 1.70E+02

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 6.07E+00

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = (6.07g)/(170.22g) = 3.57E-02

V_{SX}, (cm^3) = $(V_{ELE})(K_N)$ = 502.

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	9.97E+02	3.56E+01	1.84E-04
^{234}U	1.52E+00	5.40E-02	2.78E-07
^{235}U	1.92E+02	6.84E+00	3.49E-05
^{236}U	5.60E-01	2.00E-02	1.02E-07
^{238}U	1.21E+01	4.33E-01	2.18E-06
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.1%

Reactor: PB

$Start_{Irradiation}$: 07/11/71

$End_{Irradiation}$: 01/07/72

EFPD, (d) = 133.0

Φ_{FAST}, (nvt) = 6.00E+20

Peak $FIMA_{Fissile}$ = 11.3 %

Peak $FIMA_{Fertile}$ = 0.06 %

ρ_P, (W/cm^3) = $(P_{AVG})(P)$ = (8.3)(1.00) = 8.3

$P(t)_{SX}$, (W) = $(V_{SX})(\rho_P)$ = (502)(8.3) = 4166

Ref.: 5-6, 5-11

5.4.2 PB Fuel Test Element FTE-4

Sample Designation = PB/FTE-4

V_{ELE}, (cm^3) = 14076.

^{235}U $Wt_{ELE,EOL}$, (g/ELE) = 1.07E+02

^{235}U $Wt_{SX,EOL}$, (g/SX) = 5.10E+00

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = (5.10g)/(107.0g) = 4.77E-02

V_{SX}, (cm^3) = $(V_{ELE})(K_N)$ = 671.

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	1.07E+03	5.11E+01	1.98E-04
^{234}U	2.83E+00	1.35E-01	5.18E-07
^{235}U	1.76E+02	8.37E+00	3.20E-05
^{236}U	9.42E-01	4.50E-02	1.71E-07
^{238}U	9.14E+00	4.36E-01	1.64E-06
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.1%

Reactor: PB

$Start_{Irradiation}$: 07/11/71

$End_{Irradiation}$: 09/14/73

EFPD, (d) = 448.8

Φ_{FAST}, (nvt) = 1.98E+21

Peak $FIMA_{Fissile}$ = 36.4 %

Peak $FIMA_{Fertile}$ = 0.75 %

ρ_P, (W/cm^3) = $(P_{AVG})(P)$ = (8.3)(1.00) = 8.3

$P(t)_{SX}$, (W) = $(V_{SX})(\rho_P)$ = (671)(8.3) = 5570

Refs.: 5-7, 5-11

5.4.3 PB Fuel Test Element FTE-6

Sample Designation = PB/FTE-6

V_{ELE}, (cm^3) = 14076.

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = 1.02E+02

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 1.22E+01

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = (12.2g)/(101.84g) = 1.20E-01

V_{SX}, (cm^3) = $(V_{ELE})(K_N)$ = 1689.

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	9.02E+02	1.08E+02	1.66E-04
^{234}U	1.65E+00	1.98E-01	3.02E-07
^{235}U	2.08E+02	2.49E+01	3.78E-05
^{236}U	6.00E-01	7.20E-02	1.09E-07
^{238}U	1.30E+01	1.56E+00	2.34E-06
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.2%

Reactor: PB

$Start_{Irradiation}$: 07/11/71

$End_{Irradiation}$: 11/30/74

EFPD, (d) = 644.9

Φ_{FAST}, (nvt) = 2.90E+21

Peak $FIMA_{Fissile}$ = 45.6 %

Peak $FIMA_{Fertile}$ = 1.60 %

ρ_P, (W/cm^3) = $(P_{AVG})(P)$ = (8.3)(1.15) = 9.5

$P(t)_{SX}$, (W) = $(V_{SX})(\rho_P)$ = (1689)(9.5) = 16120

Refs.: 5-8, 5-11

5.4.4 PB Fuel Test Element FTE-14

Sample Designation = PB/FTE-14

V_{ELE}, (cm^3) = 14076.

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = 1.31E+02

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 2.87E+01

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = (28.67g)/(131.36g) = 2.18E-01

V_{SX}, (cm^3) = $(V_{ELE})(K_N)$ = 3072.

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	1.92E+03	4.20E+02	3.55E-04
^{234}U	2.88E+00	6.29E-01	5.27E-07
^{235}U	1.79E+02	3.91E+01	3.26E-05
^{236}U	9.60E-01	2.10E-01	1.74E-07
^{238}U	9.19E+00	2.01E+00	1.65E-06
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.2%

Reactor: PB

$Start_{Irradiation}$: 06/18/72

$End_{Irradiation}$: 09/14/73

EFPD, (d) = 315.8

Φ_{FAST}, (nvt) = 1.50E+21

Peak $FIMA_{Fissile}$ = 24.5 %

Peak $FIMA_{Fertile}$ = 0.4 %

ρ_P, (W/cm^3) = $(P_{AVG})(P)$ = (8.3)(1.00) = 8.3

$P(t)_{SX}$, (W) = $(V_{SX})(\rho_P)$ = (3072)(8.3) = 25500

Refs.: 5-9, 5-11

5.4.5 PB Fuel Test Element FTE-15

Sample Designation = PB/FTE-15

V_{ELE}, (cm^3) = 14076.

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = 1.04E+02

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 2.02E+01

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = (20.15g)/(104.30g) = 1.93E-01

V_{SX}, (cm^3) = $(V_{ELE})(K_N)$ = 2719.

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	1.88E+03	3.62E+02	3.48E-04
^{234}U	2.88E+00	5.57E-01	5.27E-07
^{235}U	1.79E+02	3.46E+01	3.26E-05
^{236}U	9.60E-01	1.85E-01	1.74E-07
^{238}U	9.19E+00	1.78E+00	1.65E-06
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.2%

Reactor: PB

$Start_{Irradiation}$: 07/06/72

$End_{Irradiation}$: 10/31/74

EFPD, (d) = 511.9

Φ_{FAST}, (nvt) = 2.00E+21

Peak $FIMA_{Fissile}$ = 37.7 %

Peak $FIMA_{Fertile}$ = 1.0 %

ρ_P, (W/cm^3) = $(P_{AVG})(P)$ = (8.3)(1.00) = 8.3

$P(t)_{SX}$, (W) = $(V_{SX})(\rho_P)$ = (2719)(8.3) = 22570

Refs.: 5-9, 5-11

5.4.6 PB Fuel Test Element FTE-16

Sample Designation = PB/FTE-16

V_{ELE}, (cm^3) = 14076.

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = 7.53E+01

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 2.60E+01

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = (26.00g)/(75.31g) = 3.45E-01

V_{SX}, (cm^3) = $(V_{ELE})(K_N)$ = 4860.

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	1.05E+03	3.61E+02	1.93E-04
^{234}U	2.17E+00	7.50E-01	3.97E-07
^{235}U	1.35E+02	4.67E+01	2.46E-05
^{236}U	7.20E-01	2.50E-01	1.31E-07
^{238}U	6.95E+00	2.40E+00	1.25E-06
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.2%

Reactor: PB

$Start_{Irradiation}$: 07/06/72

$End_{Irradiation}$: 10/31/74

EFPD, (d) = 511.9

Φ_{FAST}, (nvt) = 2.30E+21

Peak $FIMA_{Fissile}$ = 40.0 %

Peak $FIMA_{Fertile}$ = 1.0 %

ρ_P, (W/cm^3) = $(P_{AVG})(P)$ = (8.3)(1.00) = 8.3

$P(t)_{SX}$, (W) = $(V_{SX})\ (\rho_P)$ = (4860)(8.3) = 40300

Refs.: 5-10, 5-11

5.5 AVR IFM Fuel Pebbles

Sample Designation = AVR/1&2; (Sample is 2 ea. injection-molded spherical elements)

V_{ELE}, (cm^3) = 205.

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = 6.03E-01

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 1.21E+00

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = (1.205g)/(0.6025g) = 2.000E+00

V_{SX}, (cm^3) = $(V_{ELE})(K_N)$ = 410.

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	5.00E+00	1.00E+01	6.32E-05
^{234}U	7.00E-03	1.50E-02	9.75E-08
^{235}U	1.00E+00	2.00E+00	1.25E-05
^{236}U	3.20E-03	6.50E-03	4.02E-08
^{238}U	6.20E-02	1.25E-01	7.71E-07
$\rho_{Graphite}$, (g/cm^3)	1.30E+00	1.30E+00	6.53E-02

Initial Enrichment = 93.2%

Reactor: AVR

$Start_{Irradiation}$: 12/68

$End_{Irradiation}$: 06/73

EFPD, (d) = 985.0

Φ_{FAST}, (nvt) = 3.00E+21

Peak $FIMA_{Fissile}$ = 9.0 %

Peak $FIMA_{Fertile}$ = NR

ρ_P, (W/cm^3) = $(P_{AVG})(P)$ = (2.4)(1.00) = 2.4

$P(t)_{SX}$, (W) = $(V_{SX})(\rho_P)$ = (410)(2.4) = 984

Refs.: 5-12, 5-13

5.6 OTHER HTGR IFM

5.6.1 Fuel Test Capsule P13P

Sample Designation = P13P

V_{ELE}, (cm^3) = NR

^{235}U $Wt_{ELE,EOL}$, (g/ELE) = NR

^{235}U $Wt_{SX,EOL}$, (g/SX) = 1.43E-01

Normalization Factor, K_N = ($^{235}U_{SX}$) ÷ ($^{235}U_{ELE}$) = NR

V_{SX}, (cm^3) = 205

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	3.34E+01	NR
^{234}U	NR	7.80E-03	NR
^{235}U	NR	1.00E+00	NR
^{236}U	NR	3.20E-03	NR
^{238}U	NR	6.25E-02	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 93.2%

Reactor: ETR

$Start_{Irradiation}$: 04/15/72

$End_{Irradiation}$: 04/30/73

EFPD, (d) = 181.0

Φ_{FAST}, (nvt) = 8.10E+21

Peak $FIMA_{Fissile}$ = NR

Peak $FIMA_{Fertile}$ = NR

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 7760

Ref.: 5-14

5.6.2 Fuel Test Capsule P13Q

Sample Designation = P13Q

V_{ELE}, (cm^3) = NR

^{235}U $Wt_{ELE,EOL}$, (g/ELE) = NR

^{235}U $Wt_{SX,EOL}$, (g/SX) = 1.66E-01

Normalization Factor, K_N = ($^{235}U_{SX}$) ÷ ($^{235}U_{ELE}$) = NR

V_{SX}, (cm^3) = 394

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	4.60E+01	NR
^{234}U	NR	1.50E-02	NR
^{235}U	NR	1.92E+00	NR
^{236}U	NR	6.20E-03	NR
^{238}U	NR	1.20E-01	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 93.1%

Reactor: ORR

$Start_{Irradiation}$: 08/22/73

$End_{Irradiation}$: 02/23/75

EFPD, (d) = 391.7

Φ_{FAST}, (nvt) = 9.60E+21

Peak $FIMA_{Fissile}$ = 76.5 %

Peak $FIMA_{Fertile}$ = 5.0 %

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 7460

Refs.: 5-15, 5-16

5.6.3 Fuel Test Capsules P13R & P13S

Sample Designation	=	P13R/S
V_{ELE}, (cm^3)	=	NR
^{235}U $Wt_{ELE,EOL}$, (g/ELE)	=	NR
^{235}U $Wt_{SX,EOL}$, (g/SX)	=	7.60E-02
Normalization Factor, K_N	=	$(^{235}U_{SX}) \div (^{235}U_{ELE})$ = NR
V_{SX}, (cm^3)	=	199

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	1.86E+01	NR
^{234}U	NR	7.50E-03	NR
^{235}U	NR	9.69E-01	NR
^{236}U	NR	3.10E-03	NR
^{238}U	NR	6.06E-02	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 93.2%

Reactor:		GETR
$Start_{Irradiation}$:		12/10/73
$End_{Irradiation}$:		10/31/74
EFPD, (d)	=	258.0
Φ_{FAST}, (nvt)	=	1.25E+22
Peak $FIMA_{Fissile}$	=	75.0 %
Peak $FIMA_{Fertile}$	=	5.0 %
ρ_P, (W/cm^3)	=	NR
$P(t)_{SX}$, (W)	=	5000
Refs.:		5-17, 5-18

5.6.4 Fuel Test Capsule P13T

Sample Designation = P13T

V_{ELE}, (cm^3) = NR

^{235}U $Wt_{ELE,EOL}$, (g/ELE) = NR

^{235}U $Wt_{SX,EOL}$, (g/SX) = 2.28E-01

Normalization Factor, K_N = ($^{235}U_{SX}$) ÷ ($^{235}U_{ELE}$) = NR

V_{SX}, (cm^3) = 1144

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	7.49E+01	NR
^{234}U	NR	4.40E-02	NR
^{235}U	NR	5.57E+00	NR
^{236}U	NR	1.80E-02	NR
^{238}U	NR	3.40E-01	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 93.3%

Reactor: ORR

$Start_{Irradiation}$: 05/10/75

$End_{Irradiation}$: 07/05/76

EFPD, (d) = 363.4

Φ_{FAST}, (nvt) = 8.0E+21

Peak $FIMA_{Fissile}$ = 75.0 %

Peak $FIMA_{Fertile}$ = 4.0 %

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 21600

Refs.: 5-19, 5-20

5.6.5 Fuel Test Capsule P13V

Sample Designation = P13V

V_{ELE}, (cm^3) = NR

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = NR

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 1.79E-01

Normalization Factor, K_N = ($^{235}U_{SX}$) ÷ ($^{235}U_{ELE}$) = NR

V_{SX}, (cm^3) = 411

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	3.84E+01	NR
^{234}U	NR	1.60E-02	NR
^{235}U	NR	2.00E+00	NR
^{236}U	NR	6.50E-03	NR
^{238}U	NR	1.25E-01	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 93.1%

Reactor: GETR

$Start_{Irradiation}$: 03/01/76

$End_{Irradiation}$: 11/11/76

EFPD, (d) = 205.0

Φ_{FAST}, (nvt) = 9.00E+21

Peak $FIMA_{Fissile}$ = 75.0 %

Peak $FIMA_{Fertile}$ = 5.0 %

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 7770

Refs.: 5-21, 5-22

5.6.6 Fuel Test Capsule HB-2

Sample Designation = HB-2

V_{ELE}, (cm^3) = NR

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = NR

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 7.40E-02

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = NR

V_{SX}, (cm^3) = 51.4

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	2.47E+00	NR
^{234}U	NR	1.95E-03	NR
^{235}U	NR	2.50E-01	NR
^{236}U	NR	8.11E-04	NR
^{238}U	NR	1.57E-02	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 93.1%

