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SHIPMENT OF GENERAL ATOMICS HOT CELL IRRADIATED FUEL MATERIALS FINAL REPORT

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ACRONYMS/ABBREVIATIONS

CCTV	Closed-Circuit Television
CFR	Code of Federal Regulations
CHP	California Highway Patrol
CoC	Certificate of Compliance
CVSA	Commercial Vehicle Safety Alliance
D&D	Decontamination and Decommissioning
DOE	US Department of Energy
DOE/OAK	DOE Oakland Operations Office
DOT	US Department of Transportation
DTS	NAC Dry Transfer System
GA	General Atomics
HCF	Hot Cell Facility
HP	GA Health Physics
HTGR	High Temperature Gas-cooled Reactor
ICM	Interim Compensatory Measures (established by NRC)
IFM	Irradiated Fuel Materials
IFSF	Irradiated Fuel Storage Facility
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho National Technology and Engineering Center
ITS	NAC Interim Transfer System
LSNC	GA Licensing Safety and Nuclear Compliance
NAC	NAC International, Norcross, GA
NAC-LWT	NAC Legal Weight Truck (Shipping Cask)
NRC	US Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
PIE	Post-Irradiation Examination
PyC	Pyrolitic Carbon (HTGR fuel particle coatings)
QA	GA Quality Assurance
RFP	Request for Proposal
RERTR	Reduced Enrichment Research and Test Reactor
RWP	Radiation Work Permit
SiC	Silicon Carbide (HTGR fuel particle coatings)
SNM	Special Nuclear Material

SOW	Statement of Work
SS	Stainless Steel
TRANSCOM	Transportation Tracking and Communications System
TRIGA	Training, Research, Isotope-production, General Atomic (reactor)
TSMT	Tri-State Motor Transit Company, Joplin, MO
WA	GA Work Authorization (Radiological)
WBS	Work Breakdown Structure

1.0 SUMMARY

In September, 2003, General Atomics (GA) successfully packaged and shipped a quantity of U.S. Department of Energy (DOE) owned Irradiated Fuel Materials (IFM) from the GA Main Site, San Diego, CA, to the Idaho National Engineering and Environmental Laboratory (INEEL), Idaho Falls, ID, for long-term interim storage. INEEL will store the GA IFM at the Irradiated Fuel Storage Facility (IFSF) at the Idaho National Technology and Engineering Center (INTEC).

The IFM packaging and shipment activities performed, and described in this report, were funded under Contract No. DE-AC03-95SF20798 (Ref. 1) between GA and the DOE, under Work Breakdown Structure (WBS) 2.4.4.2, entitled "IFM Disposal". All packaging and transportation tasks were successfully completed within budget and on schedule.

The IFM shipped in this campaign included two separate spent nuclear fuel quantities, identified as High-Temperature Gas-cooled Reactor (HTGR) and Reduced Enrichment Research and Test Reactor (RERTR) fuel. Receipt of the GA IFM at INEEL complied with the requirements of INEEL Document No. STD-1120, "Standard for Receipt of Spent Nuclear Fuel" (Ref. 2).

Highway transport of the subject IFM was conducted in accordance with the requirements of all U.S. Nuclear Regulatory Commission (NRC) and U.S. Department of Transportation (DOT) regulations. Containment packaging of the IFM for highway transport was provided by the NAC International (NAC) Model No. NAC-LWT shipping cask, NRC Certificate of Compliance (CoC) No. 9225, which was specifically amended to authorize transport of the GA IFM, (see Appendix C).

This report describes the steps taken by GA and it's subcontractors to complete the packaging and shipment of the GA IFM. Included in the report Appendices are copies of pertinent documentation specifically prepared for this project, and a series of photographs taken at GA during actual IFM transfers, cask loading, and shipping operations.

