April 2008

Revision LWT-08C

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

Volume 1 of 2

Docket No. 71-9225

NAC INTERNATIONAL

Atlanta Corporate Headquarters: 3930 East Jones Bridge Road, Norcross, Georgia 30092 USA
Phone 770-447-1144, Fax 770-447-1797, www.nacintl.com

List of Effective Pages

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES (Continued)

LIST OF EFFECTIVE PAGES (Continued)

6.3.1-1 thru 6.3.1-5 .... Revision 38
6.3.2-1 thru 6.3.2-4 .... Revision 38
6.3.3-1 thru 6.3.3-8 .... Revision 38
6.3.4-1 thru 6.3.4-9 .... Revision 38
6.3.5-1 thru 6.3.5-12 ..... Revision LWT-07G
6.3.6-1 thru 6.3.6-9 .... Revision 38
6.3.7-1 thru 6.3.7-4 .... Revision 38
6.3.8-1 thru 6.3.8-7 .... Revision 38
6.3.9-1 thru 6.3.9-7 .... Revision 38
6.4.1-1 thru 6.4.1-10 .... Revision 38
6.4.2-1 thru 6.4.2-10 .... Revision 38
6.4.3-1 thru 6.4.3-33 .... Revision 38
6.4.4-1 thru 6.4.4-24 .... Revision 38
6.4.5-1 thru 6.4.5-32 ......Revision LWT-08B
6.4.6-1 thru 6.4.6-7 .... Revision LWT-07G
6.4.6-8 ....Revision LWT-08B
6.4.6-9 thru 6.4.6-23 ..... Revision LWT-07G
6.4.7-1 thru 6.4.7-14 .... Revision 38
6.4.8-1 thru 6.4.8-14 .... Revision 38
6.4.9-1 thru 6.4.9-10 .... Revision 38
6.4.10-1 thru 6.4.10-18 .... Revision 38
6.5.1-1 thru 6.5.1-13 .... Revision 38
6.5.2-1 thru 6.5.2-4 .... Revision 38
6.5.3-1 thru 6.5.3-2 .... Revision 38

Appendix 6.6

6.6-i thru 6.6-iii .... Revision LWT-07G
6.6-1 .... Revision 38
6.6.1-1 thru 6.6.1-111 .... Revision 38
6.6.2-1 thru 6.6.2-56 .... Revision 38
6.6.3-1 thru 6.6.3-73 .... Revision 38
6.6.4.-1 thru 6.6.4-77 .... Revision 38
6.6.5-1 thru 6.6.5-101 .... Revision 38
6.6.6-1 thru 6.6.6-158 ... Revision LWT-07G
6.6.7-1 thru 6.6.7-84 .... Revision 38
6.6.8-1 thru 6.6.8-189 ....Revision 38
6.6.9-1 thru 6.6.9-52 ....Revision 38
6.6.10-1 thru 6.6.10-33 ....Revision 38
6.6.11-1 thru 6.6.11-47 ....Revision 38
6.6.12-1 thru 6.6.12-20 ....Revision 38
6.6.13-1 thru 6.6.13-22 ....Revision 38
6.6.14-1 thru 6.6.14-7 ....Revision 38

Chapter 7

7-i thru 7-ii..................Revision LWT-07G
7.1-1 thru 7.1-11 .... Revision LWT-08B
7.1-12 thru 7.1-32.......Revision LWT-07G
7.1-33 thru 7.1-51 .... Revision LWT-08B
7.2-1 thru 7.2.1-12 ....Revision 38
7.3-1 thru 7.3-2 .... Revision LWT-08B

Chapter 8

8-i...Revision 38
8.1-1 thru 8.1-11 .... Revision LWT-08B
8.2-1 thru 8.2-4 .... Revision LWT-08B
8.3-1 thru 8.3-5 ....Revision 38

Chapter 9

9-i................................ Revision LWT-08B
9-1 thru 9-10 .... Revision LWT-08B

April 2008

Revision LWT-08C

NAC-LWT

Legal Weight Truck Cask System

SAFETY ANALYSIS REPORT

Volume 2 of 2

Docket No. 71-9225

NAC INTERNATIONAL

Atlanta Corporate Headquarters: 3930 East Jones Bridge Road, Norcross, Georgia 30092 USA
Phone 770-447-1144, Fax 770-447-1797, www.nacintl.com

List of Effective Pages

LIST OF EFFECTIVE PAGES

LIST OF EFFECTIVE PAGES (Continued)

LIST OF EFFECTIVE PAGES (Continued)

Chapter 5

Table of Contents

List of Figures

List of Figures (Continued)

List of Figures (Continued)

List of Tables

List of Tables (continued)

List of Tables (continued)

List of Tables (continued)

List of Tables (continued)

List of Tables (continued)

5.1 Discussion and Results

The NAC-LWT cask is designed for the safe transport of spent nuclear fuel from various commercial nuclear installations and research reactors.

5.1.1 NAC-LWT Contents

The following contents constitute the design basis for transport in the NAC-LWT cask:

- 1 PWR assembly;
- up to 2 BWR assemblies;
- up to 15 sound metallic fuel rods;
- up to 9 failed metallic fuel rods;
- up to 3 severely failed metallic fuel rods in filters;
- up to 42 MTR fuel elements;
- up to 42 DIDO fuel assemblies;
- up to 25 PWR fuel rods (including up to 14 rods classified as damaged);
- up to 25 BWR fuel rods (including up to 14 rods classified as damaged);
- up to 140 TRIGA fuel elements;
- up to 560 TRIGA fuel cluster rods;
- 2 GA IFM packages;
- up to 300 TPBARs (of which two can be prefailed);
- up to 55 TPBARs segmented during PIE, including segmentation debris;
- up to 700 PULSTAR fuel elements (intact or damaged);
- up to 42 spiral fuel assemblies;
- up to 42 MOATA plate bundles;
- any combination of individual ANSTO basket modules containing either spiral fuel assemblies or MOATA plate bundles up to a total of 42 assemblies/bundles; or
- up to 4,000 lbs of solid, irradiated and contaminated hardware.

The high burnup PWR and BWR rods may be transported in three configurations: 1) a maximum of 25 intact fuel rods loaded in the rod holder; 2) a maximum of 25 fuel rods with up to 14 damaged fuel rods or rod fragments loaded in the rod holder; and 3) a maximum of 25 intact fuel rods housed in a fuel assembly lattice within the NAC-LWT PWR basket. The fuel assembly lattice may be irradiated up to an equivalent burnup of 80,000 MWd/MTU.

The metallic fuel consists of a single rod of uranium metal clad with aluminum. The intact metallic fuel rods are placed into a transport canister that will hold five intact rods. The cask can hold three transport canisters for a total of 15 intact metallic fuel rods. In the event the metallic fuel has failed or is suspected of having failed, each fuel rod is sealed in its own container. The failed metallic fuel is loaded into either one of the three holes in the metallic fuel basket or into one of the six openings in the failed metallic fuel basket.

MTR research reactor fuel elements are typically 33 to 57 inches long, including lower nozzle and upper handle. The fuel plates typically consist of U-Al, U_3O_8-Al, or USi-Al clad with aluminum. The fuel plates are held in a parallel arrangement with two thick aluminum slotted pieces to form a fuel element. Standard fuel elements have between 10 and 23 fuel plates. The active fuel region is typically 22.75 inches in height, and the fuel meat is typically 0.023-inch thick. The highly enriched uranium (HEU) fuel has been analyzed conservatively with an enrichment of 90 wt % ^{235}U and fuel loading per element up to 380 g ^{235}U, with a separate analysis performed to accommodate up to 460 g ^{235}U. The design basis fuel parameters are provided in Table 5.1.1-1. The fuel characteristics are presented in Table 5.1.1-2. The dose rates produced from the design basis 470 g ^{235}U and 640 g ^{235}U LEU and 380 g ^{235}U MEU MTR fuel are bounded by the HEU MTR design basis fuel. Therefore, a mixed loading of LEU, MEU and HEU MTR fuel elements are also bounded by a full HEU MTR fuel element loading.

The source term characteristics of the design basis PWR fuel assembly, BWR fuel assembly, metallic rods, 25 PWR rods and MTR fuels are given in Table 5.1.1-3. The design basis PWR and BWR fuels require two years of cooling after discharge to meet the neutron and gamma source, and decay heat limits of the cask. The 25 design basis PWR rods burned to 60,000 MWd/MTU require 150 days of cooling. The design basis metallic fuel requires one year cooling. The design basis MTR fuel requires a variable number of years cooling, after discharge, to meet the decay heat limits of the cask. Loading configurations must conform to the limits stated in Section 7.1.5.

DIDO research reactor fuel elements typically consist of U-Al, U_3O_8-Al, or U_3Si_2-Al that is aluminum clad. The fuel elements are held in a concentric arrangement inside an outer aluminum cylinder to form a fuel assembly. Fuel assemblies have 4 fuel elements. The active fuel region is typically 23.6 inches in height, and the fuel meat is typically 0.026 inch thick. The highly enriched uranium (HEU) fuel has been analyzed with a minimum enrichment of 90 wt % ^{235}U and fuel loading per assembly up to 190 g ^{235}U. Low enriched (LEU) and medium enriched (MEU) assemblies are evaluated at 190 g ^{235}U with minimum enrichments of 19 and 40 wt % ^{235}U, respectively. The design basis fuel parameters are provided in Table 5.1.1-1. The fuel characteristics are presented in Table 5.1.1-2. As discussed in Section 5, the dose rates produced from the design basis LEU and MEU DIDO fuel are bounded by the HEU DIDO design basis fuel. Therefore, a mixed loading of LEU, MEU and HEU DIDO fuel assemblies is also bounded by a full HEU DIDO fuel assembly loading.