Reactor: GETR

$Start_{Irradiation}$: 09/07/75

$End_{Irradiation}$: 01/08/76

EFPD, (d) = 123.0

Φ_{FAST}, (nvt) = 5.00E+21

Peak $FIMA_{Fissile}$ = 58.0 %

Peak $FIMA_{Fertile}$ = 1.0 %

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 1300

Ref.: 5-23

5.6.7 Fuel Test Capsules HRB-14 & HRB-15A

Sample Designation = HRB-14/15A

V_{ELE}, (cm^3) = NR

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = NR

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 7.40E-02

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = NR

V_{SX}, (cm^3) = 117.2

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	1.28E+01	NR
^{234}U	NR	0.00E+00	NR
^{235}U	NR	5.70E-01	NR
^{236}U	NR	0.00E+00	NR
^{238}U	NR	2.23E+00	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 20.4%

Reactor: HFIR

$Start_{Irradiation}$: 07/26/80

$End_{Irradiation}$: 01/29/81

EFPD, (d) = 173.8

Φ_{FAST}, (nvt) = 6.50E+21

Peak $FIMA_{Fissile}$ = 29.0 %

Peak $FIMA_{Fertile}$ = 6.4 %

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 5050

Refs.: 5-24, 5-25

5.6.8 Fuel Test Capsule GF-3

Sample Designation = GF-3

V_{ELE}, (cm^3) = NR

^{235}U $Wt_{ELE,EOL}$, (g/ELE) = NR

^{235}U $Wt_{SX,EOL}$, (g/SX) = 4.20E-02

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = NR

V_{SX}, (cm^3) = 46.0

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	8.40E+00	NR
^{234}U	NR	1.70E-03	NR
^{235}U	NR	2.24E-01	NR
^{236}U	NR	7.00E-04	NR
^{238}U	NR	1.40E-02	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 93.2%

Reactor: SILOE

$Start_{Irradiation}$: 01/31/74

$End_{Irradiation}$: 07/25/75

EFPD, (d) = 363.0

Φ_{FAST}, (nvt) = 1.00E+21

Peak $FIMA_{Fissile}$ = 11.4 %

Peak $FIMA_{Fertile}$ = 3.6 %

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 870

Ref.: 5-26

5.6.9 Fuel Test Capsule DR-GB2

Sample Designation = DR-GB2

V_{ELE}, (cm^3) = NR

^{235}U $Wt_{ELE,EOL}$, (g/ELE) = NR

^{235}U $Wt_{SX,EOL}$, (g/SX) = 9.75E-01

Normalization Factor, K_N = ($^{235}U_{SX}$) ÷ ($^{235}U_{ELE}$) = NR

V_{SX}, (cm^3) = 258.0

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	1.24E+01	NR
^{234}U	NR	9.80E-03	NR
^{235}U	NR	1.26E+00	NR
^{236}U	NR	4.10E-03	NR
^{238}U	NR	7.87E-02	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 93.2%

Reactor: DRAGON

$Start_{Irradiation}$: 03/75

$End_{Irradiation}$: 12/75

EFPD, (d) = 140.0

Φ_{FAST}, (nvt) = NR

Peak $FIMA_{Fissile}$ = NR

Peak $FIMA_{Fertile}$ = NR

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 1627

Refs.: 5-27, 5-28

5.7 RERTR IFM

Sample Designation = RERTR

V_{ELE}, (cm^3) = NR

$^{235}U\ Wt_{ELE,EOL}$, (g/ELE) = NR

$^{235}U\ Wt_{SX,EOL}$, (g/SX) = 3.52E+02

Normalization Factor, K_N = $(^{235}U_{SX}) \div (^{235}U_{ELE})$ = NR

V_{SX}, (cm^3) = NR

Fuel Constituent	$Wt_{ELE,BOL}$ (g)	$Wt_{SX,BOL}$ (g)	ρ_A (atom/b-cm)
$^{232}Th_{Fissile}$	NR	NR	NR
$^{232}Th_{Fertile}$	NR	NR	NR
$^{232}Th_{Total}$	NR	0.00E+00	NR
^{234}U	NR	7.32E+00	NR
^{235}U	NR	7.62E+02	NR
^{236}U	NR	3.27E+00	NR
^{238}U	NR	3.08E+03	NR
$\rho_{Graphite}$, (g/cm^3)	NR	NR	NR

Initial Enrichment = 19.7%

Reactor: ORR

$Start_{Irradiation}$: 12/79

$End_{Irradiation}$: 08/84

EFPD, (d) = 920.0

Φ_{FAST}, (nvt) = NR

Peak $FIMA_{Fissile}$ = NR

Peak $FIMA_{Fertile}$ = NR

ρ_P, (W/cm^3) = NR

$P(t)_{SX}$, (W) = 390000

Ref.: 5-29

6.0 RESULTS

Radiological characterization results for the subject Irradiated Fuel Material (IFM) entities listed in Section 5.0 are presented herein. Separate tables are provided for each subgroup designation in the same order as that used in the foregoing Section 5.0.

The common format used to report information in this section conforms to the following Data Sheet template:

6.X IFM Group Designation

6.X.X IFM SubGroup Designation

Table 6.xx Fuel Test Capsule/Element Identity

Sample Designation	=	Identity of Archival IFM Entity
Input File	=	Computer Filename of Code Input Data
Output File	=	Computer Filename of Code Output Data
Reference Decay Date (RDD)	=	Applicable Reference Decay Date for Reported Activity Results
Total Decay Heat, (W)	=	Radioactive Decay Power in IFM Entity (normalized to RDD)

Irradiated Fuel Materials Nuclide Activities & Mass Quantities, as of Reference Decay Date

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
Constituent Radioisotope contained in IFM Entity	Decay Constant of Radioisotope (in yr^{-1})[3]	Yearly Fractional Decay Factor for Radioisotope[3]	Specific Activity of Radioisotope (in Curies/gram)[3]	Calculated Activity of Radioisotope (in Curies) (corrected to indicated RDD)	Calculated Equiv. Mass of Radioisotope (in grams) (corrected to indicated RDD)[4]

At the end of the tables for each generic group of fuel, i.e. HTGR and RERTR, summary tables are developed to summarize the radiological characteristics of each of the FHUs.

Note because of the generation of ^{90}Y from the decay of ^{90}Sr the decay/year for ^{90}Y is the net effective decay.

---

[3] Applicable radioactive decay parameters, listed above and utilized herein for characterization calculations, are drawn from published reference data, viz:

- Nuclide Decay Constants are based on half-life data published in Ref. 6-1;
- Nuclide Specific Activities are taken from data published in Ref. 6-2.

[4] There were no reportable amounts of ^{232}U in the calculations. Therefore it is assumed that each of the Fuel Type Items are classified as Detail Material Type is #71 as defined in DOE M 474.1-2 with <5ppm ^{232}U (Ref. 6-4).

6.1 HTGR IFM

6.1.1 FSV IFM

Table 6.1. Results for FSV Fuel Test Element FTE-2

Sample Designation:	FSV /FTE-2
Computer Code Input File:	FSVFTE2.I
Computer Code Output File:	FSVFTE2.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	3.10E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	5.36E-03	5.53E-07
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	1.98E-01	4.95E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	2.26E+00	1.51E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	2.26E+00	9.04E-06
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	1.09E-02	7.79E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	5.12E-02	4.27E-05
137Cs	2.30E-02	9.7726E-01	9.80E+01	2.28E+00	2.33E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	2.75E-01	2.93E-04
151Sm	7.50E-03	9.9253E-01	2.60E+01	1.08E-02	4.15E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	3.40E-02	2.27E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	5.22E-03	3.73E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	3.34E-06	3.04E+01
233U	4.36E-06	1.0000E+00	9.50E-03	4.75E-03	5.00E-01
234U	2.82E-06	1.0000E+00	6.20E-03	3.22E-04	5.19E-02
235U	9.84E-10	1.0000E+00	2.10E-06	1.20E-06	5.71E-01
236U	2.96E-08	1.0000E+00	6.30E-05	1.03E-05	1.63E-01
238U	1.55E-10	1.0000E+00	3.30E-07	2.64E-08	8.00E-02
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	2.55E-02	1.50E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	8.06E-05	1.30E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	1.38E-04	6.00E-04
241Pu	4.81E-02	9.5304E-01	1.10E+02	2.97E-02	2.70E-04
242Pu	1.85E-06	1.0000E+00	3.90E-03	5.46E-07	1.40E-04
TOTALS:				7.45E+00	3.18E+01

Table 6.2. Results for FSV Surveillance Element

Sample Designation:	FSV/SURV
Computer Code Input File:	FSVSUR.I
Computer Code Output File:	FSVSUR.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	1.10E-01

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass
3H	5.64E-02	9.4516E-01	9.70E+03	1.88E-02	1.94E-06
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	5.89E-01	1.47E-03
90Sr	2.38E-02	9.7648E-01	1.50E+02	8.58E+00	5.72E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	8.58E+00	3.43E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	1.27E-02	9.07E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	2.20E-02	1.83E-05
137Cs	2.30E-02	9.7726E-01	9.80E+01	8.73E+00	8.91E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	4.60E-01	4.89E-04
151Sm	7.50E-03	9.9253E-01	2.60E+01	7.49E-02	2.88E-03
154Eu	8.07E-02	9.2247E-01	1.50E+02	5.70E-02	3.80E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	8.20E-03	5.86E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	3.59E-05	3.26E+02
233U	4.36E-06	1.0000E+00	9.50E-03	3.10E-02	3.26E+00
234U	2.82E-06	1.0000E+00	6.20E-03	1.49E-03	2.40E-01
235U	9.84E-10	1.0000E+00	2.10E-06	1.44E-05	6.88E+00
236U	2.96E-08	1.0000E+00	6.30E-05	5.10E-05	8.10E-01
238U	1.55E-10	1.0000E+00	3.30E-07	2.24E-07	6.80E-01
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	2.72E-02	1.60E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	5.52E-04	8.90E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	6.44E-04	2.80E-03
241Pu	4.81E-02	9.5304E-01	1.10E+02	6.82E-02	6.20E-04
242Pu	1.85E-06	1.0000E+00	3.90E-03	6.63E-07	1.70E-04
TOTALS:				2.73E+01	3.38E+02

6.1.2 PB FTE IFM

Table 6.3. Results for PB Fuel Test Element FTE-3

Sample Designation:	PB/FTE-3
Computer Code Input File:	PBFTE3.I
Computer Code Output File:	PBFTE3.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	1.27E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	1.79E-03	1.85E-07
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	4.92E-02	1.23E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	1.01E+00	6.73E-03
90Y	9.48E+01	9.7648E-01	2.50E+05	1.01E+00	4.04E-06
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	0.00E+00	0.00E+00
134Cs	3.35E-01	7.1534E-01	1.20E+03	0.00E+00	0.00E+00
137Cs	2.30E-02	9.7726E-01	9.80E+01	1.05E+00	1.07E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	1.18E-02	1.26E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	2.46E-02	9.46E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	1.70E-03	1.13E-05
155Eu	1.47E-01	8.6329E-01	1.40E+03	7.45E-04	5.32E-07
232Th	4.95E-11	1.0000E+00	1.10E-07	3.89E-06	3.54E+01
233U	4.36E-06	1.0000E+00	9.50E-03	1.93E-03	2.03E-01
234U	2.82E-06	1.0000E+00	6.20E-03	3.41E-04	5.50E-02
235U	9.84E-10	1.0000E+00	2.10E-06	1.28E-05	6.11E+00
236U	2.96E-08	1.0000E+00	6.30E-05	1.15E-05	1.83E-01
238U	1.55E-10	1.0000E+00	3.30E-07	1.40E-07	4.25E-01
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	1.10E-03	6.50E-05
239Pu	2.88E-05	9.9997E-01	6.20E-02	3.66E-04	5.90E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	1.01E-04	4.40E-04
241Pu	4.81E-02	9.5304E-01	1.10E+02	3.74E-03	3.40E-05
242Pu	1.85E-06	1.0000E+00	3.90E-03	8.97E-09	2.30E-06
TOTALS:				3.17E+00	4.24E+01

Table 6.4. Results for PB Fuel Test Element FTE-4

Sample Designation:	PB/FTE-4
Computer Code Input File:	PBFTE4.I
Computer Code Output File:	PBFTE4.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	6.17E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	8.93E-03	9.21E-07
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	2.55E-01	6.38E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	4.73E+00	3.15E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	4.73E+00	1.89E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	1.82E-03	1.30E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	3.58E-03	2.98E-06
137Cs	2.30E-02	9.7726E-01	9.80E+01	4.88E+00	4.98E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	5.66E-02	6.02E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	5.62E-02	2.16E-03
154Eu	8.07E-02	9.2247E-01	1.50E+02	3.74E-02	2.49E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	3.73E-03	2.66E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	5.50E-06	5.00E+01
233U	4.36E-06	1.0000E+00	9.50E-03	8.03E-03	8.45E-01
234U	2.82E-06	1.0000E+00	6.20E-03	9.49E-04	1.53E-01
235U	9.84E-10	1.0000E+00	2.10E-06	1.12E-05	5.34E+00
236U	2.96E-08	1.0000E+00	6.30E-05	4.23E-05	6.71E-01
238U	1.55E-10	1.0000E+00	3.30E-07	1.36E-07	4.11E-01
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	5.61E-02	3.30E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	6.82E-04	1.10E-02
240Pu	1.06E-04	9.9989E-01	2.30E-01	6.21E-04	2.70E-03
241Pu	4.81E-02	9.5304E-01	1.10E+02	8.14E-02	7.40E-04
242Pu	1.85E-06	1.0000E+00	3.90E-03	8.97E-07	2.30E-04
TOTALS:				1.49E+01	5.75E+01

Table 6.5. Results for PB Fuel Test Element FTE-6

Sample Designation:	PB/FTE-6
Computer Code Input File:	PBFTE6.I
Computer Code Output File:	PBFTE6.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	2.73E-01