2.0 BACKGROUND

2.1 CONTRACTUAL REQUIREMENTS

GA Hot Cell Decommissioning Project work is being performed to meet the requirements set forth in the Statement of Work (SOW), Attachment I of Contract No. DE-AC03-95SF20798, Rev 01/12/96 (Ref. 1), issued to General Atomics, San Diego, CA, by the Oakland Operations Office of the U.S. Department of Energy, Oakland, CA. (DOE/OAK). The specific requirements of IFM packaging, characterization, interim storage at GA, and shipment of IFM from GA to the INEEL are outlined in Task 16 of the above-referenced DOE contract SOW. Task 16 of the contract SOW, entitled "Transfer of the Irradiated Fuel Materials to a DOE Location for Storage", states in part that:

i) *"GA shall transfer from the Hot Cell Facility and accommodate the temporary storage of the Irradiated Fuel Materials (in an on-site GA location)..."*

ii) *"GA shall fully describe/characterize the Irradiated Fuel Materials in accordance with the Idaho National Engineering and Environmental Laboratory acceptance criteria.....and work with the INEEL to negotiate storage of the Irradiated Fuel Materials in Idaho..."*

iii) *"GA shall coordinate with INEEL to determine...the packaging/shipping requirements for the storage of the Irradiated Fuel Materials at INEEL."*

iv) *"...GA shall provide all the services required including personnel, equipment, and supplies necessary to package and safely transport the materials to the designated DOE storage site.....Materials shall be packaged and transported between sites per applicable federal, state, and local laws and regulations. In addition, GA shall provide timely notification to DOE of shipping dates and shall provide timely notification to respective states routing agencies..."*

2.2 CONSOLIDATION AND INITIAL PACKAGING OF IFM

The GA IFM had been collected and retained in the GA Hot Cell Facility (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 active operating history of the GA HCF (1959 through 1992). In November, 1995, as part of the initial actions of active Decontamination and Decommissioning (D&D) of the GA HCF, the subject IFM was remotely inspected, inventoried, and packaged in the HCF, for the purpose of physical removal of the material from that building, and the temporary storage in a separate facility on the GA site. The removal of the IFM from the GA HCF was necessary to allow for HCF D&D actions to proceed. HCF building and site D&D operations, which were successfully carried out and completed from 1995 through 2001, are described and documented in GA Report

No. PC-000499, entitled "General Atomics Hot Cell Facility Decommissioning Project Final Report", Rev. 0, dated September, 2001 (Ref. 3).

As part of the initial IFM packaging process, the IFM mass 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 one IFM group composed of the Reduced-Enrichment Research and Test Reactor (RERTR) type fuel (designated as RERTR/IFM).

The initial packaging of each of the two IFM groups, performed in the GA HCF in November, 1995, involved the remote handling, collection, loading, and weld-encapsulation of the IFM into 304SS-construction Primary and Secondary Enclosures, the design of which allowed for the subsequent installation of the packaged IFM groups into separate, shielded GA-owned storage casks. Design details of these packages are described in GA Document No. PC-000384, entitled "HTGR/RERTR Fuel Materials Characterization and Packaging Report", Rev. 2, dated April, 2002 (Ref. 4).

The initial packaging of the HTGR/IFM involved the physical consolidation of several 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. LAF2050100), had been separately controlled, inventoried, and retained as historical samples at the GA Hot Cell Facility (HCF) in shielded, retrievable storage. However, to facilitate handling and disposal of these HTGR 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 initial packaging of the RERTR/IFM did not necessitate DOE authorization for physical consolidation, as this material was received at the GA HCF as a single discrete inventory line item. The RERTR/IFM was received at the GA HCF in 1985, from the Oak Ridge National Laboratory (ORNL) as irradiated fuel, was held at GA under the RERTR Fuel Base Program (DOE Project No. C400480000), and had been separately controlled, inventoried, and retained as historical samples at the GA HCF in shielded, retrievable storage.