Two GA IFM Fuel Handling Units (packages) are intended for a single shipment in the NAC-LWT. The first package is composed of Reduced-Enrichment Research and Test Reactor (RERTR) type fuel, which is an Incoloy clad TRIGA fuel. The second is composed of High-Temperature Gas-Cooled Reactor (HTGR) type fuel. Each set of IFM is packaged into stainless steel weld-encapsulated primary and secondary enclosures. Design basis fuel parameters are summarized in Table 5.1.1-1, with fuel characteristics presented in Table 5.1.1-2. Design basis source terms are provided in Table 5.1.1-3. NAC-LWT combined dose rates for GA IFM are bounded by the dose rates for PWR fuel shown in Table 5.1.1-4 through Table 5.1.1-6.

An inventory of up to 300 production TPBARs (of which two can be prefailed) is intended for multiple shipments in the NAC-LWT. A separate content condition is for the transport of up to 55 segmented TPBARs and associated segmentation debris from PIE contained in a waste container. Each TPBAR is a Type 316 stainless steel rod with a 0.381-inch outer diameter and a 0.336-inch inner diameter and a post-irradiation length of approximately 154 inches. Tritium is produced by irradiation of ^{6}Li. Design basis fuel parameters are summarized in Table 5.1.1-1 with characteristics presented in Table 5.1.1-2. Design basis source terms are provided in Table 5.1.1-3. NAC-LWT dose rates for the payloads of up to 300 production TPBARs in a consolidation canister, or up to 55 segmented TPBARs in the waste container, are bounded by the dose rates for PWR fuel shown in Table 5.1.1-4 through Table 5.1.1-6.

Source terms for the high burnup PWR and BWR rods are developed using the SCALE SAS2H code package. Cask dose rates are evaluated using the SCALE SAS1 shielding analysis sequence. Results presented in Section 5.3.8 give the required cool time for PWR and BWR rods as a function of burnup for up to 25 intact fuel rods loaded in the NAC-LWT rod holder. The results presented in Sections 5.3.11 and 5.3.12 demonstrate that dose rate limits are met for the shipment of fuel rods in an irradiated fuel assembly lattice and damaged fuel rods in a rod holder, respectively.

As can be seen from Table 5.1.1-3, the PWR fuel assembly has the largest source terms and was used as the design basis fuel for shielding analysis of PWR and BWR fuel in the NAC-LWT presented in this section. The metallic fuel shielding analysis is presented in Section 5.3.3. Metallic fuel is shipped with the neutron shield drained and the analysis reflects this. The MTR fuel shielding analysis is presented in Section 5.3.4. The design basis source terms for 25 PWR rods at 60,000 MWd/MTU are well below the design basis PWR fuel assembly. However, the self shielding of 25 PWR rods is less than the 204 rod design basis PWR fuel assembly. Thus, a shielding evaluation of 25 design basis PWR rods is presented in Section 5.3.5. Similarly, the self shielding for either the 25 high burnup PWR or BWR rods at 80,000 MWd/MTU is lower than that of the design basis assemblies. Shielding evaluations for these rods are presented in Sections 5.3.8, 5.3.11 and 5.3.12.

The transport of up to 140 TRIGA fuel elements is evaluated in Section 5.3.6. TRIGA fuel is a solid metal hydride, U-ZrH and may be high enriched (70%), or low enriched (20%). The fuel clad is either aluminum or stainless steel. TRIGA fuel is fabricated in several configurations, as described in Section 1.2.3.1, that vary in weight, active fuel length and overall length. The typical fuel element length and weight is 28.3 inches and 8.82 pounds. The fuel follower control rod element (FFCR) establishes the upper bound weight (13.2 pounds) and length (approximately 45 inches). These elements can only be loaded in the top module of the TRIGA fuel basket. The design basis TRIGA fuel parameters are presented in Table 5.1.1-1 and Table 5.1.1-2. Source term characteristics are presented in Table 5.1.1-3. Cooling time for TRIGA fuel is variable, down to a minimum of 90 days, based on the time required for the decay heat to reach 7.5 watts.

In addition, the transport of TRIGA fuel cluster rods is evaluated in Section 5.3.7. These rods are obtained from the disassembly of the 5×5 (25 rod) arrays comprising the cluster-type TRIGA fuel as shown in Figure 1.2-6. Only the shipment of the fuel cluster rods is analyzed here; no other activated components of the TRIGA cluster assembly are considered for shipment in this analysis. The TRIGA fuel cluster rod is considered to contain a maximum design-basis fuel mass of 50.5 g of U (evaluated at 92 wt % ^{235}U) for HEU cluster rods and 289.4 g of U (19 wt % ^{235}U) for LEU elements. Both elements are modeled with a nominal H to Zr ratio of 1.6. A manufacturing tolerance produced H to Zr ratio of 1.7 is evaluated in Chapter 6 for criticality. The manufacturing tolerances have no significant effect on the shielding evaluations. The HEU fuel contains 10 wt % uranium in the U-ZrH_x fuel meat, while the LEU material contains 45 wt % uranium. The rods are clad in Incoloy 800 and contain upper and lower stainless steel end plugs with a mass of approximately 60.5 g each. For shipment, each rod is placed inside an aluminum tube (ID 0.625 in, OD 0.750 in), with 16 rods occupying each LWT basket opening for a total of up to 112 rods per basket or 560 rods per cask.

The basis for the dose rate evaluation of the TRIGA fuel cluster rods is a source term and one-dimensional shielding analysis in which the minimum cooling time required for the dose rates produced by the TRIGA fuel cluster rods to fall below the dose rates produced by the design basis TRIGA fuel elements. Cooling time results are determined at a large number of fuel burnup values (at approximately every 2.5% increment in ^{235}U depletion).

PULSTAR fuel elements are zirconium alloy-clad UO_2 pellets with a physical design characteristic as listed in Table 5.1.1-1 and Table 5.1.1-2. PULSTAR fuel assemblies are a 5×5 rectangular array of elements surrounded by a zirconium alloy box, with aluminum upper and lower fittings. The element pitch is nominally 0.524 × 0.606 inch. PULSTAR fuel elements are analyzed at a loading of 32 grams ^{235}U per element, an initial enrichment of 6 wt % ^{235}U, and 45% ^{235}U burnup. For conservatism, a cool time of one year from discharge is employed in the shielding analysis; a cool time of at least 1.5 years is required to meet the basket cell heat load limit of 30 W. Source term characteristics are presented in Table 5.1.1-3

Spiral fuel assemblies typically consist of 10 curved plates (also referred to as elements) of metallic U-Al fuel meat that is aluminum clad. The fuel elements are held in a spiral arrangement between an inner and outer aluminum cylinder to form a fuel assembly. The active fuel region is typically 60.325 cm in height, and the fuel meat is typically 0.061 cm thick. The elements are nominally enriched to 80 wt % ^{235}U and were conservatively evaluated at 75 wt % ^{235}U. Maximum fuel loading per assembly is evaluated at 160 g ^{235}U. The design basis fuel parameters are provided in Table 5.1.1-1. The fuel characteristics are presented in Table 5.1.1-2. Applying MEU DIDO fuel assembly minimum cool time curves, which are based on a 40 wt % ^{235}U enriched 190 g ^{235}U fuel assembly, to the spiral fuel elements produces source terms that are bounded by the DIDO MEU fuel. Given similar basket designs, the dose rates produced by the spiral fuel elements are bounded by the MEU DIDO evaluation set.

MOATA fuel bundles consist of a maximum of 14 flat MTR type fuel plates. The fuel plates are composed of a metallic U-Al fuel meat that is aluminum clad. The fuel elements are held in place with aluminum outer plates and pins through the top and bottom of the plate stack in their shipment configuration. The plates are held in a typical MTR plate (12 plates per assembly) with a comb side plate configuration during reactor operations. The active fuel region is typically 58.4 cm in height, and the fuel meat is typically 0.1016-cm thick. The elements are nominally enriched to 90 wt % ^{235}U and were conservatively evaluated at 80 wt % ^{235}U. Maximum fuel loading per plate is evaluated at 25 g ^{235}U (nominal loading is 22 g ^{235}U). The design basis fuel parameters are provided in Table 5.1.1-1. The fuel characteristics are presented in Table 5.1.1-2. The gamma radiation source for the 14 fuel plate bundle is approximately 2% of the DIDO MEU assembly. Since the basket designs are similar, the dose rates produced by the plate bundle are bounded by the MEU DIDO evaluation set.

The shield materials are selected and arranged to minimize cask weight while maintaining overall shield effectiveness. Lead and steel are chosen as effective gamma radiation shields, and a water tank on the outside of the cask is provided to efficiently moderate and absorb the neutron radiation.

The total neutron and gamma dose rates calculated for the normal operations conditions are shown in Table 5.1.1-4. Note that the maximum dose rate is on the cask lid surfaces at the top end of the cask and does not exceed the design limit of 200 mrem/hour for the surface of the cask. The 10 CFR 71 limits of 10 mrem/hour at two meters from the cask surface and the design.limit of 200 mrem/hour on the cask surface are met. Table 5.1.1-4 contains the total dose rates for the hypothetical accident conditions. These dose rates are well under the 49 CFR 173 limit of 1000 mrem/hour at one meter from the cask surface. The dose rates for the lead slump accident are shown in Table 5.1.1-5. These dose rates show that even with the lead slumped, the hypothetical accident dose rate limits have not been exceeded and the cask is safe for transport.

The cask surface fuel centerline normal operations and hypothetical accident dose rates calculated include neutrons and gammas originating from the fuel, neutrons and gammas scattered from the ground and secondary gammas resulting from neutron capture in the neutron shield. All of the other dose locations also include the contribution from the ^{60}Co in the end-fittings.