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	3.93E-02	4.05E-06
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	1.12E+00	2.80E-03
90Sr	2.38E-02	9.7648E-01	1.50E+02	2.00E+01	1.33E-01
90Y	9.48E+01	9.7648E-01	2.50E+05	2.00E+01	8.00E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	9.68E-03	6.91E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	3.58E-02	2.98E-05
137Cs	2.30E-02	9.7726E-01	9.80E+01	2.06E+01	2.10E-01
147Pm	2.64E-01	7.6797E-01	9.40E+02	2.48E-01	2.64E-04
151Sm	7.50E-03	9.9253E-01	2.60E+01	1.77E-01	6.81E-03
154Eu	8.07E-02	9.2247E-01	1.50E+02	2.55E-01	1.70E-03
155Eu	1.47E-01	8.6329E-01	1.40E+03	2.24E-02	1.60E-05
232Th	4.95E-11	1.0000E+00	1.10E-07	1.14E-05	1.04E+02
233U	4.36E-06	1.0000E+00	9.50E-03	2.47E-02	2.60E+00
234U	2.82E-06	1.0000E+00	6.20E-03	2.13E-03	3.44E-01
235U	9.84E-10	1.0000E+00	2.10E-06	2.65E-05	1.26E+01
236U	2.96E-08	1.0000E+00	6.30E-05	1.63E-04	2.59E+00
238U	1.55E-10	1.0000E+00	3.30E-07	4.69E-07	1.42E+00
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	5.10E-01	3.00E-02
239Pu	2.88E-05	9.9997E-01	6.20E-02	2.98E-03	4.80E-02
240Pu	1.06E-04	9.9989E-01	2.30E-01	2.99E-03	1.30E-02
241Pu	4.81E-02	9.5304E-01	1.10E+02	6.16E-01	5.60E-03
242Pu	1.85E-06	1.0000E+00	3.90E-03	1.05E-05	2.70E-03
TOTALS:				6.37E+01	1.24E+02

Table 6.6. Results for PB Fuel Test Element FTE-14

Sample Designation: PB/FTE-14
Computer Code Input File: PBFTE14.I
Computer Code Output File: PBFTE14.O
Reference Decay Date: 1 January 96
Contained Decay Heat, (W): 2.00E-01

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	2.95E-02	3.04E-06
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	8.26E-01	2.07E-03
90Sr	2.38E-02	9.7648E-01	1.50E+02	1.54E+01	1.03E-01
90Y	9.48E+01	9.7648E-01	2.50E+05	1.54E+01	6.16E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	5.45E-03	3.89E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	8.70E-03	7.25E-06
137Cs	2.30E-02	9.7726E-01	9.80E+01	1.59E+01	1.62E-01
147Pm	2.64E-01	7.6797E-01	9.40E+02	2.12E-01	2.26E-04
151Sm	7.50E-03	9.9253E-01	2.60E+01	2.36E-01	9.08E-03
154Eu	8.07E-02	9.2247E-01	1.50E+02	8.17E-02	5.45E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	1.12E-02	8.00E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	4.57E-05	4.15E+02
233U	4.36E-06	1.0000E+00	9.50E-03	4.31E-02	4.54E+00
234U	2.82E-06	1.0000E+00	6.20E-03	4.46E-03	7.20E-01
235U	9.84E-10	1.0000E+00	2.10E-06	6.13E-05	2.92E+01
236U	2.96E-08	1.0000E+00	6.30E-05	1.46E-04	2.31E+00
238U	1.55E-10	1.0000E+00	3.30E-07	6.37E-07	1.93E+00
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	8.84E-02	5.20E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	2.91E-03	4.70E-02
240Pu	1.06E-04	9.9989E-01	2.30E-01	1.79E-03	7.80E-03
241Pu	4.81E-02	9.5304E-01	1.10E+02	1.65E-01	1.50E-03
242Pu	1.85E-06	1.0000E+00	3.90E-03	1.09E-06	2.80E-04
TOTALS:				4.84E+01	4.54E+02

Table 6.7. Results for PB Fuel Test Element FTE-15

Sample Designation:	PB/FTE-15
Computer Code Input File:	PBFTE15.I
Computer Code Output File:	PBFTE15.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	3.00E-01

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	4.47E-02	4.61E-06
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	1.29E+00	3.23E-03
90Sr	2.38E-02	9.7648E-01	1.50E+02	2.26E+01	1.51E-01
90Y	9.48E+01	9.7648E-01	2.50E+05	2.26E+01	9.04E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	1.21E-02	8.64E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	2.76E-02	2.30E-05
137Cs	2.30E-02	9.7726E-01	9.80E+01	2.32E+01	2.37E-01
147Pm	2.64E-01	7.6797E-01	9.40E+02	3.30E-01	3.51E-04
151Sm	7.50E-03	9.9253E-01	2.60E+01	2.56E-01	9.85E-03
154Eu	8.07E-02	9.2247E-01	1.50E+02	2.04E-01	1.36E-03
155Eu	1.47E-01	8.6329E-01	1.40E+03	2.09E-02	1.49E-05
232Th	4.95E-11	1.0000E+00	1.10E-07	3.91E-05	3.55E+02
233U	4.36E-06	1.0000E+00	9.50E-03	5.46E-02	5.75E+00
234U	2.82E-06	1.0000E+00	6.20E-03	4.72E-03	7.62E-01
235U	9.84E-10	1.0000E+00	2.10E-06	4.47E-05	2.13E+01
236U	2.96E-08	1.0000E+00	6.30E-05	1.85E-04	2.93E+00
238U	1.55E-10	1.0000E+00	3.30E-07	5.48E-07	1.66E+00
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	3.06E-01	1.80E-02
239Pu	2.88E-05	9.9997E-01	6.20E-02	3.04E-03	4.90E-02
240Pu	1.06E-04	9.9989E-01	2.30E-01	2.76E-03	1.20E-02
241Pu	4.81E-02	9.5304E-01	1.10E+02	4.07E-01	3.70E-03
242Pu	1.85E-06	1.0000E+00	3.90E-03	4.68E-06	1.20E-03
TOTALS:				7.14E+01	3.88E+02

Table 6.8. Results for PB Fuel Test Element FTE-16

Sample Designation:	PB/FTE-16
Computer Code Input File:	PBFTE16.I
Computer Code Output File:	PBFTE16.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	5.33E-01

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	8.04E-02	8.29E-06
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	2.28E+00	5.70E-03
90Sr	2.38E-02	9.7648E-01	1.50E+02	4.01E+01	2.67E-01
90Y	9.48E+01	9.7648E-01	2.50E+05	4.01E+01	1.60E-04
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	1.82E-02	1.30E-05
134Cs	3.35E-01	7.1534E-01	1.20E+03	5.63E-02	4.69E-05
137Cs	2.30E-02	9.7726E-01	9.80E+01	4.13E+01	4.21E-01
147Pm	2.64E-01	7.6797E-01	9.40E+02	5.54E-01	5.89E-04
151Sm	7.50E-03	9.9253E-01	2.60E+01	3.15E-01	1.21E-02
154Eu	8.07E-02	9.2247E-01	1.50E+02	4.08E-01	2.72E-03
155Eu	1.47E-01	8.6329E-01	1.40E+03	3.73E-02	2.66E-05
232Th	4.95E-11	1.0000E+00	1.10E-07	3.86E-05	3.51E+02
233U	4.36E-06	1.0000E+00	9.50E-03	6.52E-02	6.86E+00
234U	2.82E-06	1.0000E+00	6.20E-03	6.57E-03	1.06E+00
235U	9.84E-10	1.0000E+00	2.10E-06	4.89E-05	2.33E+01
236U	2.96E-08	1.0000E+00	6.30E-05	3.00E-04	4.76E+00
238U	1.55E-10	1.0000E+00	3.30E-07	7.36E-07	2.23E+00
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	6.80E-01	4.00E-02
239Pu	2.88E-05	9.9997E-01	6.20E-02	3.35E-03	5.40E-02
240Pu	1.06E-04	9.9989E-01	2.30E-01	4.14E-03	1.80E-02
241Pu	4.81E-02	9.5304E-01	1.10E+02	6.71E-01	6.10E-03
242Pu	1.85E-06	1.0000E+00	3.90E-03	1.29E-05	3.30E-03
TOTALS:				1.27E+02	3.90E+02

6.1.3 AVR IFM Pebbles

Table 6.9. Results for AVR IFM Pebbles

Sample Designation:	AVR/1&2
Computer Code Input File:	AVR.I
Computer Code Output File:	AVR.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	2.33E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	3.57E-03	3.68E-07
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	9.67E-02	2.42E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	1.80E+00	1.20E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	1.80E+00	7.20E-06
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	4.06E-04	2.90E-07
134Cs	3.35E-01	7.1534E-01	1.20E+03	1.02E-03	8.50E-07
137Cs	2.30E-02	9.7726E-01	9.80E+01	1.86E+00	1.90E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	1.77E-02	1.88E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	1.08E-02	4.15E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	1.36E-02	9.07E-05
155Eu	1.47E-01	8.6329E-01	1.40E+03	9.69E-04	6.92E-07
232Th	4.95E-11	1.0000E+00	1.10E-07	1.07E-06	9.73E+00
233U	4.36E-06	1.0000E+00	9.50E-03	1.55E-03	1.63E-01
234U	2.82E-06	1.0000E+00	6.20E-03	1.36E-04	2.20E-02
235U	9.84E-10	1.0000E+00	2.10E-06	1.79E-06	8.54E-01
236U	2.96E-08	1.0000E+00	6.30E-05	1.30E-05	2.07E-01
238U	1.55E-10	1.0000E+00	3.30E-07	3.93E-08	1.19E-01
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	1.87E-02	1.10E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	8.68E-05	1.40E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	1.84E-04	8.00E-04
241Pu	4.81E-02	9.5304E-01	1.10E+02	1.87E-02	1.70E-04
242Pu	1.85E-06	1.0000E+00	3.90E-03	5.07E-07	1.30E-04
TOTALS:				5.64E+00	1.11E+01

6.1.4 OTHER HTGR IFM

Table 6.10. Results for Fuel Test Capsule P13P

Sample Designation:	P13P
Computer Code Input File:	P13P.I
Computer Code Output File:	P13P.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	4.00E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	5.36E-03	5.53E-07
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	1.85E-01	4.62E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	2.83E+00	1.89E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	2.83E+00	1.13E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	1.82E-03	1.30E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	4.09E-03	3.41E-06
137Cs	2.30E-02	9.7726E-01	9.80E+01	2.85E+00	2.91E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	1.77E-02	1.88E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	1.18E-02	4.54E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	2.98E-02	1.99E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	2.24E-03	1.60E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	3.47E-06	3.15E+01
233U	4.36E-06	1.0000E+00	9.50E-03	8.55E-03	9.00E-01
234U	2.82E-06	1.0000E+00	6.20E-03	1.44E-03	2.32E-01
235U	9.84E-10	1.0000E+00	2.10E-06	3.00E-07	1.43E-01
236U	2.96E-08	1.0000E+00	6.30E-05	9.32E-06	1.48E-01
238U	1.55E-10	1.0000E+00	3.30E-07	1.78E-08	5.40E-02
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	8.50E-02	5.00E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	9.30E-05	1.50E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	1.56E-04	6.80E-04
241Pu	4.81E-02	9.5304E-01	1.10E+02	2.31E-02	2.10E-04
242Pu	1.85E-06	1.0000E+00	3.90E-03	1.76E-06	4.50E-04
TOTALS:				8.89E+00	3.30E+01

Table 6.11. Results for Fuel Test Capsule P13Q

Sample Designation:	P13Q
Computer Code Input File:	P13Q.I
Computer Code Output File:	P13Q.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	8.30E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	1.07E-02	1.10E-06
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	3.87E-01	9.68E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	5.91E+00	3.94E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	5.91E+00	2.36E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	4.24E-03	3.03E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	1.18E-02	9.83E-06
137Cs	2.30E-02	9.7726E-01	9.80E+01	5.95E+00	6.07E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	3.66E-02	3.89E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	1.87E-02	7.19E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	6.38E-02	4.25E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	5.22E-03	3.73E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	4.74E-06	4.31E+01
233U	4.36E-06	1.0000E+00	9.50E-03	1.03E-02	1.08E+00
234U	2.82E-06	1.0000E+00	6.20E-03	1.72E-03	2.78E-01
235U	9.84E-10	1.0000E+00	2.10E-06	3.49E-07	1.66E-01
236U	2.96E-08	1.0000E+00	6.30E-05	1.81E-05	2.88E-01
238U	1.55E-10	1.0000E+00	3.30E-07	3.40E-08	1.03E-01
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	2.04E-01	1.20E-02
239Pu	2.88E-05	9.9997E-01	6.20E-02	1.74E-04	2.80E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	3.45E-04	1.50E-03
241Pu	4.81E-02	9.5304E-01	1.10E+02	4.62E-02	4.20E-04
242Pu	1.85E-06	1.0000E+00	3.90E-03	4.29E-06	1.10E-03
TOTALS:				1.86E+01	4.51E+01

Table 6.12. Results for Fuel Test Capsules P13R & P13S

Sample Designation:	P13R/S
Computer Code Input File:	P13RS.I
Computer Code Output File:	P13RS.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	3.62E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	4.47E-03	4.61E-07
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	1.76E-01	4.40E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	2.60E+00	1.73E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	2.60E+00	1.04E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	2.42E-03	1.73E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	6.14E-03	5.12E-06
137Cs	2.30E-02	9.7726E-01	9.80E+01	2.64E+00	2.69E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	2.12E-02	2.26E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	7.88E-03	3.03E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	2.81E-02	1.87E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	2.24E-03	1.60E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	1.93E-06	1.75E+01
233U	4.36E-06	1.0000E+00	9.50E-03	4.42E-03	4.65E-01
234U	2.82E-06	1.0000E+00	6.20E-03	7.87E-04	1.27E-01
235U	9.84E-10	1.0000E+00	2.10E-06	1.59E-07	7.56E-02
236U	2.96E-08	1.0000E+00	6.30E-05	9.26E-06	1.47E-01
238U	1.55E-10	1.0000E+00	3.30E-07	1.75E-08	5.29E-02
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	8.50E-02	5.00E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	7.44E-05	1.20E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	1.68E-04	7.30E-04
241Pu	4.81E-02	9.5304E-01	1.10E+02	1.98E-02	1.80E-04
242Pu	1.85E-06	1.0000E+00	3.90E-03	2.11E-06	5.40E-04
TOTALS:				8.20E+00	1.84E+01

Table 6.13. Results for Fuel Test Capsule P13T

Sample Designation:	P13T
Computer Code Input File:	P13T.I
Computer Code Output File:	P13T.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	2.27E-01

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	3.22E-02	3.32E-06
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	1.13E+00	2.82E-03
90Sr	2.38E-02	9.7648E-01	1.50E+02	1.62E+01	1.08E-01
90Y	9.48E+01	9.7648E-01	2.50E+05	1.62E+01	6.48E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	1.94E-02	1.39E-05
134Cs	3.35E-01	7.1534E-01	1.20E+03	7.16E-02	5.97E-05
137Cs	2.30E-02	9.7726E-01	9.80E+01	1.64E+01	1.67E-01
147Pm	2.64E-01	7.6797E-01	9.40E+02	1.59E-01	1.69E-04
151Sm	7.50E-03	9.9253E-01	2.60E+01	3.94E-02	1.52E-03
154Eu	8.07E-02	9.2247E-01	1.50E+02	1.79E-01	1.19E-03
155Eu	1.47E-01	8.6329E-01	1.40E+03	1.57E-02	1.12E-05
232Th	4.95E-11	1.0000E+00	1.10E-07	7.59E-06	6.90E+01
233U	4.36E-06	1.0000E+00	9.50E-03	1.77E-02	1.86E+00
234U	2.82E-06	1.0000E+00	6.20E-03	3.67E-03	5.92E-01
235U	9.84E-10	1.0000E+00	2.10E-06	4.79E-07	2.28E-01
236U	2.96E-08	1.0000E+00	6.30E-05	5.36E-05	8.50E-01
238U	1.55E-10	1.0000E+00	3.30E-07	9.90E-08	3.00E-01
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	5.95E-01	3.50E-02
239Pu	2.88E-05	9.9997E-01	6.20E-02	4.53E-04	7.30E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	1.22E-03	5.30E-03
241Pu	4.81E-02	9.5304E-01	1.10E+02	1.32E-01	1.20E-03
242Pu	1.85E-06	1.0000E+00	3.90E-03	1.72E-05	4.40E-03
TOTALS:				5.12E+01	7.32E+01

Table 6.14. Results for Fuel Test Capsule P13V

Sample Designation:	P13V
Computer Code Input File:	P13V.I
Computer Code Output File:	P13V.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	7.20E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	1.07E-02	1.10E-06
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	3.69E-01	9.23E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	5.20E+00	3.47E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	5.20E+00	2.08E-05
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	6.66E-03	4.76E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	2.25E-02	1.88E-05
137Cs	2.30E-02	9.7726E-01	9.80E+01	5.25E+00	5.36E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	7.67E-02	8.16E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	1.58E-02	6.08E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	6.21E-02	4.14E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	5.96E-03	4.26E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	3.97E-06	3.61E+01
233U	4.36E-06	1.0000E+00	9.50E-03	9.30E-03	9.79E-01
234U	2.82E-06	1.0000E+00	6.20E-03	1.59E-03	2.56E-01
235U	9.84E-10	1.0000E+00	2.10E-06	3.76E-07	1.79E-01
236U	2.96E-08	1.0000E+00	6.30E-05	1.90E-05	3.02E-01
238U	1.55E-10	1.0000E+00	3.30E-07	3.63E-08	1.10E-01
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	1.53E-01	9.00E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	1.49E-04	2.40E-03
240Pu	1.06E-04	9.9989E-01	2.30E-01	3.22E-04	1.40E-03
241Pu	4.81E-02	9.5304E-01	1.10E+02	4.18E-02	3.80E-04
242Pu	1.85E-06	1.0000E+00	3.90E-03	3.78E-06	9.70E-04
TOTALS:				1.64E+01	3.80E+01

Table 6.15. Results for Fuel Test Capsule HB-2

Sample Designation:	HB-2
Computer Code Input File:	HB2.I
Computer Code Output File:	HB2.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	4.30E-03

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	8.93E-04	9.21E-08
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	2.02E-02	5.05E-05
90Sr	2.38E-02	9.7648E-01	1.50E+02	3.24E-01	2.16E-03
90Y	9.48E+01	9.7648E-01	2.50E+05	3.24E-01	1.30E-06
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	3.03E-04	2.16E-07
134Cs	3.35E-01	7.1534E-01	1.20E+03	8.70E-04	7.25E-07
137Cs	2.30E-02	9.7726E-01	9.80E+01	3.34E-01	3.41E-03
147Pm	2.64E-01	7.6797E-01	9.40E+02	8.85E-03	9.41E-06
151Sm	7.50E-03	9.9253E-01	2.60E+01	1.38E-03	5.31E-05
154Eu	8.07E-02	9.2247E-01	1.50E+02	3.40E-03	2.27E-05
155Eu	1.47E-01	8.6329E-01	1.40E+03	2.98E-04	2.13E-07
232Th	4.95E-11	1.0000E+00	1.10E-07	2.63E-07	2.39E+00
233U	4.36E-06	1.0000E+00	9.50E-03	4.85E-04	5.10E-02
234U	2.82E-06	1.0000E+00	6.20E-03	4.96E-05	8.00E-03
235U	9.84E-10	1.0000E+00	2.10E-06	1.55E-07	7.40E-02
236U	2.96E-08	1.0000E+00	6.30E-05	1.95E-06	3.10E-02
238U	1.55E-10	1.0000E+00	3.30E-07	4.95E-09	1.50E-02
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	4.08E-03	2.40E-04
239Pu	2.88E-05	9.9997E-01	6.20E-02	1.12E-05	1.80E-04
240Pu	1.06E-04	9.9989E-01	2.30E-01	2.53E-05	1.10E-04
241Pu	4.81E-02	9.5304E-01	1.10E+02	3.30E-03	3.00E-05
242Pu	1.85E-06	1.0000E+00	3.90E-03	1.33E-07	3.40E-05
TOTALS:				1.03E+00	2.58E+00

Table 6.16. Results for Fuel Test Capsules HRB-14 & HRB-15A

Sample Designation: HRB-14/15A
Computer Code Input File: HRB15A.I
Computer Code Output File: HRB15A.O
Reference Decay Date: 1 January 96
Contained Decay Heat, (W): 3.10E-02

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	5.36E-03	5.53E-07
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	1.49E-01	3.73E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	1.80E+00	1.20E-02
90Y	9.48E+01	9.7648E-01	2.50E+05	1.80E+00	7.20E-06
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	7.87E-03	5.62E-06
134Cs	3.35E-01	7.1534E-01	1.20E+03	3.74E-02	3.12E-05
137Cs	2.30E-02	9.7726E-01	9.80E+01	2.13E+00	2.17E-02
147Pm	2.64E-01	7.6797E-01	9.40E+02	8.26E-02	8.79E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	9.85E-03	3.79E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	4.68E-02	3.12E-04
155Eu	1.47E-01	8.6329E-01	1.40E+03	5.96E-03	4.26E-06
232Th	4.95E-11	1.0000E+00	1.10E-07	1.32E-06	1.20E+01
233U	4.36E-06	1.0000E+00	9.50E-03	3.56E-03	3.75E-01
234U	2.82E-06	1.0000E+00	6.20E-03	6.26E-04	1.01E-01
235U	9.84E-10	1.0000E+00	2.10E-06	1.55E-07	7.40E-02
236U	2.96E-08	1.0000E+00	6.30E-05	5.17E-06	8.20E-02
238U	1.55E-10	1.0000E+00	3.30E-07	6.47E-07	1.96E+00
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	5.44E-02	3.20E-03
239Pu	2.88E-05	9.9997E-01	6.20E-02	1.98E-03	3.20E-02
240Pu	1.06E-04	9.9989E-01	2.30E-01	3.45E-03	1.50E-02
241Pu	4.81E-02	9.5304E-01	1.10E+02	8.03E-01	7.30E-03
242Pu	1.85E-06	1.0000E+00	3.90E-03	4.68E-05	1.20E-02
TOTALS:				6.94E+00	1.47E+01

Table 6.17. Results for Fuel Test Capsule GF-3

Sample Designation:	GF-3
Computer Code Input File:	GF3.I
Computer Code Output File:	GF3.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	9.40E-03

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	8.93E-04	9.21E-08
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	4.83E-02	1.21E-04
90Sr	2.38E-02	9.7648E-01	1.50E+02	6.77E-01	4.51E-03
90Y	9.48E+01	9.7648E-01	2.50E+05	6.77E-01	2.71E-06
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	6.05E-04	4.32E-07
134Cs	3.35E-01	7.1534E-01	1.20E+03	1.54E-03	1.28E-06
137Cs	2.30E-02	9.7726E-01	9.80E+01	6.78E-01	6.92E-03
147Pm	2.64E-01	7.6797E-01	9.40E+02	7.08E-03	7.53E-06
151Sm	7.50E-03	9.9253E-01	2.60E+01	2.96E-03	1.14E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	8.51E-03	5.67E-05
155Eu	1.47E-01	8.6329E-01	1.40E+03	7.45E-04	5.32E-07
232Th	4.95E-11	1.0000E+00	1.10E-07	8.78E-07	7.98E+00
233U	4.36E-06	1.0000E+00	9.50E-03	1.91E-03	2.01E-01
234U	2.82E-06	1.0000E+00	6.20E-03	2.36E-04	3.80E-02
235U	9.84E-10	1.0000E+00	2.10E-06	8.82E-08	4.20E-02
236U	2.96E-08	1.0000E+00	6.30E-05	2.02E-06	3.20E-02
238U	1.55E-10	1.0000E+00	3.30E-07	3.96E-09	1.20E-02
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	1.65E-02	9.70E-04
239Pu	2.88E-05	9.9997E-01	6.20E-02	2.11E-05	3.40E-04
240Pu	1.06E-04	9.9989E-01	2.30E-01	3.22E-05	1.40E-04
241Pu	4.81E-02	9.5304E-01	1.10E+02	5.83E-03	5.30E-05
242Pu	1.85E-06	1.0000E+00	3.90E-03	3.12E-07	8.00E-05
TOTALS:				2.13E+00	8.32E+00

Table 6.18. Results for Fuel Test Capsule DR-GB2

Sample Designation:	DR-GB2
Computer Code Input File:	DRGB2.I
Computer Code Output File:	DRGB2.O
Reference Decay Date:	1 January 96
Contained Decay Heat, (W):	5.70E-03

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	8.93E-04	9.21E-08
54Mn	8.11E-01	4.4441E-01	8.30E+03	0.00E+00	0.00E+00
55Fe	2.54E-01	7.7569E-01	2.20E+03	0.00E+00	0.00E+00
60Co	1.32E-01	8.7634E-01	1.10E+03	0.00E+00	0.00E+00
59Ni	9.12E-06	9.9999E-01	8.10E-02	0.00E+00	0.00E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	0.00E+00	0.00E+00
85Kr	6.46E-02	9.3744E-01	4.00E+02	2.64E-02	6.60E-05
90Sr	2.38E-02	9.7648E-01	1.50E+02	4.58E-01	3.05E-03
90Y	9.48E+01	9.7648E-01	2.50E+05	4.58E-01	1.83E-06
99Tc	3.25E-06	9.9999E-01	1.70E-02	0.00E+00	0.00E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	0.00E+00	0.00E+00
125Sb	2.51E-01	7.7802E-01	1.40E+03	2.42E-04	1.73E-07
134Cs	3.35E-01	7.1534E-01	1.20E+03	2.56E-04	2.13E-07
137Cs	2.30E-02	9.7726E-01	9.80E+01	4.68E-01	4.78E-03
147Pm	2.64E-01	7.6797E-01	9.40E+02	1.36E-02	1.45E-05
151Sm	7.50E-03	9.9253E-01	2.60E+01	6.90E-03	2.65E-04
154Eu	8.07E-02	9.2247E-01	1.50E+02	1.70E-03	1.13E-05
155Eu	1.47E-01	8.6329E-01	1.40E+03	3.73E-04	2.66E-07
232Th	4.95E-11	1.0000E+00	1.10E-07	1.36E-06	1.24E+01
233U	4.36E-06	1.0000E+00	9.50E-03	8.17E-04	8.60E-02
234U	2.82E-06	1.0000E+00	6.20E-03	6.82E-05	1.10E-02
235U	9.84E-10	1.0000E+00	2.10E-06	2.05E-06	9.75E-01
236U	2.96E-08	1.0000E+00	6.30E-05	3.53E-06	5.60E-02
238U	1.55E-10	1.0000E+00	3.30E-07	2.54E-08	7.70E-02
237Np	3.24E-07	1.0000E+00	6.90E-04	0.00E+00	0.00E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	5.44E-04	3.20E-05
239Pu	2.88E-05	9.9997E-01	6.20E-02	4.90E-05	7.90E-04
240Pu	1.06E-04	9.9989E-01	2.30E-01	3.45E-05	1.50E-04
241Pu	4.81E-02	9.5304E-01	1.10E+02	1.76E-03	1.60E-05
242Pu	1.85E-06	1.0000E+00	3.90E-03	9.75E-09	2.50E-06
TOTALS:				1.44E+00	1.36E+01

6.1.5 Summation of Results for HTGR IFM

Table 6.19 summarizes the activity, in Ci for all of the HTGR IFM that were loaded into the HTGR FHU.

Table 6.19. Summation of Activity & Mass for HTGR FHU as of 1/1/96

Nuclide	Activity Ci	Mass g
3H	3.04E-01	3.13E-05
54Mn	0.00E+00	0.00E+00
55Fe	0.00E+00	0.00E+00
60Co	0.00E+00	0.00E+00
59Ni	0.00E+00	0.00E+00
63Ni	0.00E+00	0.00E+00
85Kr	9.19E+00	2.30E-02
90Sr	1.52E+02	1.02E+00
90Y	1.52E+02	6.10E-04
99Tc	0.00E+00	0.00E+00
106Ru	0.00E+00	0.00E+00
125Sb	1.15E-01	8.20E-05
134Cs	3.62E-01	3.02E-04
137Cs	1.57E+02	1.60E+00
147Pm	2.59E+00	2.75E-03
151Sm	1.28E+00	4.91E-02
154Eu	1.52E+00	1.01E-02
155Eu	1.49E-01	1.07E-04
232Th	2.10E-04	1.91E+03
233U	2.92E-01	3.07E+01
234U	3.13E-02	5.05E+00
235U	2.27E-04	1.08E+02
236U	1.04E-03	1.66E+01
238U	3.84E-06	1.16E+01
Total U	3.24E-01	1.72E+02
237Np	0.00E+00	0.00E+00
238Pu	2.91E+00	1.71E-01
239Pu	1.71E-02	2.75E-01
240Pu	1.91E-02	8.32E-02
241Pu	3.14E+00	2.85E-02
242Pu	1.08E-04	2.77E-02
Total Pu	6.08E+00	5.86E-01
Total of all isotopes	4.83E+02	

6.1.5.1 Packaging of HTGR IFM

The HTGR IFM were collected as a single unit in November 1995 and package into a primary enclosure (Fig. 6.1). The primary enclosure was then seal welded. This primary enclosure was then inserted into a secondary enclosure (Fig. 6.2). The secondary enclosure was also seal welded. The assembly of enclosures is identified as the HTGR FHU. A schematic of the HTGR FHU is shown in Fig. 6.3. The HTGR FHU is identified with stamped ¼" high characters on the upper end cap and stencil painted with 1" high characters along the side. The Material Safety Data Sheet (MSDS) for the paint used in given in Ref. 6-3. The estimated quantity of paint is 0.2 g. The individual identification number applied corresponds with the six-digit GA engineering drawing number associated with the enclosure. Since only the identifying numbers of the secondary enclosure are visible, this is used for identification of the loaded HTGR FHU assembly.