Table 5.1.1-1 Type, Form, Quantity and Potential Sources of Design Basis Fuel

Fuel Type	-	PWR, Assembly
	-	3.7 wt % ^{235}U maximum initial enrichment
	-	35,000 MWd/MTU maximum burnup
	-	2.5 kW per assembly maximum decay heat
	-	2 years (or more) decay time after reactor discharge
Fuel Form	-	Intact assemblies
Quantity	-	1 design basis fuel assembly
Source of Fuel	-	Commercial PWR nuclear power reactors
Transport Index	-	35
Fuel Type	-	BWR, Assembly
	-	4.0 wt % ^{235}U maximum initial enrichment
	-	30,000 MWd/MTU maximum burnup
	-	1.1 kW per assembly maximum decay heat, 2.2 kW per cask for 2 assemblies
	-	2 years (or more) decay time after reactor discharge
Fuel Form	-	Intact assemblies
Quantity	-	2 design basis fuel assemblies
Source of Fuel	-	Commercial BWR nuclear power reactors
Transport Index	-	35
Fuel Type		High Burnup PWR or BWR rods
	-	5.0 wt % maximum ^{235}U initial enrichment
	-	80,000 MWd/MTU maximum average burnup
	-	2.3 kW /cask maximum decay heat
	-	Minimum cool time dependent on burnup (See Table 5.3.8-29)
Fuel Form	-	Intact rods in a fuel assembly lattice or rod holder and intact rods with up to 14 fuel rods classified as damaged in a rod holder
Quantity	-	Up to 25
Source of Fuel	-	Commercial PWR or BWR nuclear power reactor
Transport Index	-	36 (intact rods) 28 (intact rods in a fuel assembly lattice) 37 (intact rods with 14 rods classified as damaged)
Fuel Type	-	Uranium metal fuel rods
	-	Natural wt % ^{235}U
	-	1,600 MWd/MTU maximum burnup
	-	0.0357 kW per sound rod maximum decay heat, 0.54 kW per cask for 15 sound fuel rods
	-	1 year (or more) decay time after reactor discharge
Fuel Form	-	Intact or encapsulated failed fuel rods
Quantity	-	15 design basis fuel rods, or 6 design basis failed fuel rods
Source of Fuel	-	Research reactors
Transport Index	-	25

Table 5.1.1-1 Type, Form, Quantity and Potential Sources of Design Basis Fuel (cont'd)

Fuel Type	-	Material Test Reactor (MTR) Fuel Elements
	-	HEU: 90 wt % ^{235}U, Maximum burnup variable up to 660,000 MWd/MTU for 380 g ^{235}U and 577,500 MWd/MTU for 460 g ^{235}U
	-	MEU: 40 wt % ^{235}U, Maximum burnup variable up to 293,300 MWd/MTU for 380 g ^{235}U
	-	LEU: 19 wt % ^{235}U, Maximum burnup variable up to 139,300 MWd/MTU for 470 g ^{235}U and 640 g ^{235}U
	-	210 W per basket decay heat
	-	Variable cool time down to 90 days using the procedure in Section 7.1.5
Fuel Form	-	Intact aluminum clad parallel plates
Quantity	-	Up to 42 fuel elements
Source of Fuel	-	Research and Material Test Reactors
Transport Index	-	45
Fuel Type	-	TRIGA Fuel Element
	-	20 to 70 wt % ^{235}U
	-	80% ^{235}U depletion (approximately 151 GWd/MTU for LEU fuel and 460 GWd/MTU HEU fuel)
	-	7.5 watts per element decay heat
	-	Variable cool time down to 90 days
Fuel Form	-	Aluminum or stainless steel (304) clad rods, intact, failed or as debris
Quantity	-	Up to 140 fuel elements
Source of Fuel	-	Test, Research and Isotope Reactors
Transport Index	-	25
Fuel Type	-	HEU and LEU TRIGA Fuel Cluster Rods
	-	Minimum 92 wt % ^{235}U (HEU) and minimum 19 wt % ^{235}U (LEU)
	-	80% ^{235}U depletion (approximately 600 GWd/MTU for HEU and approximately 140 GWd/MTU for LEU)
	-	1.875 watts per rod decay heat
	-	Variable cool time down to 90 days
Fuel Form	-	Incoloy 800 clad damaged or undamaged rods
Quantity	-	Up to 560 fuel rods
Source of Fuel	-	Test, Research and Isotope Reactors
Transport Index	-	17.9
Fuel Type	-	DIDO Fuel Assemblies
	-	HEU: 90 wt % ^{235}U, Maximum burnup variable up to 577,460 MWd/MTU or 70% ^{235}U depletion
	-	MEU: 40 wt % ^{235}U, Maximum burnup variable up to 256,650 MWd/MTU or 70% ^{235}U depletion
	-	LEU: 19 wt % ^{235}U, Maximum burnup variable up to 121,910 MWd/MTU or 70% ^{235}U depletion

Table 5.1.1-1 Type, Form, Quantity and Potential Sources of Design Basis Fuel (cont'd)

	-	175 or 126 W per basket decay heat
	-	Variable cool time down to 180 days using the procedure in Section 7.1.4
Fuel Form	-	Intact aluminum clad concentric fuel tubes
Quantity	-	Up to 42 fuel assemblies
Source of Fuel	-	Research Reactors
Transport Index	-	40.1
Fuel Type	-	General Atomics (GA) Irradiated Fuel Material (IFM)
	-	RERTR (see activity inventory in Table 5.3.10-1)
	-	HTGR (see activity inventory in Table 5.3.10-1)
	-	<13.05 W
	-	Transport after 1/1/96
Fuel Form	-	RERTR: 13 intact TRIGA elements, 7 sectioned elements
	-	HTGR: Spherical loose fuel particles, cylindrical fuel rods, 2 fuel pebbles
Quantity	-	1 Fuel Handling Unit holding RERTR IFM and 1 Fuel Handling Unit holding HTGR IFM
Source of Fuel	-	Research reactors, commercial LWR reactors
Transport Index	-	<1
Maximum Activity	-	3,403 Ci
Material Type	-	Tritium Producing Burnable Absorber Rods (TPBARs)
	-	3.35 W/TPBARs; 1.005 kW total[1]
	-	30 days minimum cool time
Material Form	-	Type 316 stainless steel clad TPBARs
Quantity	-	Up to 300 TPBARs (of which two can be prefailed)
Source of Material	-	Commercial LWR reactors
Transport Index	-	22
Maximum Activity	-	12,800 Ci/TPBAR; 3,840,000 Ci total[1]
Material Type	-	Tritium Producing Burnable Poison Rods (TPBARs)
	-	2.31 W/TPBAR, 127 W total
	-	90 days
Material Form	-	Type 316 stainless steel clad TPBARs segmented for PIE
Quantity	-	Up to 55 segmented TPBARs
Source of Material	-	Commercial LWRs
Transport Index	-	22[2]
Maximum Activity	-	12,000 Ci/TPBAR, 665,500 Ci total

[1] Conservatively calculated for 30-day minimum cooling time. Actual minimum cooling period for thermal requirements is 90 days.
[2] Conservatively applied 300 TPBAR shipment transport index.

Table 5.1.1-1 Type, Form, Quantity and Potential Sources of Design Basis Fuel (cont'd)

Fuel Type	-	PULSTAR Fuel Elements
	-	6 wt % ^{235}U
	-	32 grams ^{235}U per element
	-	45% ^{235}U depletion (burnup)
	-	210 W per basket decay heat (30 watts per basket cell) × 4 = 840W Minimum cool time from discharge of 1.5 years[3]
Fuel Form	-	Intact assemblies; intact elements in fuel rod insert; canned intact or failed elements
Quantity	-	Up to 700 elements (25 elements per cell)
Sources of Fuel	-	Research reactors
Transport Index	-	25

Fuel Type	-	Spiral Fuel Assemblies
	-	75 wt % ^{235}U, maximum burnup variable up to 70% ^{235}U depletion
	-	18 W per assembly , 126 W per basket (at given cool time and burnup limits, maximum heat load is 15.7 W per assembly or 110 W per basket) Variable cool time down to 270 days using the procedure in Section 7.1.4 for 18 W DIDO MEU fuel
Fuel Form	-	Intact aluminum clad fuel plates within concentric aluminum inner and outer shells
Quantity	-	Up to 42 fuel assemblies
Sources of Fuel	-	Research reactors
Transport Index	-	40.1 (applied bounding MEU DIDO limit)

Fuel Type	-	MOATA Plate Bundles
	-	80 wt % ^{235}U, maximum burnup variable up to a 30,000 MWd/MTU or 4.1% ^{235}U depletion
Fuel Form	-	Intact aluminum-clad fuel plates
Quantity	-	Up to 42 bundles
Sources of Fuel	-	Research reactors
Transport Index	-	40.1 (applied bounding MEU DIDO limit)

3 Conservatively evaluated at a one-year cool time and 38 watts per basket cell.

Table 5.1.1-2 Design Basis Fuel for Shielding Evaluation

Parameter	PWR	BWR	Metallic	MTR (HEU)	MTR (MEU)	MTR (LEU)	DIDO
Assembly Array	15 × 15	7 × 7	N/A	Parallel Plates	Parallel Plates	Parallel Plates	Fuel Tubes
Assembly or Element Weight (lbs)	1650	750	1805 (15 rods)	13.0 (max)	13.0 (max)	13.0 (max)	15.0 (max)
Assembly/Element/Rod Length (in)	162	176	120.5	25.23[5]	26.14[5]	26.14[5]	24.6
Active Fuel Length (in)	144	144	120.0	24.80	25.59	25.59	23.6
No. Rods per Assembly	204	49	N/A	N/A	N/A	N/A	N/A
No. of Plates per Element	N/A	N/A	N/A	23	23	23	4
Fuel Rod Diameter/Plate Thickness (in)	0.422	0.563	1.36	0.050	0.050	0.050	0.059
Clad Material	Zr-4	Zr-4	Al	Al	Al	Al	Al
Clad Thickness (in)	0.0243	0.032	0.080	0.0150	0.0150	0.0150	0.0167
Pellet Diameter/Meat Thickness (in)	0.3659	0.487	1.36	0.020	0.020	0.020	0.026
Fuel Material	UO_2	UO_2	U metal	U_3O_8-Al; U-Al; or U_3Si_2-Al	U_3O_8-Al; U-Al; or U_3Si_2-Al	U_3O_8-Al; U-Al; or U_3Si_2-Al	U_3O_8-Al; U-Al; or U_3Si_2-Al
Percent Theoretical Density	95	95	100	N/A	N/A	N/A	N/A
Enrichment (wt % ^{235}U)	3.7	4.0	Natural	90[8]	40[8]	19[8]	90 (HEU) 400 (MEU) 199 (LEU)
Maximum Average Burnup (MWd/MTU)	35,000	30,000	1,600	Variable up to 660,000[2,9]	Variable up to 293,300[2]	Variable up to 139,300[2]	Variable up to 577,460 (HEU) 256,650 (MEU) 121,910 (LEU)
Minimum Cool Time	2 Years	2 Years	1 Year	Variable down to 90 days[2]	Variable down to 90 days[2]	Variable down to 90 days[2]	Variable down to 180 days[10]
U Weight (kg/assembly)	475	198	N/A	N/A	N/A	N/A	N/A
U Weight (kg/element)	N/A	N/A	54.5	0.422 0.511	0.950	2.4737 3.3684	0.2111 (HEU) 0.4750 (MEU) 1.0000 (LEU)
UO_2 Weight (kg/assembly)	538.9	224.3	N/A	N/A	N/A	N/A	N/A