6.1.5.2 Uranium Isotopes Before and After Burn-up

The total uranium isotope distribution in the HTGR FHU before and after burn-up is presented in Table 6.20. There are no reportable quantities of ^{232}U in any of the referenced documents. Therefore it is assumed that there is < 5 ppm of ^{232}U in the FHU classifying the Detail Material Type as 71 as defined in Ref. 6-4.

Table 6.20. HTGR FHU Uranium Isotopic Distribution, Before and After Burn-up

Condition	^{233}U (g)	^{234}U (g)	^{235}U (g)	^{236}U (g)	^{238}U (g)	Total U (g)	Burn-up %
Before burn-up	0.00E+00	2.54E+00	1.89E+02	8.74E-01	1.26E+01	2.05E+02	
After burn-up	3.07E+01	5.05E+00	1.08E+02	1.66E+01	1.16E+01	1.72E+02	42.9%

6.1.5.3 Plutonium Isotopes After Burn-up.

The total plutonium isotope distribution in the HTGR FHU after burn-up is presented in Table 6.19.

6.1.5.4 Weight of HTGR FHU

The weight of the HTGR FHU is the combined weight of the fuel materials, 10.668 kg. (Ref. Table 2.2) plus the measured weight of the enclosures; 21.764 kg. or a total weight of 32.432 kg.

6.1.5.5 Measured Radiation Dose Rates for Unshielded HTGR FHU Assembly

After loading the HTGR IFM into the enclosures, the radiation dose rates were measured in six positions along the surface: Front (top, middle, and bottom) and Back (top, middle, and bottom). The readings were repeated at a distance of 1 m. from the surface. Results are summarized in Table 6.21.

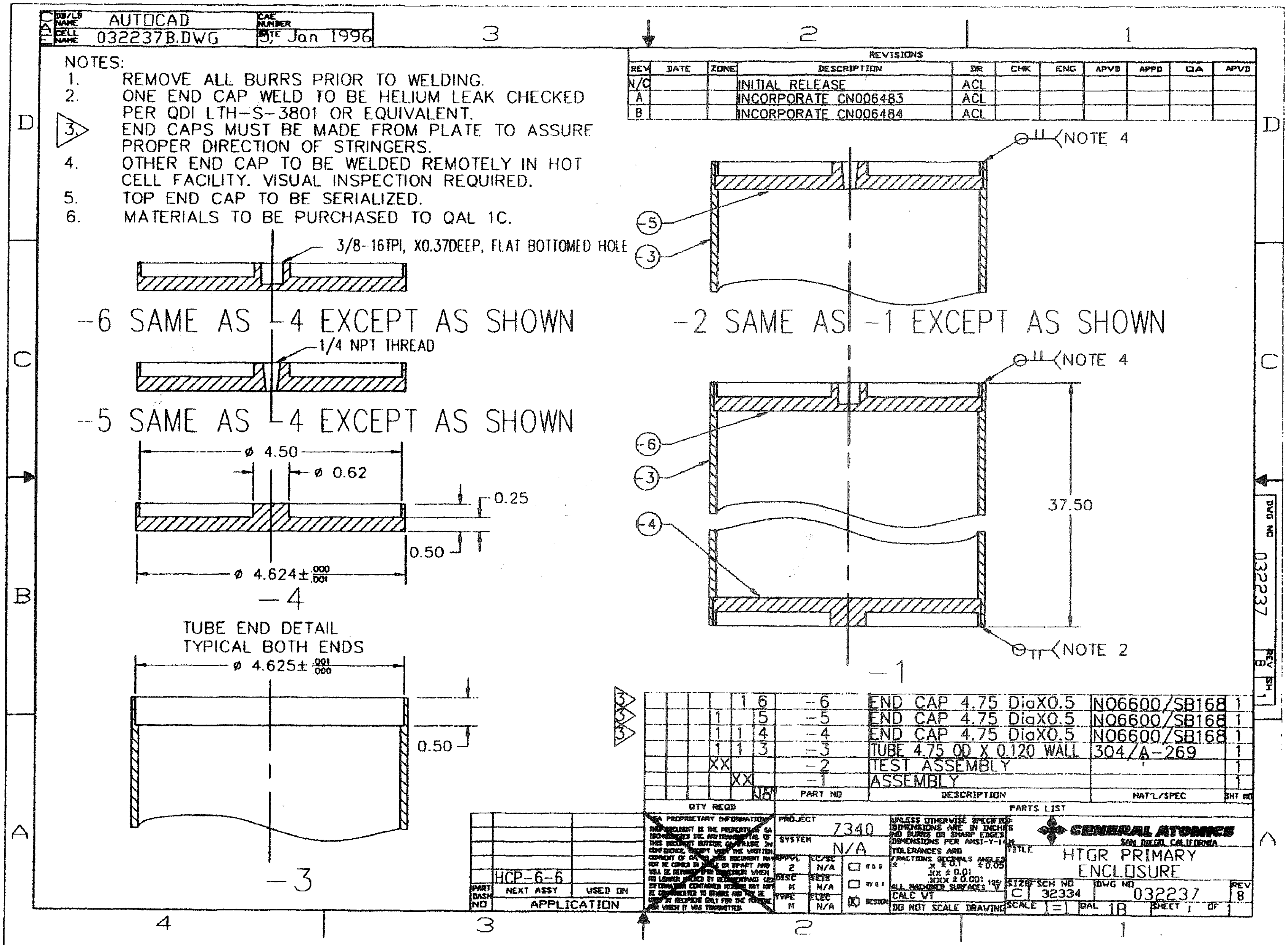

Fig. 6.1. HTGR Primary Enclosure, GA Dwg. No. 032237/B.

PC-000384/2

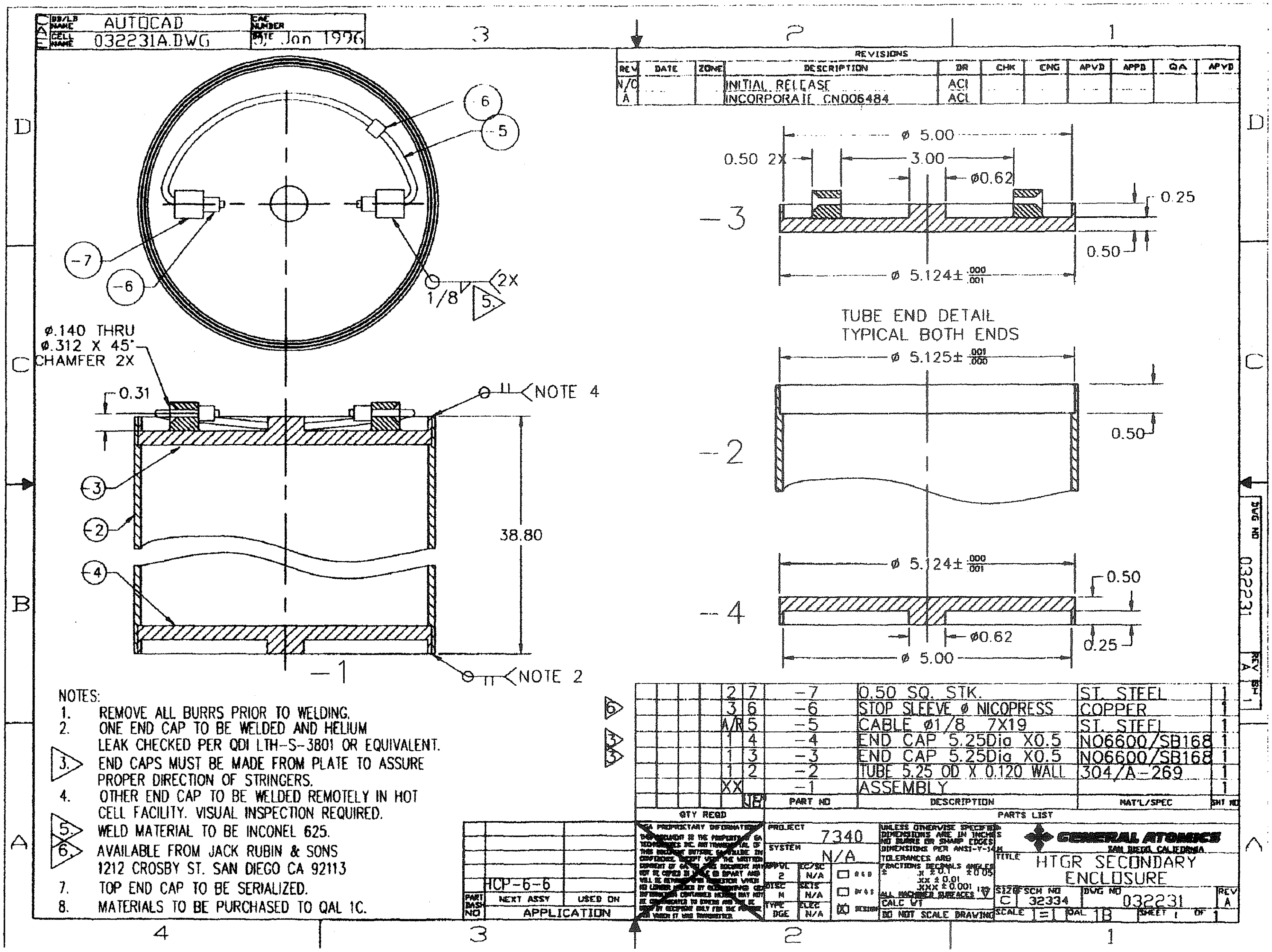

Fig. 6.2. HTGR Secondary Enclosure, GA Dwg. NO. 032231/A.

PC-000384/2

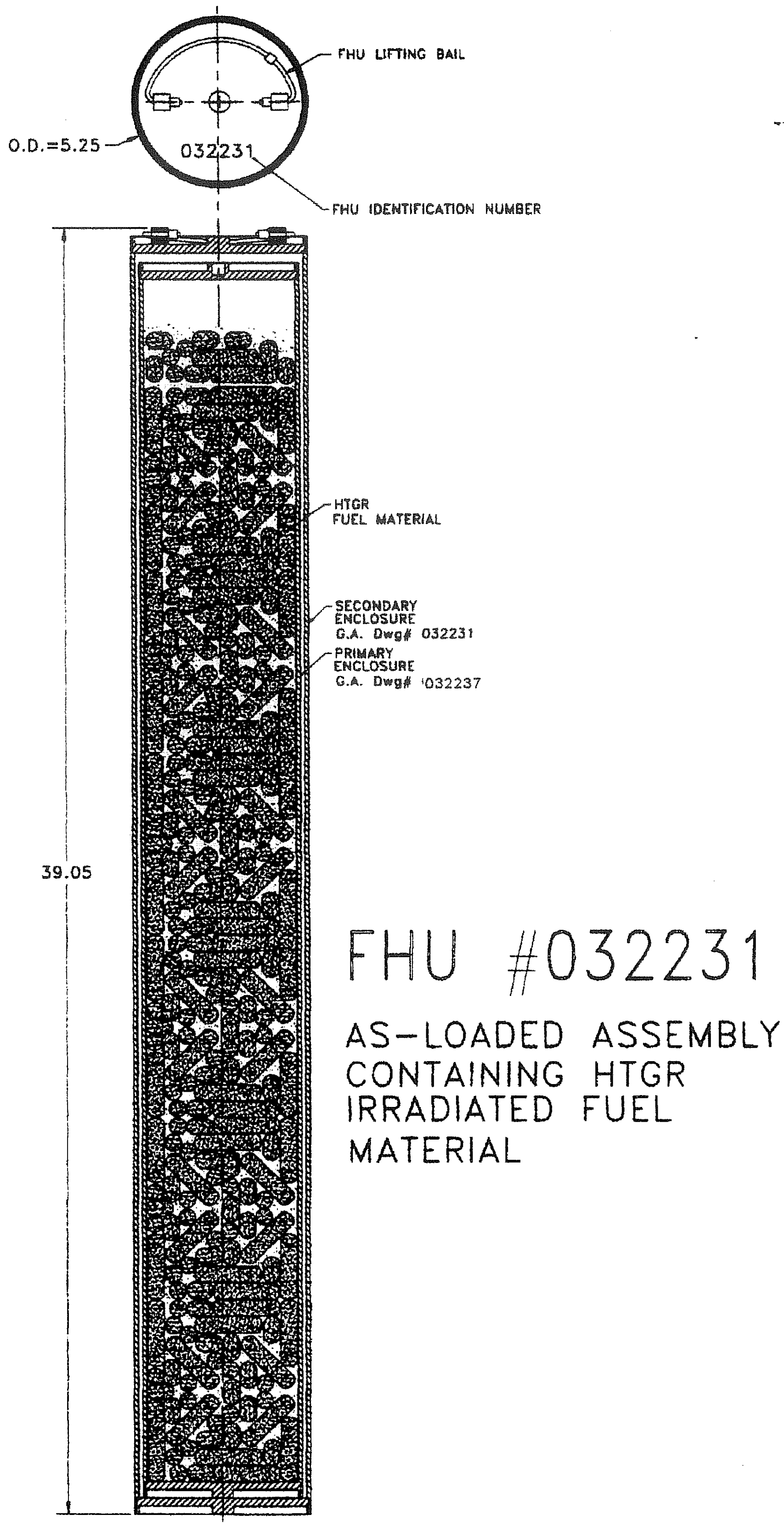

Fig. 6.3. FHU # 032231 (HTGR IFM) Assembly Schematic.

Table 6.21. Measured Radiation Dose Rates for Unshielded HTGR FHU Assembly

FHU ID	Vertical Survey Location[5]	Dose Rate (R-hr^{-1}) at indicated distance[6]			
		@ FHU surface		@ 1 m from surface	
		0°	180°	0°	180°
032231	Top	175	150	62	58
	Middle	1067	1033	100	92
	Bottom	708	583	81	80

6.1.5.6 Calculated Decay Heat

By summing the individual values for the decay heat from Tables 6.x, the decay heat for the HTGR FHU as of January 1, 1996 is 2.05E+00 watts.

6.2 RERTR IFM

The results of the RERTR irradiation is shown in Table 6.22.

6.2.1 Packaging of RERTR IFM

The RERTR IFM were collected in November 1995 and placed horizontally one at a time in a carrier basket. The description of the basket is given in Fig. 6.4. The basket was then lifted and lowered into the primary closure (Fig. 6.5). The primary closure was then seal welded. The primary enclosure was then inserted into a secondary enclosure (Fig. 6.6). The secondary enclosure was also seal welded. The assembly of enclosures is identified as the RERTR FHU. A schematic of the RERTR FHU is shown in Fig. 6.4. The RERTR FHU is identified with stamped ¼" high characters on the upper end cap and stencil painted with 1" high characters along the side. The MSDS for the paint used in given in Ref. 6-3. The estimated quantity of paint is 0.2 g. The individual identification number applied corresponds with the six-digit GA engineering drawing number associated with the enclosure. Since only the identifying numbers of the secondary enclosure are visible, this is used for identification of the loaded RERTR FHU assembly.

6.2.2 Summation of Results for RERTR FHU

6.2.2.1 Uranium Isotopes Before and After Burn-up

Total uranium in RERTR FHU before burn-up and after burn-up is shown in Table 6.23. There are no reportable quantities of ^{232}U in any of the referenced documents. Therefore it is assumed that there is < 5 ppm of ^{232}U in the FHU classifying the Detail Material Type as 71 as defined in Ref. 6-4.

---

[5] Vertical survey location identified as "Top" was taken ~3" down from top closure weld; the "Middle" survey point was taken at the FHU longitudinal center-line; the "Bottom" survey point was taken ~3" up from bottom closure weld.

[6] All survey measurements were taken on 5 December 95, using the following instrumentation: Gamma Ionization Chamber, Model #RS-C4-1606-203, Serial #Y7210, manufactured by Reuter-Stokes, Twinsburg, OH; Calibration Due Date: 9 May 96.

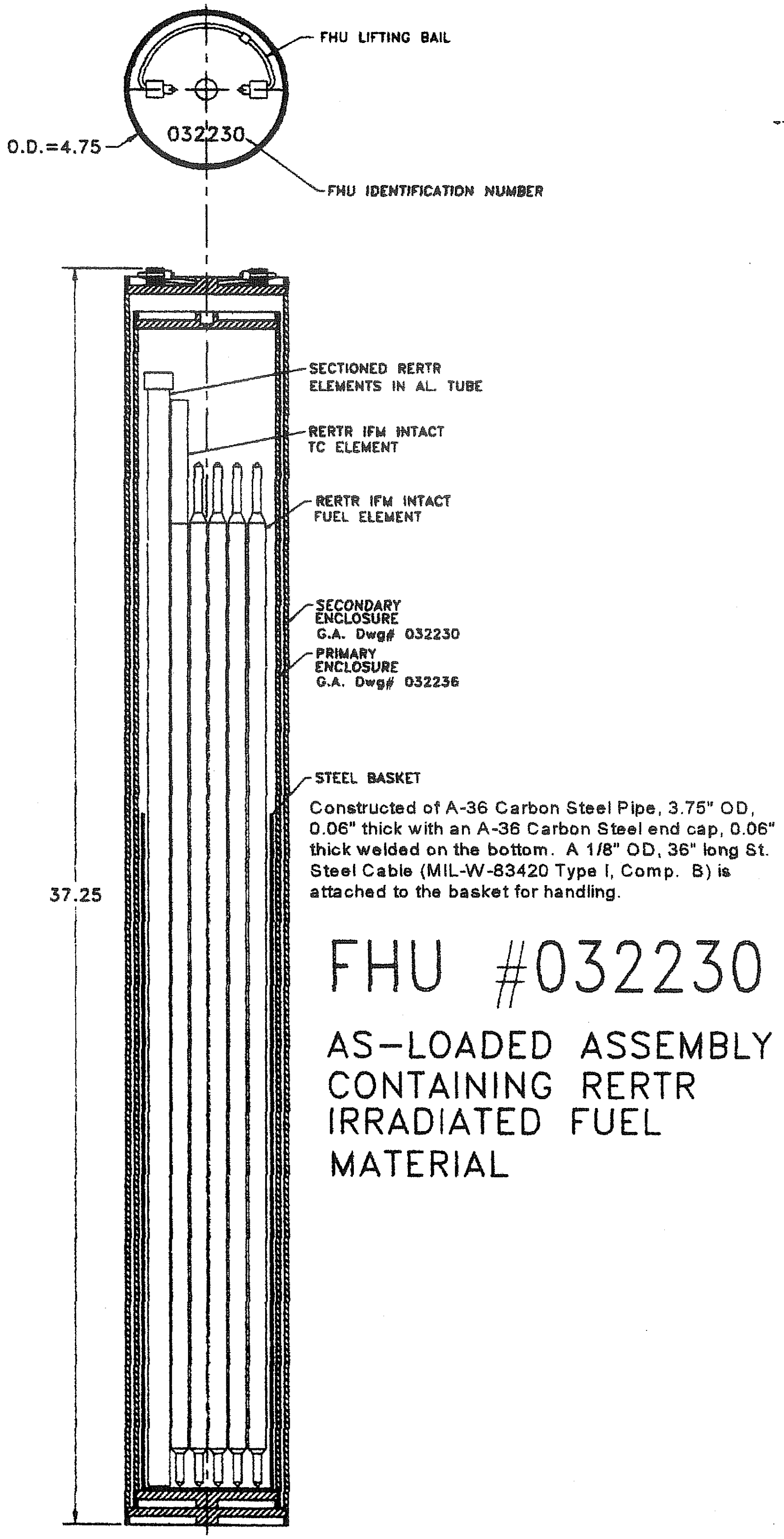

Fig. 6.4. FHU #032230 (RERTR IFM) Assembly Schematic

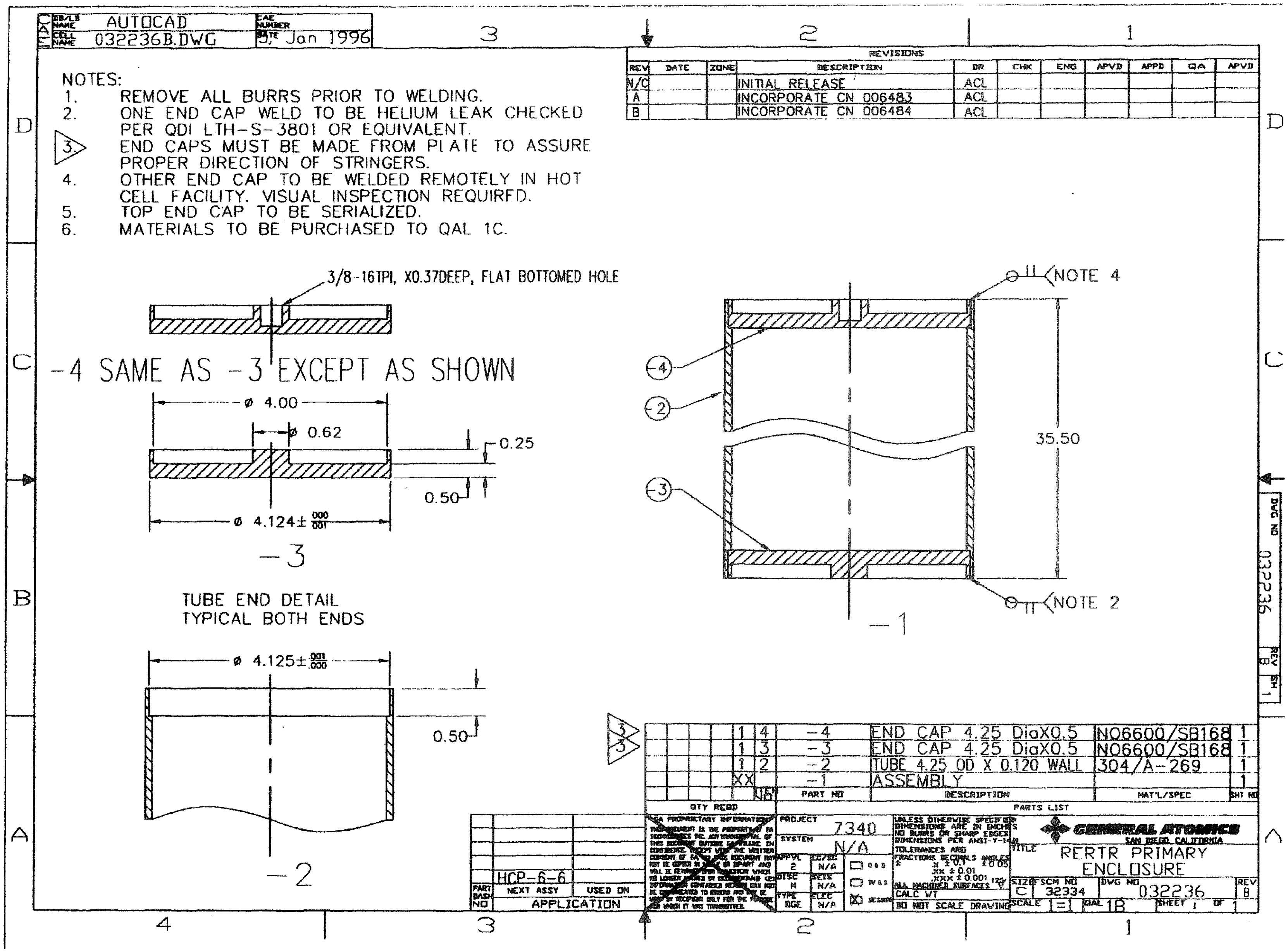

Fig. 6.5. RERTR Primary Enclosure, GA Dwg. # 032236/B

PC-000384/2

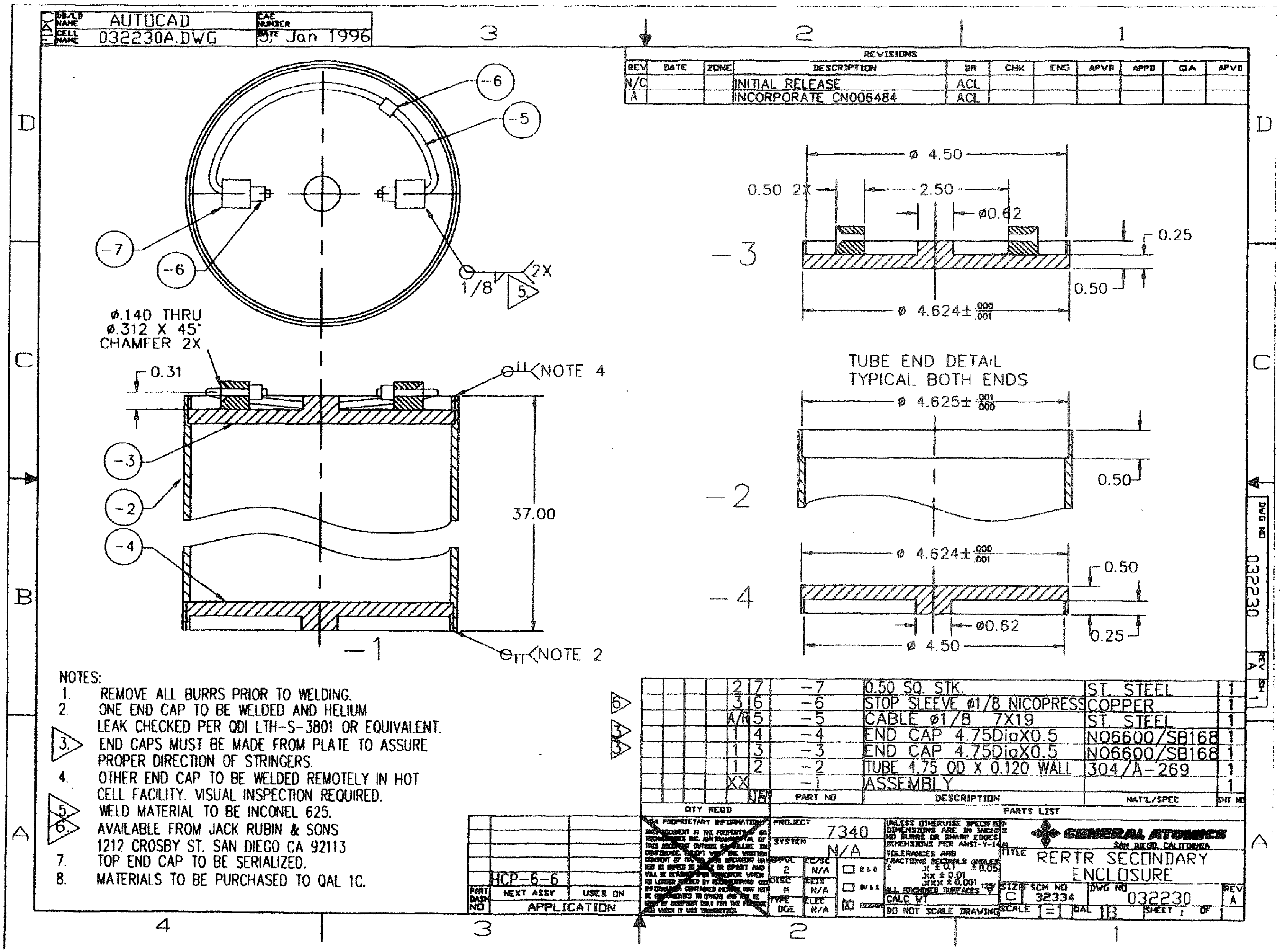

Fig. 6.6. RERTR Secondary Enclosure, GA Dwg. # 032230/A

Table 6.22. Results for RERTR Fuel

Sample Designation: RERTR
Computer Code Input File: RERTR.I
Computer Code Output File: RERTR.O
Reference Decay Date: 1 January 96
Contained Decay Heat, (W): 1.10E+01