Notes:

1. Up to 2 of the PWR rods may have a maximum average burnup of 65,000 MWd/MTU.
2. Variable cool time down to 90 days using the procedure in Section 7.1.4.
3. Design Basis normal condition source term is for ACPR fuel with 86,100 MWd/MTU (50% ^{235}U depletion) and accident condition source term is for FLIP-LEU-II with 151,100 MWd/MTU (80% ^{235}U depletion).
4. Detailed fuel data is presented in Tables 1.2-1 and 6.2.5-1. The values presented here are the physical values for the bounding source terms of the ACPR and FLIP-LEU-II fuel types.
5. For MTR fuel assemblies, which are cut to remove non-fuel bearing hardware prior to transport, a nominal 0.28 inch of non-fuel hardware will remain above and below the active fuel region to allow for fuel handling operations
6. Minimum cool time varies with burnup such that maximum decay heat is 1.875 watts/rod.
7. Varies with burnup – see Table 5.3.8-29.
8. For the shielding evaluation, lower values are conservatively assumed.
9. Maximum burnup of 660,000 MWd/MTU for 380 g ^{235}U and 577,500 MWd/MTU for 460 g ^{235}U.
10. Variable cool time down to 180 days using the procedure in Section 7.1.4.

Table 5.1.1-2 Design Basis Fuel for Shielding Evaluation (continued)

Parameter	PWR Rods	High B/U PWR Rods	High B/U BWR Rods	TRIGA[4]	TRIGA Fuel Cluster Rods	TPBARs
Assembly Array	N/A	N/A	N/A	N/A	N/A	N/A
Assembly or Element Weight (lbs)	N/A	N/A	N/A	8.82 (nominal) 13.2 (max)		2.655
Assembly/Element/Rod Length (in)	162	162	176.1	45	31.0	153.035 (pre-irradiation)
Active Fuel Length (in)	144	150	150	15	22	N/A
No. Rods per Assembly per Shipment	25	25	25	1	1	300 Production or 55 Segmented
No. of Plates per Element	N/A	N/A	N/A	N/A	N/A	N/A
Fuel Rod Diameter/Plate Thickness (in)	0.422	0.440	0.570 (7×7) 0.4961 (other)	1.478	0.542	0.381
Clad Material	Zr-4	Zr-4	Zr-2	304SS	Incoloy 800	316 SS
Clad Thickness (in)	0.242	0.026	0.036 (7×7) 0.0343 (other)	0.02	0.016	0.0225
Pellet Diameter/Meat Thickness (in)	0.3659	0.3805	0.4900 (7×7) 0.4213 (other)	1.435 (max)	0.510	N/A
Fuel Material	UO2	UO2	UO2	U-ZrH	U-ZrH	N/A
Percent Theoretical Density	97	95	95	95	95	N/A
Enrichment (wt % ^{235}U)	5.0	5.0	5.0	20	92 (HEU) 19 (LEU)	N/A
Maximum Average Burnup (MWd/MTU)	60,000[1]	80,000	60,000 – 80,000	ACPR 86,100 (50% ^{235}U)[3] FLIP-LEU-II 151,100 (80% ^{235}U)[3]	Variable up to 600,000 (HEU) Variable up to 140,000 (LEU)	N/A
Minimum Cool Time	150 (days)	150 days	Varies with burnup[7]	ACPR 231 days FLIP-LEU-II 908 days	Varies with burnup[6]	30 days for production TPBAR; 90 days for PIE TPBAR
U Weight (kg/assembly)	58.2	65.6	108.8 (7×7) 91.3 (other)	N/A	N/A	N/A
U Weight (kg/element)	N/A	N/A	N/A	ACPR 0.280 FLIP-LEU-II 0.824	0.0505 (HEU) 0.2894 (LEU)	N/A
UO_2 Weight (kg/assembly)	66.0	66.0	74.5	N/A	N/A	N/A

Notes:
1. Up to 2 of the PWR rods may have a maximum average burnup of 65,000 MWd/MTU.
2. Variable cool time down to 90 days using the procedure in Section 7.1.4.
3. Design Basis normal condition source term is for ACPR fuel with 86,100 MWd/MTU (50% ^{235}U depletion) and accident condition source term is for FLIP-LEU-II with 151,100 MWd/MTU (80% ^{235}U depletion).
4. Detailed fuel data is presented in Tables 1.2-1 and 6.2.5-1. The values presented here are the physical values for the bounding source terms of the ACPR and FLIP-LEU-II fuel types.
5. For MTR fuel assemblies, which are cut to remove nonfuel-bearing hardware prior to transport, a nominal 0.28 inch of non-fuel hardware will remain above and below the active fuel region to allow for fuel handling operations
6. Minimum cool time varies with burnup such that maximum decay heat is 1.875 watts/rod.
7. Varies with burnup – see Table 5.3.8-29.
8. For the shielding evaluation, lower values are conservatively assumed.
9. Maximum burnup of 660,000 MWd/MTU for 380 g ^{235}U and 577,500 MWd/MTU for 460 g ^{235}U.
10. Variable cool time down to 180 days using the procedure in Section 7.1.4.

Table 5.1.1-2 Design Basis Fuel for Shielding Evaluation (continued)

Parameter	GA IFM RERTR	GA IFM HTGR	PULSTAR Fuel	Spiral Fuel Assembly	MOATA Plate Bundle
Assembly Array	N/A	N/A	5×5	Spiral Plates	Parallel Plates
Assembly or Element Weight (lbs)	23.73	23.52	45 (assembly); 1.3 (element)	7.9	13.6[11]
Assembly/Element/Rod Length (in)	29.92	N/A	38 (assembly) 26.2 (element)	63.5 cm	58.4 cm[12]
Active Fuel Length (in)	22.05	N/A	24.1	60.325 cm	58.4 cm
No. Rods per Assembly	13 intact; 7 sectioned	N/A	25	N/A	N/A
No. of Plates per Element	N/A	N/A	N/A	10	maximum 14
Fuel Rod Diameter/Plate Thickness (in)	0.543	N/A	0.47	0.147 cm	0.203 cm
Clad Material	Incoloy	N/A	Zirconium alloy	Al	Al
Clad Thickness (in)	0.031	N/A	0.0185	0.043 cm	N/A
Pellet Diameter/Meat Thickness (in)	0.512	N/A	0.423	0.061 cm	0.1016 cm
Fuel Material	U-ZrH	UC_2; UCO; UO_2; $(Th,U)C_2$; or $(Th,U)O_2$	UO_2	U-Al	U-Al
Percent Theoretical Density	N/A	N/A	94.9% (nominal); 99.5% (analyzed)	N/A	N/A
Enrichment (wt % ^{235}U)	19.7	93.15 (maximum)	6	75	80
Maximum Average Burnup (MWd/MTU)	N/A	N/A	45	70% ^{235}U depletion	30,000 MWd/MTU 4.1% ^{235}U depletion
Minimum Cool Time	None	None	1.0 Year	see MEU DIDO	10 yr
U Weight (kg/assembly)	8.49	0.45	13.33	0.213[13]	0.4375[14]
U Weight (kg/element)	0.42	N/A	0.53	0.0213[15]	0.03125[16]
UO_2 Weight (kg/assembly)	N/A	N/A	15.13	N/A	N/A

Notes: (cont'd)

11. For 14-fuel plate bundle.
12. Not available for in-core configuration. Analysis input restricted to active fuel length.
13. Based on a 160 g ^{235}U fissile material load and listed enrichment.
14. Based on fuel mass per plate multiplied by 14 plates.
15. Based on 10 plates per assembly.
16. Based on 25 g ^{235}U and listed enrichment.

Table 5.1.1-3 Nuclear and Thermal Source Parameters

Payload	Decay Heat (kW)	Gamma Source (g/sec)	Neutron Source (n/sec)	Top End-Fitting (g/sec)	Bottom End-Fitting (g/sec)
1 PWR Assembly	2.5	1.27E+16	2.21E+08	1.49E+13	1.25E+13
2 BWR Assemblies	2.2	1.04E+16	1.34E+08	1.16E+12	2.78E+12
15 Sound Metallic Fuel Rods[2]	0.532	4.37E+15	1.61E+05	N/A	N/A
6 Failed Metallic Fuel Rods[1]	0.03	1.75E+15	6.44E+04	N/A	N/A
42 HEU MTR Elements[3,9]	1.26	7.42E+15	1.40E+08	N/A[15]	N/A[15]
42 MEU MTR Elements[3,8]	1.26	7.86E+15	2.88E+07	N/A[15]	N/A[15]
42 LEU MTR Elements[3,8,14]	1.26	7.51E+15	3.96E+07	N/A[15]	N/A[15]
42 DIDO Assemblies[10]	1.05	6.07E+15	9.73E+04	N/A	N/A
25 PWR Rods[2]	1.41	8.39E+15	1.40E+08	N/A	N/A
TRIGA (140 Elements) Normal Condition	1.05	6.52E+15[4]	1.57E+06	Note 6	Note 6
TRIGA (140 Elements) Accident Condition	1.05	5.97E+15[5]	1.06E+08	Note 6	Note 6
HEU TRIGA Cluster Rod[7]	1.875E-03	1.12E+13	4.918E+01	N/A	N/A
LEU TRIGA Cluster Rod[7]	1.875E-03	1.11E+13	4.005E+02	N/A	N/A
General Atomics Irradiated Fuel Material	0.013	3.429E+13	1.279E+04	Note 11	Note 11
300 Production TPBARs	1.005	6.681E+15	N/A	N/A	N/A
55 PIE TPBARs	1,005	5.6E+13	N/A	N/A	N/A
PULSTAR Fuel	1.05[12]	6.206E+15	2.115E+07	N/A	N/A
Spiral Fuel Assembly[13]	0.756	1.07E +14	4.54E+03	N/A	N/A
MOATA Plate Bundle	0.042	2.2E +12	< 1E+03	N/A	N/A

Notes:

1. Gamma and neutron source terms conservatively calculated based on design basis sound metallic fuel rods.
2. 23 rods with 60,000 MWd/MTU burnup and two rods with 65,000 MWd/MTU burnup. Source terms as a function of cool time for the 80,000 MWd/MTU burnup PWR and BWR rods are presented in Section 5.3.8.
3. Bounding values of the gamma and neutron source terms presented for 30W uniform loading for 80% burnup.
4. Based on TRIGA ACPR fuel (86,100 MWd/MTU, 231 days cooling, 50% ^{235}U depletion).
5. Based on TRIGA FLIP-LEU-II fuel (151,100 MWd/MTU, 908 days cooling, 80% ^{235}U depletion).
6. Total hardware gamma is 7.64E+14 gamma/second for ACPR fuel (86,100 MWd/MTU, 231 days cooling, 50% ^{235}U depletion).
7. Source term at TRIGA cluster rods maximum dose rate burnup/cool time combination. For HEU fuel, 150 GWd/MTU, 1.34 years cooled. For LEU fuel, 30 GWd/MTU, 1.5 years cooled. Gamma source includes source from activated inconel clad.
8. Moderator used is light water, H_20.
9. Moderator used is heavy water, D_20.
10. Bounding values of the gamma and neutron source terms presented for 25W uniform loading for 70% burnup HEU fuel.
11. Hardware activation, including end-fitting sources, for the TRIGA elements included in the total gamma source for GA IFM.
12. Cool time required to meet 30 watt per cell heat load limit is 1.5 years.
13. Based on 18 W per assembly heat load.
14. Fuel source represents maximum magnitude gamma source obtained from the 470 g ^{235}U analysis, and the maximum neutron source obtained from the 640 g ^{235}U analysis.
15. A maximum 100 grams of cadmium may be included as part of the MTR fuel element or plate construction. Activation of the cadmium produces no significant source per Section 5.3.4.

Table 5.1.1-4 Combined Dose Rates for Normal Operations Conditions

(1 PWR assembly, 35,000 MWd/MTU, 2-year cool time)

Location	Detector I.D.	Radiation	Normal Dose Rate (mrem/hr)
Radial at 2 m from personnel barrier, Fuel midplane	1	Neutron	1.25
		Secondary Gamma	0.18
		Primary Gamma	6.71
		TOTAL	8.14
Radial surface, Fuel midplane	2	Neutron	6.53
		Secondary Gamma	1.37
		Primary Gamma	43.44
		TOTAL	51.34
Bottom surface, Axial centerline	3	Neutron	0.33
		Primary Gamma	35.51
		End-fitting Gamma	17.02
		TOTAL	52.86
Bottom at 2 m from impact limiter, Axial centerline	4	Neutron	0.03
		Primary Gamma	2.19
		End-fitting Gamma	0.79
		TOTAL	3.01
Top surface, Axial centerline	5	Neutron	0.12
		Primary Gamma	54.17
		End-fitting Gamma	41.45
		TOTAL	95.74
Top at 2 m from impact limiter, Axial centerline	6	Neutron	0.01
		Primary Gamma	3.82
		End-fitting Gamma	2.17
		TOTAL	6.00
Top at Cab	7	Neutron	0.00135
		Primary Gamma	0.47
		End-fitting Gamma	0.25
		TOTAL	0.72

Table 5.1.1-5 Hypothetical Accident – Loss of Shielding Materials

(1 PWR assembly, 35,000 MWd/MTU, 2-year cool time)

Location	Detector I.D.	Radiation	Normal Dose Rate (mrem/hr)
Radial surface, Fuel midplane, With neutron shield	8	Neutron	6.53
		Secondary Gamma	1.37
		Primary Gamma	43.44
		TOTAL	51.34
Radial surface, Fuel midplane, Without neutron shield	9	Neutron	177.13
		Secondary Gamma	0.39
		Primary Gamma	75.00
		TOTAL	252.52
Radial at 1 m from surface, Fuel midplane, Without neutron shield	10	Neutron	50.93
		Secondary Gamma	1.52
		Primary Gamma	54.59
		TOTAL	107.04

Table 5.1.1-6 Hypothetical Accident –Lead Slump

Location	Detector I.D.	Radiation	Normal Dose Rate (mrem/hr)
Radial at 1 m from surface, PWR top end-fitting	11	End-fitting Gamma	3.60
		TOTAL	3.60
Radial at 1 m from surface, PWR top end-fitting	12	End-fitting Gamma	1.31
		TOTAL	1.31
Radial at 1 m from surface, PWR top end-fitting	13	End-fitting Gamma	0.80
		TOTAL	0.80
Radial at 1 m from surface, PWR bottom end-fitting	14	End-fitting Gamma	0.01
		TOTAL	0.01
Radial at 1 m from surface, PWR bottom end-fitting	15	End-fitting Gamma	0.35
		TOTAL	0.35
Radial at 1 m from surface, PWR bottom end-fitting	16	End-fitting Gamma	1.48
		TOTAL	1.48
Radial at 1 m from surface, BWR bottom end-fitting	17	End-fitting Gamma	0.10
		TOTAL	0.10
Radial at 1 m from surface, BWR bottom end-fitting	18	End-fitting Gamma	0.54
		TOTAL	0.54
Radial at 1 m from surface, BWR bottom end-fitting	19	End-fitting Gamma	0.84
		TOTAL	0.84

5.3.7 TRIGA Fuel Cluster Rod Model Specification and Shielding Evaluation

TRIGA fuel cluster rods are shown to comply with regulatory dose rate limits by determining the cool time required for the fuel to fall below the design basis TRIGA element-type dose rates. Source terms for the fuel cluster rods are determined using the SCALE SAS2H (Herman) code at various fuel burnup values. For each burnup considered, the resulting set of cool time and decay heat values is interpolated to find the cool time required for the fuel to meet the maximum allowed heat load (1.875W). A final ORIGEN-S calculation is performed at this cool time in order to compute the source spectra at the required decay time.

For consistency, the one-dimensional SCALE SAS1 dose rates computed for the fuel cluster rods are compared with one-dimensional dose rates for the design basis TRIGA fuel elements. Both normal condition and accident condition dose rates are considered, and the maximum cool time required to meet both limits is reported. Further, since the TRIGA fuel cluster rods have approximately the same end-fitting mass as the element-type fuel, the comparison is made on the basis of fuel radiation sources alone.

The maximum computed one-dimensional dose rate at 2 meters from the conveyance for the normal condition TRIGA fuel element analysis is 4.5 mrem/hr. The more accurate three-dimensional analysis predicts a dose rate at the same location of 3.18 mrem/hr. Hence, the one-dimensional analysis provides a conservative estimate of computed dose rates. Similarly, in the loss of neutron shielding hypothetical accident scenario, the maximum computed one-dimensional dose rate for the design basis fuel is 35 mrem/hr and the three-dimensional result is 28.07 mrem/hr.

Section 5.3.7.1 presents the SAS2H source term model for the TRIGA fuel cluster rods. Section 5.3.7.2 discusses the methodology used to compute one-dimensional dose rates for each burnup and cool time case. The resulting required cool times for the fuel cluster rods are presented in Section 5.3.7.3.

5.3.7.1 TRIGA Fuel Cluster Rod Source Terms

The SAS2H description of the TRIGA fuel cluster rods is based on the material and dimensional parameters given in Table 5.3.7-1. The irradiation parameters are based on a nominal 14 MW TRIGA reactor operating with 29 cluster-type assemblies consisting of 25 rods each. The SAS2H models use a fuel and clad temperature of 517K, and a moderator temperature of 363K (unpressurized nonboiling reactor), at a density of 0.981 g/cm^3. For each burnup case, the required exposure time is computed at this fixed power level. This conservatively assumes that all fuel irradiation occurs during the period immediately prior to fuel discharge. Representative gamma and neutron source terms are summarized in Table 5.3.7-3 and Table 5.3.7-4 for HEU fuel material and Table 5.3.7-5 and Table 5.3.7-6 for LEU fuel material. Source terms are based on a heat load limit of 1.875W per rod. SAS2H input files for the maximum burnup HEU and LEU cases are shown in Figure 5.3.7-1 and Figure 5.3.7-2, respectively.

The Incoloy 800 (density 7.94 g/cm^3) clad material composition is given in Table 5.3.7-2. A cobalt impurity concentration of 1.2 g/kg is assumed. No cobalt is listed in the manufacturer specifications for this material.

5.3.7.2 TRIGA Fuel Cluster Rod One-Dimensional Dose Rate Analysis

The task of computing one-dimensional dose rates for the dozens of source terms developed in this analysis is simplified by the use of a dose response methodology. In this approach, a dose rate response function is computed at various detector locations outside the cask. The response function for a location gives the contribution to the total dose rate from an unit source strength in each energy group. Hence, the computation of a response function involves the solution of a one-dimensional problem for each energy group in the spectrum. Here, 18 gamma responses and 7 neutron responses are computed. Only seven neutron responses are required because the spent fuel neutron spectrum is non-zero only in the first seven groups.