Nuclide	Decay k yr^{-1}	Decay/yr $1 - e^{-kt}$	Specific Act. Ci/g	Activity Ci	Mass g
3H	5.64E-02	9.4516E-01	9.70E+03	2.50E+00	2.58E-04
54Mn	8.11E-01	4.4441E-01	8.30E+03	1.15E-02	1.39E-06
55Fe	2.54E-01	7.7569E-01	2.20E+03	3.06E+01	1.39E-02
60Co	1.32E-01	8.7634E-01	1.10E+03	2.46E+00	2.24E-03
59Ni	9.12E-06	9.9999E-01	8.10E-02	3.30E-01	4.07E+00
63Ni	6.93E-03	9.9309E-01	4.60E+01	3.96E+01	8.61E-01
85Kr	6.46E-02	9.3744E-01	4.00E+02	5.86E+01	1.46E-01
90Sr	2.38E-02	9.7648E-01	1.50E+02	7.60E+02	5.07E+00
90Y	9.48E+01	9.7648E-01	2.50E+05	7.60E+02	3.04E-03
99Tc	3.25E-06	9.9999E-01	1.70E-02	1.40E-01	8.24E+00
106Ru	6.80E-01	5.0662E-01	3.40E+03	6.67E-01	1.96E-04
125Sb	2.51E-01	7.7802E-01	1.40E+03	4.18E+00	2.99E-03
134Cs	3.35E-01	7.1534E-01	1.20E+03	2.29E+01	1.91E-02
137Cs	2.30E-02	9.7726E-01	9.80E+01	8.26E+02	8.43E+00
147Pm	2.64E-01	7.6797E-01	9.40E+02	9.44E+01	1.00E-01
151Sm	7.50E-03	9.9253E-01	2.60E+01	3.35E+00	1.29E-01
154Eu	8.07E-02	9.2247E-01	1.50E+02	2.39E+01	1.59E-01
155Eu	1.47E-01	8.6329E-01	1.40E+03	6.71E+00	4.79E-03
232Th	4.95E-11	1.0000E+00	1.10E-07	0.00E+00	0.00E+00
233U	4.36E-06	1.0000E+00	9.50E-03	1.71E-07	1.80E-05
234U	2.82E-06	1.0000E+00	6.20E-03	3.91E-04	6.30E-02
235U	9.84E-10	1.0000E+00	2.10E-06	7.39E-04	3.52E+02
236U	2.96E-08	1.0000E+00	6.30E-05	5.61E-03	8.90E+01
238U	1.55E-10	1.0000E+00	3.30E-07	8.58E-04	2.60E+03
Total U				7.60E-03	3.04E+03
237Np	3.24E-07	1.0000E+00	6.90E-04	2.48E-03	3.60E+00
238Pu	7.90E-03	9.9213E-01	1.70E+01	0.00E+00	0.00E+00
239Pu	2.88E-05	9.9997E-01	6.20E-02	1.30E+00	2.10E+01
240Pu	1.06E-04	9.9989E-01	2.30E-01	1.35E+00	5.86E+00
241Pu	4.81E-02	9.5304E-01	1.10E+02	2.84E+02	2.58E+00
242Pu	1.85E-06	1.0000E+00	3.90E-03	3.35E-03	8.60E-01
Total Pu				2.87E+02	3.03E+01
TOTALS:				2.92E+03	3.10E+03

Table 6.23. RERTR FHU Uranium Isotopic Distribution, Before and After Burn-up

Condition	^{233}U (g)	^{234}U (g)	^{235}U (g)	^{236}U (g)	^{238}U (g)	Total U (g)	Burn-up %
Before burn-up	0.00E+00	7.32E+00	7.62E+02	3.27E-01	3.08E+03	3.85E+02	
After burn-up	1.80E-05	6.30E-02	3.52E+02	8.90E+01	2.60E+03	3.04E+03	53.8%

6.2.2.2 Weight of FHU

The weight of the FHU is the combined weight of the fuel materials, 10,766.54 kg. (Ref. Table 2.4), the estimated weight of the non-fuel element components (3.433 kg), the measured tare weight of the internal steel basket (2.048 kg.), plus the measured weight of the enclosures; 18.226 kg. or a total weight of 34.474 kg.

6.2.2.3 Measured Radiation Dose Rates for Unshielded FHU Assembly

After loading the RERTR IFM into the enclosures, the radiation dose rates were measured in six positions along the surface: Front (top, middle, and bottom) and Back (top, middle, and bottom). The readings were repeated at a distance of 1 m. from the surface. Results are summarized in Table 6.24.

Table 6.24. Measured Radiation Dose Rates for Unshielded RERTR FHU Assembly

FHU ID	Vertical Survey Location[7]	Dose Rate ($R\text{-}hr^{-1}$) at indicated distance[8]			
		@ FHU surface		@ 1 m from surface	
		0°	180°	0°	180°
032230	Top	733	616	235	208
	Middle	3507	2782	349	418
	Bottom	1502	1319	341	381

6.2.2.4 Plutonium Isotopes After Burn-Up

The total plutonium isotope distribution in the RERTR FHU after burn-up is presented in Table 6.22.

[7] Vertical survey location identified as "Top" was taken ~3" down from top closure weld; the" Middle" survey point was taken at the FHU longitudinal center-line; the "Bottom" survey point was taken ~3" up from bottom closure weld

[8] All survey measurements were taken on 5 December 95, using the following instrumentation: Gamma Ionization Chamber, Model #RS-C4-1606-203, Serial #Y7210, manufactured by Reuter-Stokes, Twinsburg, OH; Calibration Due Date: 9 May 96.

7.0 VERIFICATION

Verification of the gross activity in the HTGR and RERTR irradiated fuel materials (IFM) was performed, based on analysis of the contained gamma emitting radionuclides contained in each of the two material groups. Two separate measurements of the gamma radiation dose rates from the HTGR IFM and RERTR IFM packages were made. The results of these measurements were compared with the calculated dose rates, utilizing the activity data for the gamma-emitting constituents, as reported in Section 6 of this document. A summary of the verification methodology is presented in the following paragraphs.

It should be noted that this verification study was based on gamma radiation dose measurements of the two groups of Irradiated Fuel Materials prior to the final consolidation and weld-encapsulation of the IFM, as described in Section 6. At that time, both groups of IFM were contained in separate 5 gal. capacity steel cans, which was the normal storage configuration for material archival storage in the Hot Cell Facility below-ground storage wells.

7.1 Gamma Dose Rate Measurements

The gamma radiation dose rate of both the HTGR IFM and RERTR IFM were measured on 4 January 94, (prior to the final consolidation of the two fuel types and subsequent weld-encapsulation of each type into the Primary and Secondary Enclosures in which these materials are presently contained and stored). This work was performed remotely, in the Low-Level Cell (Rm. 23/113) of the GA Hot Cell Facility, using a 2" diameter ionization chamber. The detector was positioned at the exterior surface of each of the IFM storage packages for the surveys. The physical orientation of the detector with respect to the IFM package, was adjusted during the surveys, as necessary, to obtain the maximum contact dose rates.

7.1.1 HTGR IFM Survey

The HTGR IFM package was comprised of two, vertically-stacked 5 gal. capacity steel cans (type DOT-37A), containing the entire mass of HTGR IFM. The irradiated fuel materials in each of the cans occupied approximately 75% of the can internal volume. The DOT-37A cans have been used at the GA Hot Cell Facility, for retrievable fuel storage containment. The cans were of steel construction, with the following dimensions: ~ 11.25" (~ 28.6 cm) ID x ~ 12.5" (~ 31.8 cm) interior height, with wall and closure lid thickness of ~ 0.024" (~ 0.6 mm).

The maximum unshielded radiation dose rate, measured on 4 January 94, at the exterior surface of the two vertically-stacked HTGR IFM cans, was ~ 800 R/hr (γ).

7.1.2 RERTR IFM Survey

The RERTR IFM package containment consisted of a welded vertical assembly of three 5 gal. capacity steel cans (type DOT-37A), with an open interior to accommodate the RERTR fuel element length of 29.92" (76.0 cm). For survey measurements, the vertical stack assembly contained the entire RERTR IFM mass.

The maximum unshielded radiation dose rate, measured on 4 January 94, at the exterior surface of the three-high vertically-stacked HTGR IFM can assembly, was ~ 5200 R/hr (γ).

7.2 Gamma Dose Rate Calculations

The gamma radiation dose rate calculations for both the HTGR IFM and RERTR IFM were performed with the PATH shielding code, (Ref. 7-1), using the activity calculated as of 1 January 94. The deviation of this date from the measurement date of 4 January 94 should pose no significant impact on the dose rate results.

For the HTGR IFM, the primary contributing nuclide to the overall gamma rate was the $^{137}Cs/^{137m}Ba$. The total activity level of $^{137}Cs/^{137m}Ba$ from the entire mass of HTGR IFM, as of 1 January 94, was 155 Ci.

For the gamma dose rate calculation, the $^{137}Cs/^{137m}Ba$ activity was equally distributed in the two 5 gal. capacity cans, each can containing 77.5 Ci. The total mass of the HTGR IFM is ~14 kg, which was also split between the two cans for calculation purposes. The volumetric source activity in each can is 5.07E-03 Ci/cm^3 $^{137}Cs/^{137m}Ba$, based on the ¾ full volume of each of the 5 gal. cans.

The RERTR IFM was contained in a three-high vertical stack of 5 gal. capacity cans. The active height of the irradiated fuel elements is 22 in. This height was used to obtain the volumetric source activity in the package. The total mass of the RERTR IFM is ~ 15 kg.

The gamma emitters considered in the RERTR IFM included ^{60}Co, ^{134}Cs, $^{137}Cs/^{137m}Ba$, and ^{154}Eu, with 3.2, 44.8, 819, and 28.1 Ci respectively. The corresponding volumetric activities are 8.95E-05 Ci/cm^3 ^{60}Co, 1.25E-03 Ci/cm^3 ^{134}Cs, 2.29E-02 Ci/cm^3 $^{137}Cs/^{137m}Ba$, and 7.84E-04 Ci/cm^3 ^{154}Eu.

For each IFM package, the dose rate was calculated at 1 in. from the surface of the 5 gal. capacity can containment. The 1 in. distance represents the effective centerline of the ionization chamber detector used for the measurements. The results of the calculations are provided below:

HTGR IFM	~ 950 R/hr (γ)
RERTR IFM	~ 7100 R/hr (γ)

The 950 R/hr for the HTGR IFM includes the contributions from two 5 gal. capacity cans. The calculated maximum dose rate from a single 5 gal. capacity can was ~ 820 R/hr.

7.3 Comparison Between Measurement and Calculation

The calculated gamma dose rates are higher than the measured results for both the HTGR IFM and RERTR IFM packages. The ratios of calculation to measurement values are ~ 1.2 for the HTGR IFM and ~ 1.4 for the RERTR IFM. These ratios are within the calculational uncertainties for this IFM characterization. Therefore, comparison of the calculated and measured gamma dose rates confirms that the calculated activity is conservative, and is considered acceptable for both the HTGR IFM and RERTR IFM packages.

7.4 Input and Output Files

The PATH computer code Input and Output data files for the gamma dose calculations are stored on the PC disk, as follows:

IFM Package	Input File	Output File
HTGR	HCELL01.I	HCELL01.O
RERTR	HCELL02.I	HCELL02.O

8.0 DEFINITIONS

atom/b-cm: Atoms per barn-centimeter: A unit of atomic density, used herein to define the effective density of individual isotope constituents present in a fuel target entity during a reactor irradiation.

AVR: Arbeitsgemeinschaft Versuchs-Reaktor, (German Pebble-Bed Reactor); Julich, Germany FRG.

b: barn: A unit of area equal to 1.0E-24 cm^2. This unit is primarily used to define the effective cross-sectional area of an atom for nuclear interaction.

BISO: A multi-layered, high-temperature chemical vapor deposition, spherical fuel particle coating design, comprised of an isotropic pyrolytic-carbon coating, over a relatively low-density pyrolytic-carbon buffer coating, encapsulating a ceramic fuel particle kernel.

BOL: Beginning-of-Life: Refers to the pre-irradiation condition of the target material at the time the reactor irradiation exposure is started.

Bq: Becquerel: A standard, derived unit of radioactivity, equal to the activity of a radionuclide decaying at the rate of one spontaneous nuclear transition, (or disintegration), per second; i.e., 1 Bq = 1 dps.

CEA: Commissariat a L'Energie Atomique, (French Atomic Energy Commission).

Ci: Curie: A standard, derived unit of radioactivity, used herein to specify IFM activity levels at a given Reference Decay Date. 1 Ci = 3.70E+10 Bq.

D&D: Decontamination and Decommissioning (Project).

DOE: (U.S.) Department of Energy, Washington, DC.

dps: Disintegrations per second: A measured unit of radioactivity; (also see definitions of Bq and Ci). 1 dps = 1 Bq.

DRAGON: Dragon Test Reactor, (English HTGR Reactor), Winfrith, United Kingdom.

EFPD: Effective Full-Power Days.

ELE: (Fuel) Element.

EOL: End-of-Life: Refers to the post-irradiation condition of the target material at the time the reactor irradiation exposure is terminated.

ETR: Engineering Test Reactor, Aerojet Nuclear Company, Idaho National Engineering Laboratory, Idaho Falls, ID.

ev: Electron Volt: A unit of energy equal to ~ 1.602E-19 joules. This term is used herein to specify the average velocity of neutrons to which a fuel target entity is exposed during reactor irradiation.

Φ_{FAST}: Peak Fast Neutron Fluence to which a fuel target entity has been exposed during reactor irradiation, in nvt, unless otherwise indicated.

Fast Neutron: In the context of this document, the term Fast Neutron is used to specify a population of neutrons, present in a reactor irradiation environment, with a velocity, (or energy), greater than 1.80E+05 ev, (i.e., E > 0.18 MeV). Also see definition of Peak Fast Fluence.

FHU Fuel Handling Unit, a term used to describe the separate independent assembly, used to contain and handle the consolidated HTGR and RERTR Irradiated Fuel Materials described herein. In this application, the FHU assemblies are each comprised of double-containment seal-welded 304SS construction Primary and Secondary Enclosures.

FIMA: Fissions per Initial (Heavy) Metal Atom: A term related to the fractional depletion of fissile materials contained in a target assembly, under specifically defined irradiation exposure conditions; FIMA is usually expressed as "percent burn-up" or "%FIMA".

FSV: Fort St. Vrain Nuclear Generating Station, Public Service Company of Colorado, Platteville, CO.

FTE: Fuel Test Element.

GA: General Atomics, San Diego, CA.

GETR: General Electric Test Reactor, General Electric Company, Vallecitos, CA.

HCF: Hot Cell Facility, Bldg. 23, General Atomics, San Diego, CA.

HEU: High-Enrichment Uranium, i.e., material comprised of, or containing, uranium of > 20.0% ^{235}U Enrichment. In the context of this report, the term HEU is used to describe the "fully" enriched uranium used as raw material by GA for the manufacture of HTGR fuel particles, (i.e., ~ 93.15% ^{235}U Enrichment).

HFIR: High-Flux Irradiation Reactor, Oak Ridge National Laboratory, Oak Ridge, TN.

HOBEG: Hochtemperaturreaktor-Brennelement, GmbH; (German HTGR fuel manufacturer).

HTGR: High-Temperature Gas-Cooled Reactor.

IFM: Irradiated Fuel Materials: A collective term which refers to the collective archival irradiated fuel specimens previously stored in the Hot Cell Facility, Bldg. 23, General Atomics, San Diego, CA.

INEEL Idaho National Engineering and Environmental Laboratory.

KFA: Kernforschungsanlage Julich, GmbH; (German HTGR fuel manufacturer); Julich, Germany FRG.

LEU: Low-Enrichment Uranium, i.e., material comprised of, or containing, uranium of ≤ 20.0% ^{235}U Enrichment.

LWR: Light Water Reactor.

MSDS: Material Safety Data Sheets

NR: Not recorded.

NUKEM: Nukem, GmbH, (German HTGR Fuel Manufacturer).

nvt: Neutron-Velocity-Time: A unit of neutron fluence integrated over time. This term is used in herein to describe the effective neutron fluence to which a fuel target entity is exposed during a reactor irradiation.

ORNL: Oak Ridge National Laboratory, administered for the U.S. Department of Energy by Martin-Marrietta Energy Systems, Inc., Oak Ridge, TN.

ORR: Oak Ridge 30 MW(t) Research Reactor, Oak Ridge National Laboratory, Oak Ridge, TN.

ρ: Gravimetric density, in g/cm^3, unless otherwise indicated.

ρ_A: Atomic Density, used herein to define the effective density of individual isotope constituents present in a fuel target entity during a reactor irradiation. Units are expressed as atoms per barn-centimeter (atom/b-cm), unless otherwise specified.

ρ_P: Power Density in reactor during fuel target entity irradiation, in W/cm^3, unless otherwise indicated.

P(t): Peak Power (thermal) generated in fuel target entity during reactor irradiation, in W, unless otherwise indicated.

PB: Peach Bottom Atomic Power Station, Unit No. 1, Philadelphia Electric Company, Delta, PA.

PIE: Post-Irradiation Examination: A general term to describe the activities involved in the remote examination of an irradiated target assembly in a Hot Cell laboratory. Activities include receipt, off-loading, remote visual examination, macro- and micro-photo-documentation, remote metrology, disassembly, mechanical sectioning, physical testing, sampling, preparation and polishing of metallographic specimens, and metallographic examination of various target assembly component materials.

RDD: Reference Decay Date: The time at which measurement of transient characterization parameters has been performed. For calculated transient parameters, the RDD is the time to which the reported values are normalized. Examples of transient parameters in this report are IFM contained activity levels (Ci), or contained decay heat (W).

RERTR: Reduced-Enrichment Research & Test Reactor fuel development program.

RX: Reactor

SILOE: Siloe Test Reactor, Commissariat a L'Energie Atomique, Grenoble, France.

SNM: Special Nuclear Material: i.e., ^{233}U, ^{235}U, and Pu (all isotopes).

SX: (Fuel) Sample, used herein to define the individual fuel entities associated with each fuel target inventory group represented in the subject irradiated fuel materials.

TRISO: A multi-layered, high-temperature chemical vapor deposition, spherical fuel particle, comprised of an outer isotropic pyrolytic-carbon coating, over a silicon-carbide coating, over an inner isotropic pyrolytic-carbon coating, over a relatively low-density pyrolytic-carbon buffer coating, encapsulating a ceramic fuel particle kernel.

V: Volume of fuel element or fuel target sample entity, expressed as cm^3, unless otherwise indicated.

VSM: Vanek, Simnad, Meyers; A process developed by GA used in the manufacture of HTGR fuel particle carbide kernels. The VSM kernel manufacturing process entails the melting and spheridization of UC_2, $(Th,U)C_2$, and ThC_2 fuel material, under inert atmosphere conditions, in a high-temperature drop furnace. The resulting product is a high-density, sintered fuel kernel, which is used as substrate for subsequent fuel particle coating processes. The term VSM honors the three General Atomics engineers involved in the original development of this process.

W: Watt: A standard unit of power, used herein to specify a quantity of decay heat associated with the various groups of highly-radioactive irradiated fuel materials. One watt is the power which gives rise to the production of energy at the rate of one joule per second.

WAR: Weak-Acid Resin-based spherical kernel, utilized by GA, to a very limited extent, for advanced HTGR fuel fabrication processes. In this fuel kernel manufacturing method, WAR microspheres were brought to specific heavy metal loading levels in the WAR matrix by chemical adsorption, by immersion in a saturated aqueous solution. The loaded microspheres were utilized as fuel particle substrate material, which were then sintered and subsequently coated to yield BISO and/or TRISO particle product.

9.0 LIST OF REFERENCES

1-1 "General Atomics Hot Cell Decontamination & Decommissioning Project, Phases 2 and 3 Activities; Statement of Work", Attachment I to USDOE Contract DE-AC03-95SF20798, issued to General Atomics, San Diego, CA.

1-2 Deleted.

1-3 "Encapsulation and Transfer of HTGR/RERTR Irradiated Fuel Material Between Bldgs. 23 and 30", Hot Cell Operating Procedure Document No. HCD-6.6, Hot Cell Decommissioning Project, General Atomics, San Diego, CA; Issue A, dated 13 November 95 and supporting documents.

1-4 "Dry Well Storage Fuel Receipt Criteria Questionnaire", Form WINCO-6468AX (10-87), Lockheed-Martin Idaho Technologies, Idaho National Engineering Laboratory, Idaho Falls, ID, Revision dated October 1987.

1-5 Greenwood, J. S., "Safeguards and Security Measures for the Irradiated Fuel Material Temporary Storage Facility at General Atomics", GA Document No. PC-000457, GA Hot Cell Decommissioning Project, General Atomics, San Diego, CA; Rev. 1, Janurary 2000.

1-6 Deleted.

2-1 Boshoven, J., "Hot Cell Facility Stored Irradiated Fuel Materials Inventory Report", GA Document No. PC-000377, GA Hot Cell Decommissioning Project, General Atomics, San Diego, CA; Revision 0, December 2, 1993.

3-1 Aycock, M. T., *et al*, "General Atomics Irradiated Fuel Materials and Non-Fuel Radioactive Waste Materials Characterization Plan", GA Document No. PC-000382, GA Hot Cell Decommissioning Project, General Atomics, San Diego, CA; Revision 2, October 1995.

3-2 Shirley, G., "User's Manual and Description for GARGOYLE: A Fuel Cycle Analysis Code with a Decay Heat Calculation Capability", CEGA Document No. CEGA-002922, CEGA/General Atomics, San Diego, CA; Revision N/C, October 20, 1993.

3-3 "ORIGEN2: Isotope Generation and Depletion Code", Document No. CCC-371, RSIC Computer Code Collection, Oak Ridge National Laboratory, Martin-Marrietta Energy Systems, Inc., Oak Ridge, TN; September 1989.

3-4 Rucker, R. A.., "Extended Decay Heat Values for the GT and PC-MHR", GA Document No. 816:RAR:0010:94, August 12, 1994.

3-5 Su, S., "Radiotoxicity of PC-MHR Spent Fuel Elements", GA Document No. 816:SS:012:95, May 23, 1995.

5-1 Miller, C. M., "Pre-Irradiation Characterization of FSV Fuel Test Elements FTE-1 through FTE-8", GA Document No. 904451, General Atomic Company, San Diego, CA; Issue A, May 1, 1980.

5-2 McCord, F. and Miller, C. M., "Test Procedure for Destructive Examination of FSV FTE-2, S/N 8-0206", GA Document No. 906771, GA Technologies, Inc., San Diego, CA; Issue B, January 27, 1986.

5-3 McCord, F., "Destructive Examination of Fort St. Vrain Test Element FTE-2", GA Document No. 908909, GA Technologies, Inc., San Diego, CA; Issue N/C, July 11, 1986.

5-4 Miller, C. M. and Saurwein, J. J., "Non-Destructive Examination of 51 Fuel and Reflector Elements from Fort St. Vrain Core Segment 1", U.S. Department of Energy Report No. GA-A16000, General Atomic Company, San Diego, CA; December 1980.

5-5 McCord, F., "Post-Irradiation Examination and Evaluation of FSV Fuel Element S/N 1-2415", GA Document No. 907079, GA Technologies, Inc., San Diego, CA, Issue N/C, November 7, 1984.

5-6 Wallroth, C. F. *et al*, "Post-Irradiation Examination of Peach Bottom Fuel Test Element FTE-3", U.S. Atomic Energy Commission Report No. GA-A13004, General Atomic Company, San Diego, CA; August 15, 1974.

5-7 Wallroth, C. F. *et al*, "Post-Irradiation Examination of Peach Bottom Fuel Test Element FTE-4", U.S. Energy Research & Development Administration Report No. GA-A13452, General Atomic Company, San Diego, CA; July 1977.

5-8 Wallroth, C. F. *et al*, "Post-Irradiation Examination and Evaluation of Peach Bottom Fuel Test Element FTE-6", U.S. Energy Research & Development Administration Report No. GA-A13943, General Atomic Company, San Diego, CA; September 1977.

5-9 Holzgraf, J. F. *et al*, "Post-Irradiation Examination and Evaluation of Peach Bottom Fuel Test Elements FTE-14 and FTE-15", U.S. Department of Energy Report No. GA-A13944, General Atomic Company, San Diego, CA; February 1979.

5-10 Holzgraf, J. F. *et al*, "Post-Irradiation Examination and Evaluation of Peach Bottom Fuel Test Element FTE-16", U.S. Department of Energy Report No. GA-A15727, General Atomic Company, San Diego, CA; January 1980.

5-11 Saurwein, J. J. *et al*, "Final Report on the Peach Bottom Test Element Program", U.S. Department of Energy Report No. GA-A15999, GA Technologies, Inc., San Diego, CA; November 1982.

5-12 Nabielek, H. and Goodin, D. T., "German TRISO Fuel Particle Performance", GA Document No. 907779, GA Technologies, Inc., San Diego, CA; Issue 0, January 16, 1985.

5-13 Stansfield, O. M. *et al*, "Review of Pebble Bed HTGR Fuel and Graphite Technology for Potential Application in the U.S.", GA Document No. 907634, GA Technologies, Inc., San Diego, CA; Issue 1, September 28, 1985.

5-14 Wattier, J. B., "Final Test Operations Report - Capsule P13P", U.S. Energy Research & Development Administration Report No. GA-A13332, General Atomic Company, San Diego, CA; July 31, 1975.

5-15 Smith, C. L., "Pre-Irradiation Report: Fuel Materials for Irradiation P13Q", U.S. Energy Research & Development Administration Report No. GA-A13265, General Atomic Company, San Diego, CA; June 13, 1975.

5-16 Young, C. A. and Scott, C. B., "Post-Irradiation Examination of Capsule P13Q", U.S. Energy Research & Development Administration Report No. GA-A14174, General Atomic Company, San Diego, CA; September 1977.

5-17 Young, C. A. and Harmon, D. P., "Pre-Irradiation Report: Fuel Material for P13R and P13S Irradiation Capsules", U.S. Atomic Energy Commission Report No. GA-A13026, General Atomic Company, San Diego, CA; November 1974.

5-18 Scott, C. B. *et al*, "Post-Irradiation Examination of Capsules P13R and P13S", U.S. Energy Research & Development Administration Report No. GA-A13827, General Atomic Company, San Diego, CA; October 8, 1976.

5-19 Young, C. A. and Harmon, D. P., "Pre-Irradiation Report of Fuel Materials for P13T Capsule Irradiation", U.S. Energy Research & Development Administration Report No. GA-A13343, General Atomic Company, San Diego, CA; April 1976.

5-20 Scheffel, W. J., "Post-Irradiation Examination of Capsule P13T", U.S. Department of Energy Report No. GA-A15608, General Atomic Company, San Diego, CA; August 1981.

5-21 Scheffel, W. J., "Irradiation Test Plan for Capsules P13U and P13V (P13UV-101)", GA Document No. 903803, General Atomic Company, San Diego, CA; Issue C, Jan. 17, 1979.

5-22 Sedlak, B. J. and Ketterer, J. W., "Capsule P13V Final Evaluation", GA Document No. 904305, General Atomic Company, San Diego, CA; Issue A, December 14, 1979.

5-23 Sterling, S. A., "HB-2 Pre-Irradiation Report", GA Document No. GA-A13774, General Atomic Company, San Diego, CA; December 1975.

5-24 Young, C. A., "Pre- and Post-Irradiation Evaluation of Fuel in Capsule HRB-14", U.S. Department of Energy Report No. GA-A15969, General Atomic Company, San Diego, CA; September 1980.

5-25 Ketterer, J. W. *et al*, "Capsule HRB-15A Post-Irradiation Examination Report", U.S. Department of Energy Report No. GA-A16758, GA Technologies, Inc., San Diego, CA; July 1984.

5-26 Kovacs, W. J. *et al*, "Post-Irradiation Evaluation of Capsules GF-1, GF-2, and GF-3", U.S. Energy Research & Development Administration Report No. GA-A15859, General Atomic Company, San Diego, CA; September 1980.

5-27 Wallroth, C. F., "Feasibility Study of the Dragon Reactor for HTGR Fuel Testing", U.S. Energy Research & Development Administration Report No. GA-A13333, General Atomic Company, San Diego, CA; March 3, 1975.

5-28 Young, C. A. *et al*, "Pre-Irradiation Report on Fuel Materials for the DR-GB2 Irradiation Experiment", GA Document No. GA-A13760, General Atomic Company, San Diego, CA; March 1, 1976.

5-29 Simnad, M. T. and West, G. B., "Post-Irradiation Examination and Evaluation of TRIGA LEU Fuel Irradiated in the Oak Ridge Research Reactor", GA Document No. GA-A18599, GA Technologies, Inc., San Diego, CA; May 1986.

6-1 "Nuclides and Isotopes; Chart of the Nuclides Fourteenth Edition", Nuclear Energy Operations, General Electric Company, San Jose, CA; 1989.

6-2 U.S. Code of Federal Regulations, 10 CFR 71, Appendix A, Table A-1: "A1 and A2 Values for Radionuclides"; Office of the Federal Register, Washington, DC.

6-3 Greenwood, J. S., Memo to Project Files, "Spray Paint Used for Identification of HTGR/RERTR FHUs", GA Internal Correspondence DDI:125:JSG:02, April 19, 2002.

6-4 DOE Directive, "Nuclear Materials Management and Safeguards System Reporting and Data Submission", DOE M 474.1-2, February 10, 1998.

7-1 Su, S. *et al*, "PATH Gamma Shielding Code User's Manual", GA Document No. GA-A16772, General Atomics, San Diego, CA; Rev. 1, May 1987