For the one-dimensional response calculations, the LWT basket region is represented as a homogenized smear of the fuel, fuel tube, and basket structural materials. The resulting composition of the smear is shown in Table 5.3.7-7. The basket smear is homogenized on the basis of a cylinder of radius 14.329 cm and height 279.40 cm (5 times the active fuel height).

With these response functions, the dose rate at the corresponding detector location is determined for any given burnup and cool time combination by simply multiplying the fuel source spectrum by the appropriate response function. The SCALE SAS1 sequence is used to develop dose rate response functions. The computed response functions are shown in Table 5.3.7-8 through Table 5.3.7-11 for HEU material and Table 5.3.7-12 though Table 5.3.7-15 for LEU material.

5.3.7.3 TRIGA Fuel Cluster Rod Required Cool Times

The cool time results for the HEU and LEU TRIGA fuel cluster rods are shown in Table 5.3.7-16 and Table 5.3.7-17, respectively. The result tables include the cool time required to meet the 1.875W heat load limit per rod. For each burnup and cool time combination, the normal condition 2-meter and accident condition 1-meter dose rates are given. As the cask surface dose rates are documented for the TRIGA rods to be significantly below the limits (by over a factor of 5), normal condition surface dose rates are not included in the result summary. All dose rates are significantly below licensing limits, with HEU fuel producing bounding dose rates. The maximum 1- meter dose rate under normal conditions is 17.9 mrem/hr for a transport index (TI) of 17.9.

Figure 5.3.7-1 HEU TRIGA Cluster Fuel Rod SAS2H Sample Input (600 GWd/MTU)

```
=SAS2H    PARM=(HALT04,SKIPSHIPDATA)
TRIGA v1.2 - CLST - U=10.0% - U235=92.0% - H/Zr=1.6 - 505g - 600.0 GWD/MTU
27GROUPNDF4 LATTICECELL
U 1 DEN=6.8570E-01 1.0 517 92235 92.0 92238 8.0 END
Zr 1 DEN=6.0641E+00 1.0 517     END
H 1 DEN=1.0720E-01 1.0 517     END
NI 2 DEN=7.94 0.3250 517     END
FE 2 DEN=7.94 0.4182 517     END
CR 2 DEN=7.94 0.2100 517     END
C 2 DEN=7.94 0.0010 517     END
MN 2 DEN=7.94 0.0150 517     END
S 2 DEN=7.94 0.0002 517     END
SI 2 DEN=7.94 0.0100 517     END
CU 2 DEN=7.94 0.0075 517     END
CO 2 DEN=7.94 0.0012 517     END
AL 2 DEN=7.94 0.0060 517     END
TI 2 DEN=7.94 0.0060 517     END
H2O 3 DEN=9.8060E-01 1.0 363     END
C 4 DEN=2.4823E+00 1.0 300     END
Zr 5  1.0 517     END
U 6 DEN=6.8570E-01 1.0 517 92235 92.0 92238 8.0 END
Zr 6 DEN=6.0641E+00 1.0 517     END
H 6 DEN=1.0720E-01 1.0 517     END
END COMP
SQUAREPITCH 1.74752 1.295 1 3 1.377 2   END
MORE DATA  ISN=16 IIM=50 ICM=50    END
NPIN/ASSM=1 FUELNGTH=55.88 NCYCLES=4 NLIB/CYC=1 LIGHTEL=5
PRINTLEVEL=6 INPLEVEL=2
NUMZTOTAL=1 MXREPEATS=1 END
500 1.39432
POWER=1.9310E-02 BURN=392.2838 DOWN=10 END
POWER=1.9310E-02 BURN=392.2838 DOWN=10 END
POWER=1.9310E-02 BURN=392.2838 DOWN=10 END
POWER=1.9310E-02 BURN=392.2838 DOWN=90 END
FE 0.672
CR 0.19
NI 0.115
MN 0.02
CO 0.0012
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71 E
1$$  1 1T
Decay - fission product gamma rebin
3$$  21 0 1 A33 -86 E
54$$  A8 1 E T
35$$  0 T
56$$  0 1 A13 -2 5 3 E
57**    0.2464 E T
Decay - fission product gamma rebin
SINGLE Rod
60**     3.4759
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F1.e-6
81$$  2 51 26 1 E
82$$  F6 T
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
56$$  F0 T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71 E
1$$  1 1T
Decay - actinide gamma rebin
3$$  21 0 1 A33 -86 E
54$$  A8 1 E T
35$$  0 T
56$$  0 1 A13 -2 5 3 E
57**    0.2464 E T
Decay - actinide gamma rebin
SINGLE Rod
60**     3.4759
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F1.e-6
```

```
81$$  2 51 26 1 E
82$$  F5 T
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
56$$  F0 T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71 E
1$$  1 1T
Decay -  light element gamma rebin
3$$  21 0 1 A33 -86 E
54$$  A8 1 E T
35$$  0 T
56$$  0 1 A13 -2 5 3 E
57**    0.2464 E T
Decay - light element gamma rebin
SINGLE Rod
60**      3.4759
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F1.e-6
81$$  2 51 26 1 E
82$$  F4 T
LIGHT ELEMENT SCALE GROUP STRUCTURE
56$$  F0 T
END
```

Figure 5.3.7-2 LEU TRIGA Cluster Fuel Rod SAS2H Sample Input (140 GWd/MTU)

```
=SAS2H     PARM=(HALT04,SKIPSHIPDATA)
TRIGA v1.2 - CLST - U=45.0% - U235=19.0% - H/Zr=1.6 - 643g - 140.0 GWD/MTU
27GROUPNDF4 LATTICECELL
U 1 DEN=3.9289E+00 1.0 517 92235 19.0 92238 81.0 END
Zr 1 DEN=4.7186E+00 1.0 517     END
H 1 DEN=8.3417E-02 1.0 517     END
NI 2 DEN=7.94 0.3250 517     END
FE 2 DEN=7.94 0.4182 517     END
CR 2 DEN=7.94 0.2100 517     END
C 2 DEN=7.94 0.0010 517     END
MN 2 DEN=7.94 0.0150 517     END
S 2 DEN=7.94 0.0002 517     END
SI 2 DEN=7.94 0.0100 517     END
CU 2 DEN=7.94 0.0075 517     END
CO 2 DEN=7.94 0.0012 517     END
AL 2 DEN=7.94 0.0060 517     END
TI 2 DEN=7.94 0.0060 517     END
H2O 3 DEN=9.8060E-01 1.0 363     END
C 4 DEN=2.4823E+00 1.0 300     END
Zr 5  1.0 517     END
U 6 DEN=3.9289E+00 1.0 517 92235 19.0 92238 81.0 END
Zr 6 DEN=4.7186E+00 1.0 517     END
H 6 DEN=8.3417E-02 1.0 517     END
END COMP
SQUAREPITCH 1.74752 1.295 1 3 1.377 2   END
MORE DATA  ISN=16 IIM=50 ICM=50    END
NPIN/ASSM=1 FUELNGTH=55.88 NCYCLES=4 NLIB/CYC=1 LIGHTEL=5
PRINTLEVEL=6 INPLEVEL=2
NUMZTOTAL=1 MXREPEATS=1 END
500 1.39432
POWER=1.9310E-02 BURN=524.4562 DOWN=10 END
POWER=1.9310E-02 BURN=524.4562 DOWN=10 END
POWER=1.9310E-02 BURN=524.4562 DOWN=10 END
POWER=1.9310E-02 BURN=524.4562 DOWN=90 END
FE 0.672
CR 0.19
NI 0.115
MN 0.02
CO 0.0012
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71 E
1$$  1 1T
D Decay - fission product gamma rebin
3$$  21 0 1 A33 -86 E
54$$  A8 1 E T
35$$  0 T
56$$  0 1 A13 -2 5 3 E
57**    0.2464 E T
Decay - fission product gamma rebin
SINGLE Rod
60**     5.2795
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F1.e-6
81$$  2 51 26 1 E
82$$  F6 T
FISSION PRODUCT GAMMA SPECTRA IN SCALE 18 GROUPS
56$$  F0 T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71 E
1$$  1 1T
Decay - actinide gamma rebin
3$$  21 0 1 A33 -86 E
54$$  A8 1 E T
35$$  0 T
56$$  0 1 A13 -2 5 3 E
57**    0.2464 E T
Decay - actinide gamma rebin
SINGLE Rod
60**     5.2795
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F1.e-6
```

```
81$$  2 51 26 1 E
82$$  F5 T
ACTINIDE GAMMA SPECTRA IN SCALE 18 GROUPS
56$$  F0 T
END
=ORIGENS
0$$  A4 21 A8 26 A10 51 71 E
1$$  1 1T
Decay - light element gamma rebin
3$$  21 0 1 A33 -86 E
54$$  A8 1 E T
35$$  0 T
56$$  0 1 A13 -2 5 3 E
57**    0.2464 E T
Decay - light element gamma rebin
SINGLE Rod
60**     5.2795
65$$  A4 1 A7 1 A10 1 A25 1 A28 1 A31 1 A46 1 A49 1 A52 1 E
61**  F1.e-6
81$$  2 51 26 1 E
82$$  F4 T
LIGHT ELEMENT SCALE GROUP STRUCTURE
56$$  F0 T
END
```

Table 5.3.7-1 TRIGA Fuel Cluster Rod Parameters

Parameter	[in]	[cm]
Overall length	30.130	76.530
Fuel height	22.000	55.880
Fuel diameter	0.510	1.295
U mass fraction	10% (HEU) / 45% (LEU)	-
^{235}U enrichment[1]	92% (HEU) / 19% (LEU)	-
Fuel mass [g]	50.5 (HEU) / 289.4 (LEU)	-
H to Zr ratio	1.6	-
Cladding thickness	0.016	0.041
Clad diameter	0.542	1.377
Clad material	Incoloy 800	-
Tube ID	0.625	1.588
Tube OD	0.750	1.905
Tube material	aluminum	-
Power [MW]	0.0193	-
Number cycles	4	-
Down time between cycles [d]	10	-
Exposure [d]	varies	-

Note: Fuel dimensions represent the nominal configuration values.

[1]Enrichments represent minimum values. Lower limit enrichments produce maximum source terms.

Table 5.3.7-2 Incoloy 800 Clad Composition

ρ=7.94 g/cm³ Isotope	Mass Fraction	Number Density [atm/b-cm]
NI	0.3250	2.6479E-02
FE	0.4182	3.5803E-02
CR	0.2100	1.9312E-02
C	0.0010	3.9846E-04
MN	0.0150	1.3055E-03
S	0.0002	2.2369E-05
SI	0.0100	1.7025E-03
CU	0.0075	5.5949E-04
CO	0.0012	9.7361E-05
AL	0.0060	1.0633E-03
TI	0.0060	5.9920E-04

Table 5.3.7-3 Representative HEU TRIGA Fuel Cluster Rod Gamma Spectra at 150 GWd/MTU and 1.342 Year Cool Time

Group	E_{min}	E_{max}	E_{av}	Fuel Gamma g/s/assy	Hardware g/s/kg
1	8.00E+00	1.00E+01	9.00E+00	1.3715E-02	0.0000E+00
2	6.50E+00	8.00E+00	7.25E+00	6.5356E-02	0.0000E+00
3	5.00E+00	6.50E+00	5.75E+00	3.3880E-01	0.0000E+00
4	4.00E+00	5.00E+00	4.50E+00	8.5991E-01	0.0000E+00
5	3.00E+00	4.00E+00	3.50E+00	1.3994E+07	2.1078E-18
6	2.50E+00	3.00E+00	2.75E+00	1.5076E+08	1.5024E+04
7	2.00E+00	2.50E+00	2.25E+00	4.6634E+10	9.6891E+06
8	1.66E+00	2.00E+00	1.83E+00	4.5791E+09	1.0911E+08
9	1.33E+00	1.66E+00	1.50E+00	3.4986E+10	4.0828E+11
10	1.00E+00	1.33E+00	1.17E+00	5.4462E+10	1.4458E+12
11	8.00E-01	1.00E+00	9.00E-01	1.3568E+11	2.4795E+11
12	6.00E-01	8.00E-01	7.00E-01	1.7889E+12	8.9675E+06
13	4.00E-01	6.00E-01	5.00E-01	4.9752E+11	7.0972E+09
14	3.00E-01	4.00E-01	3.50E-01	2.7316E+11	1.2509E+08
15	2.00E-01	3.00E-01	2.50E-01	3.5064E+11	1.1931E+08
16	1.00E-01	2.00E-01	1.50E-01	1.5849E+12	1.3870E+09
17	5.00E-02	1.00E-01	7.50E-02	1.5295E+12	5.2359E+09
18	1.00E-02	5.00E-02	3.00E-02	4.6914E+12	2.5711E+10
Total				1.0992E+13	2.1418E+12

Table 5.3.7-4 Representative HEU TRIGA Fuel Cluster Rod Neutron Spectrum at 150 GWd/MTU and 1.342 Year Cool Time

Group	E_{min}	E_{max}	E_{av}	Fuel Neutron n/s/assy
1	6.43E+00	2.00E+01	1.32E+01	3.907E-01
2	3.00E+00	6.43E+00	4.72E+00	1.025E+01
3	1.85E+00	3.00E+00	2.43E+00	2.027E+01
4	1.40E+00	1.85E+00	1.63E+00	6.741E+00
5	9.00E-01	1.40E+00	1.15E+00	5.921E+00
6	4.00E-01	9.00E-01	6.50E-01	4.708E+00
7	1.00E-01	4.00E-01	2.50E-01	9.014E-01
8-27				0
Total				4.918E+01

Table 5.3.7-5 Representative LEU TRIGA Fuel Cluster Rod Gamma Spectra

				30 GWd/MTU – 1.5 Years		140 GWd/MTU – 5.3 Years	
				Fuel Gamma	Hardware	Fuel Gamma	Hardware
Group	E_{min}	E_{max}	E_{av}	g/s/rod	g/s/kg	g/s/rod	g/s/kg
1	8.00E+00	1.00E+01	9.00E+00	1.6759E-01	0.0000E+00	8.3995E+01	0.0000E+00
2	6.50E+00	8.00E+00	7.25E+00	7.9334E-01	0.0000E+00	3.9568E+02	0.0000E+00
3	5.00E+00	6.50E+00	5.75E+00	4.0740E+00	0.0000E+00	2.0176E+03	0.0000E+00
4	4.00E+00	5.00E+00	4.50E+00	1.0234E+01	0.0000E+00	5.0286E+03	0.0000E+00
5	3.00E+00	4.00E+00	3.50E+00	1.9609E+07	3.5241E-18	5.9296E+06	1.0031E-15
6	2.50E+00	3.00E+00	2.75E+00	1.9278E+08	1.5442E+04	4.9029E+07	4.3586E+04
7	2.00E+00	2.50E+00	2.25E+00	4.4530E+10	9.9589E+06	2.4023E+09	2.8109E+07
8	1.66E+00	2.00E+00	1.83E+00	4.8799E+09	7.9346E+07	8.9544E+08	1.1584E+02
9	1.33E+00	1.66E+00	1.50E+00	3.5841E+10	4.1965E+11	3.9671E+10	1.1845E+12
10	1.00E+00	1.33E+00	1.17E+00	6.1250E+10	1.4860E+12	1.8088E+11	4.1943E+12
11	8.00E-01	1.00E+00	9.00E-01	1.5418E+11	2.7182E+11	5.3290E+11	2.3524E+10
12	6.00E-01	8.00E-01	7.00E-01	1.6820E+12	7.0347E+06	4.4478E+12	4.9556E+06
13	4.00E-01	6.00E-01	5.00E-01	4.4530E+10	5.1626E+09	1.0886E+12	1.4277E+07
14	3.00E-01	4.00E-01	3.50E-01	4.8799E+09	1.1079E+08	1.1637E+11	2.2578E+08
15	2.00E-01	3.00E-01	2.50E-01	3.5841E+10	1.0458E+08	1.7638E+11	1.7208E+08
16	1.00E-01	2.00E-01	1.50E-01	6.1250E+10	1.3677E+09	6.2780E+11	3.4656E+09
17	5.00E-02	1.00E-01	7.50E-02	1.5418E+11	5.2957E+09	7.8533E+11	1.4365E+10
18	1.00E-02	5.00E-02	3.00E-02	1.6820E+12	2.6174E+10	2.7120E+12	7.2383E+10
Total				1.0913E+13	2.2158E+12	1.0711E+13	5.4930E+12

Table 5.3.7-6 Representative LEU TRIGA Fuel Cluster Rod Neutron Spectrum

				30 GWd/MTU – 1.5 Years	140 GWd/MTU – 5.3 Years
Group	E_{min}	E_{max}	E_{av}	n/s/rod	n/s/rod
1	6.43E+00	2.00E+01	1.32E+01	3.907E-01	2.738E+03
2	3.00E+00	6.43E+00	4.72E+00	1.025E+01	3.151E+04
3	1.85E+00	3.00E+00	2.43E+00	2.027E+01	3.559E+04
4	1.40E+00	1.85E+00	1.63E+00	6.741E+00	1.972E+04
5	9.00E-01	1.40E+00	1.15E+00	5.921E+00	2.644E+04
6	4.00E-01	9.00E-01	6.50E-01	4.708E+00	2.870E+04
7	1.00E-01	4.00E-01	2.50E-01	9.014E-01	5.616E+03
8-27				0	0
Total				4.918E+01	1.503E+05

Table 5.3.7-7 Fuel Basket Region Material Composition Used in Shielding Analysis

Isotope	Number Density [a/b-cm]
HYDROGEN	1.4660E-02 (HEU) 1.1407E-02 (LEU)
CARBON-12	1.1803E-05
ALUMINUM	9.1431E-03
SILICON	5.0432E-05
SULFUR	8.8350E-07
TITANIUM	1.7750E-05
CHROMIUM	5.7206E-04
CHROMIUM(SS304)	2.9885E-03
MANGANESE	3.3640E-04
IRON	1.0607E-03
IRON(SS304)	1.0178E-02
COBALT-59	2.8841E-06
NICKEL	7.8439E-04
NICKEL(SS304)	1.3239E-03
COPPER	1.6574E-05
ZIRCONIUM	9.1613E-03 (HEU) 7.1286E-03 (LEU)
URANIUM-235	3.6989E-04 (HEU) 4.3769E-04 (LEU)
URANIUM-238	3.1758E-05 (HEU) 1.8424E-03 (LEU)

Table 5.3.7-8 Normal Condition Dose Response to Gammas for HEU TRIGA Fuel Cluster Rods

Conveyance +2m Response to Gammas [mrem/hr per 10^{10} g/sec/cm^3]

Gamma Group	Neutron	Gamma	Total
1	0.0000E+00	8.7827E+02	8.7827E+02
2	0.0000E+00	1.1271E+03	1.1271E+03
3	0.0000E+00	1.2216E+03	1.2216E+03
4	0.0000E+00	1.1507E+03	1.1507E+03
5	0.0000E+00	9.3368E+02	9.3368E+02
6	0.0000E+00	6.0752E+02	6.0752E+02
7	0.0000E+00	3.3223E+02	3.3223E+02
8	0.0000E+00	1.3049E+02	1.3049E+02
9	0.0000E+00	3.8963E+01	3.8963E+01
10	0.0000E+00	4.8262E+00	4.8262E+00
11	0.0000E+00	1.6823E-01	1.6823E-01
12	0.0000E+00	2.2095E-03	2.2095E-03
13	0.0000E+00	~0	~0
14	0.0000E+00	~0	~0
15	0.0000E+00	~0	~0
16	0.0000E+00	0.0000E+00	0.0000E+00
17	0.0000E+00	0.0000E+00	0.0000E+00
18	0.0000E+00	0.0000E+00	0.0000E+00

Table 5.3.7-9 Normal Condition Dose Response to Neutrons for HEU TRIGA Fuel Cluster Rods

Conveyance +2m Response to Neutrons [mrem/hr per 10^{10} n/sec/cm^3]

Neutron Group	Neutron	N-Gamma	Total
1	1.5108E+07	3.5697E+06	1.8678E+07
2	9.7886E+06	3.4320E+06	1.3221E+07
3	9.3287E+06	3.5005E+06	1.2829E+07
4	8.1394E+06	3.6006E+06	1.1740E+07
5	7.6472E+06	3.6808E+06	1.1328E+07
6	7.6040E+06	3.8733E+06	1.1477E+07
7	8.5034E+06	4.2076E+06	1.2711E+07

Table 5.3.7-10 Accident Condition Dose Response to Gammas for HEU TRIGA Fuel Cluster Rods

Accident 1m Response to Gammas [mrem/hr per 10^{10} g/sec/cm^3]

Gamma Group	Neutron	Gamma	Total
1	0.0000E+00	3.3595E+03	3.3595E+03
2	0.0000E+00	4.3690E+03	4.3690E+03
3	0.0000E+00	4.8553E+03	4.8553E+03
4	0.0000E+00	4.7248E+03	4.7248E+03
5	0.0000E+00	3.9709E+03	3.9709E+03
6	0.0000E+00	2.6934E+03	2.6934E+03
7	0.0000E+00	1.5268E+03	1.5268E+03
8	0.0000E+00	6.2615E+02	6.2615E+02
9	0.0000E+00	1.9495E+02	1.9495E+02
10	0.0000E+00	2.5529E+01	2.5529E+01
11	0.0000E+00	9.5430E-01	9.5430E-01
12	0.0000E+00	1.3397E-02	1.3397E-02
13	0.0000E+00	~0	~0
14	0.0000E+00	~0	~0
15	0.0000E+00	~0	~0
16	0.0000E+00	0.0000E+00	0.0000E+00
17	0.0000E+00	0.0000E+00	0.0000E+00
18	0.0000E+00	0.0000E+00	0.0000E+00

Table 5.3.7-11 Accident Condition Dose Response to Neutrons for HEU TRIGA Fuel Cluster Rods

Accident 1m Response to Neutrons [mrem/hr per 10^{10} n/sec/cm^3]

Neutron Group	Neutron	N-Gamma	Total
1	6.9825E+08	1.8022E+06	7.0005E+08
2	6.4426E+08	1.4226E+06	6.4568E+08
3	6.5723E+08	1.3323E+06	6.5856E+08
4	6.5822E+08	1.3207E+06	6.5954E+08
5	6.5912E+08	1.3375E+06	6.6046E+08
6	6.5766E+08	1.4370E+06	6.5910E+08
7	6.6848E+08	1.7133E+06	6.7019E+08

Table 5.3.7-12 Normal Condition Dose Response to Gammas for LEU TRIGA Fuel Cluster Rods

Conveyance +2m Response to Gammas [mrem/hr per 10^{10} g/sec/cm^3]			
Gamma Group	Neutron	Gamma	Total
1	0.0000E+00	7.3425E+02	7.3425E+02
2	0.0000E+00	9.5273E+02	9.5273E+02
3	0.0000E+00	1.0435E+03	1.0435E+03
4	0.0000E+00	9.9317E+02	9.9317E+02
5	0.0000E+00	8.1421E+02	8.1421E+02
6	0.0000E+00	5.3413E+02	5.3413E+02
7	0.0000E+00	2.9313E+02	2.9313E+02
8	0.0000E+00	1.1505E+02	1.1505E+02
9	0.0000E+00	3.4153E+01	3.4153E+01
10	0.0000E+00	4.1479E+00	4.1479E+00
11	0.0000E+00	1.4023E-01	1.4023E-01
12	0.0000E+00	1.7465E-03	1.7465E-03
13	0.0000E+00	~0	~0
14	0.0000E+00	~0	~0
15	0.0000E+00	~0	~0
16	0.0000E+00	0.0000E+00	0.0000E+00
17	0.0000E+00	0.0000E+00	0.0000E+00
18	0.0000E+00	0.0000E+00	0.0000E+00

Table 5.3.7-13 Normal Condition Dose Response to Neutrons for LEU TRIGA Fuel Cluster Rods

Conveyance +2m Response to Neutrons [mrem/hr per 10^{10} n/sec/cm^3]			
Neutron Group	Neutron	N-Gamma	Total
1	1.3321E+07	2.8932E+06	1.6214E+07
2	7.7218E+06	2.6000E+06	1.0322E+07
3	7.1589E+06	2.6093E+06	9.7682E+06
4	5.6939E+06	2.6182E+06	8.3120E+06
5	4.9653E+06	2.6149E+06	7.5802E+06
6	4.4947E+06	2.6660E+06	7.1607E+06
7	4.8302E+06	2.7813E+06	7.6116E+06

Table 5.3.7-14 Accident Condition Dose Response to Gammas for LEU TRIGA Fuel Cluster Rods

Accident 1m Response to Gammas [mrem/hr per 10^{10} g/sec/cm^3]

Gamma Group	Neutron	Gamma	Total
1	0.0000E+00	2.8056E+03	2.8056E+03
2	0.0000E+00	3.6896E+03	3.6896E+03
3	0.0000E+00	4.1438E+03	4.1438E+03
4	0.0000E+00	4.0750E+03	4.0750E+03
5	0.0000E+00	3.4606E+03	3.4606E+03
6	0.0000E+00	2.3668E+03	2.3668E+03
7	0.0000E+00	1.3464E+03	1.3464E+03
8	0.0000E+00	5.5180E+02	5.5180E+02
9	0.0000E+00	1.7081E+02	1.7081E+02
10	0.0000E+00	2.1931E+01	2.1931E+01
11	0.0000E+00	7.9507E-01	7.9507E-01
12	0.0000E+00	1.0585E-02	1.0585E-02
13	0.0000E+00	~0	~0
14	0.0000E+00	~0	~0
15	0.0000E+00	~0	~0
16	0.0000E+00	0.0000E+00	0.0000E+00
17	0.0000E+00	0.0000E+00	0.0000E+00
18	0.0000E+00	0.0000E+00	0.0000E+00

Table 5.3.7-15 Accident Condition Dose Response to Neutrons for LEU TRIGA Fuel Cluster Rods

Accident 1m Response to Neutrons [mrem/hr per 10^{10} n/sec/cm^3]

Neutron Group	Neutron	N-Gamma	Total
1	5.8927E+08	1.4562E+06	5.9073E+08
2	5.0824E+08	1.0199E+06	5.0925E+08
3	5.1272E+08	9.0017E+05	5.1362E+08
4	4.9699E+08	8.4472E+05	4.9784E+08
5	4.8366E+08	8.2164E+05	4.8448E+08
6	4.5526E+08	8.4466E+05	4.5610E+08
7	4.1740E+08	1.0303E+06	4.1843E+08

Table 5.3.7-16 HEU TRIGA Fuel Cluster Rod Dose Rate Results at Various Fuel Burnups

Burnup [GWd/MTU]	Depletion [% ^{235}U]	Cool Time (days)	Accident (mrem/hr)	Normal 2-m (mrem/hr)
20	2.8%	170.5	11.4	2.4
30	4.1%	213.4	14.6	3.1
40	5.6%	246.4	17.5	3.7
50	6.9%	273.8	19.9	4.3
60	8.4%	298.7	21.9	4.7
70	9.7%	322.6	23.6	5.1
80	11.2%	344.6	24.9	5.3
90	12.5%	366.9	26.0	5.6
100	14.0%	388.2	26.8	5.7
110	15.3%	410.1	27.4	5.9
120	16.8%	429.5	27.9	6.0
130	18.1%	451.0	28.2	6.0
140	19.6%	471.4	28.4	6.1
150	20.9%	490.1	28.5	6.1
160	22.2%	510.0	28.5	6.1
170	23.7%	527.6	28.5	6.1
180	25.0%	545.3	28.5	6.1
190	26.5%	563.4	28.3	6.0
200	27.8%	580.6	28.2	6.0
225	31.3%	620.7	27.8	5.9
250	34.7%	660.3	27.3	5.8
275	38.1%	697.4	26.8	5.6
300	41.4%	734.5	26.2	5.5
325	44.8%	772.7	25.5	5.3
350	48.3%	806.1	25.1	5.2
375	51.5%	845.3	24.5	5.1
400	55.0%	881.8	24.0	5.0
425	58.2%	922.6	23.4	4.8
450	61.4%	963.2	23.0	4.7
475	64.7%	1005.8	22.6	4.6
500	67.9%	1051.3	22.2	4.4
525	71.1%	1099.4	21.9	4.3
550	74.4%	1150.5	21.7	4.2
575	77.6%	1207.2	21.6	4.2
600	80.6%	1269.6	21.6	4.1

Table 5.3.7-17 LEU TRIGA Fuel Cluster Rod Dose Rate Results at Various Fuel Burnups

Burnup [GWd/MTU]	Depletion [% ^{235}U]	Cool Time (years)	Accident (mrem/hr)	Normal 2-m (mrem/hr)
2	1.3%	0.32	10.3	2.2
3	2.0%	0.42	9.3	2.0
4	2.7%	0.50	10.8	2.3
5	3.3%	0.57	12.4	2.7
10	6.6%	0.80	18.8	4.0
15	9.8%	1.0	22.3	4.8
20	13.1%	1.2	24.1	5.1
30	19.5%	1.5	24.5	5.2
40	25.7%	1.8	23.6	5.0
50	31.7%	2.0	22.3	4.7
60	37.5%	2.2	21.1	4.4
70	43.4%	2.5	20.1	4.0
80	48.8%	2.7	19.5	3.8
90	54.3%	3.0	19.2	3.5
100	59.6%	3.3	19.7	3.2
110	64.5%	3.7	21.1	3.0
120	69.4%	4.1	23.9	2.9
130	74.0%	4.6	28.2	2.8
140	78.5%	5.3	34.6	2.7