2.3 TEMPORARY STORAGE OF IFM AT GA

GA safely controlled and stored the initially packaged IFM in two separate unlicensed shipping casks from December, 1995 through September, 2003. Over this time period, GA utilized a succession of three different storage locations on the GA site, including GA Bldgs. 30, 31, and finally Bldg. 21. In each of these storage locations, GA provided appropriate security and radiological/nuclear safety measures necessary for the safe control and surveillance of the stored IFM. Information related to the temporary storage of the subject IFM at GA has been previously provided to the DOE in GA Document No. PC-000457, entitled "Safeguards and Security Measures for the Irradiated Fuel Material Temporary Storage Facility at General Atomics", Rev. 1, dated January, 2000 (Ref. 5).

3.0 DESCRIPTION OF GA IFM

Details of GA IFM characterization methodologies and specific fuel and packaging parameters for both HTGR and RERTR IFM are contained in GA Document No. PC-000384, entitled "HTGR/RERTR Fuel Materials Characterization and Packaging Report", Rev. 2, dated April, 2002 (Ref. 4). In order to fully comply with the requirements set forth in INEEL Document No. STD-1120, entitled "Standard for Receipt of Spent Nuclear Fuel", Rev. ID:0, dated 8/31/01 (Ref. 2), it was necessary for GA to complete and submit to INEEL specific forms containing detailed information regarding the IFM fuel and packaging characteristics. Copies of these completed and subsequently approved forms, entitled "Fuel and Packaging Required Shipper Data (RSD) Forms", are included in this report as Appendix A.

3.1 HTGR IFM

The HTGR IFM is comprised of a consolidated, previously irradiated fuel mass in three forms, loose coated fuel particles, fuel compacts, and fuel pebbles.

- Coated fuel particles are solid, spheridized, sintered ceramic fuel kernels, composed of UC_2, UCO, UO_2, $(Th,U)C_2$, or $(Th,U)O_2$ substrate, isotropically coated with discrete multi-layered fuel particle coatings, composed of pyrolitic carbon (PyC) and silicon carbide (SiC).
- Fuel compacts are multi-coated ceramic fuel particles (described above), bound in solid, cylindrical, injection-molded, high-temperature heat-treated compacts, the binding matrix of which is composed of carbonized graphite shim, coke, and graphite powder.
- Fuel pebbles are multi-coated ceramic fuel particles (described above), bound in solid, spherical, injection-molded, high-temperature heat-treated pebbles, the binding matrix of which is composed of carbonized graphite shim, coke, and graphite powder.

The initial enrichment of the HTGR IFM varied from 10.0 to 93.15 wt% U-235. The nuclear material and fission product radionuclide content of the HTGR IFM, decayed to the reference decay date of 9/30/03, is presented below.in Table 1.

3.2 RERTR IFM

The RERTR IFM is comprised of 20 irradiated TRIGA-type 0.512 in. (1.30 cm) dia. x 22.05 in (56.0 cm) intact length, Incoloy 800H clad fuel elements; 13 of the elements are intact assemblies, the remaining 7 were physically sectioned for post-irradiation examination. The RERTR fuel matrix is a metal alloy comprised of uranium-zirconium hydride. The elements contain three distinct mass loadings of uranium, i.e., 20, 30, and 45 wt% U.

The initial enrichment of the RERTR IFM was approximately 19.7 wt% U-235. The nuclear material and fission/activation product radionuclide content of the RERTR IFM, decayed to the reference decay date of 9/30/03, is presented below in Table 1.

TABLE 1: RADIONUCLIDE CONTENT OF HTGR AND RERTR IFM, AS OF 9/30/03

Nuclide	HTGR IFM		RERTR IFM		Total HTGR+RERTR	
	Activity (Ci)	Mass (g)	Activity (Ci)	Mass (g)	Activity (Ci)	Mass (g)
H-3	1.96E-01	2.02E-05	1.61E+00	1.66E-04	1.81E+00	1.87E-04
Mn-54	0.00E+00	0.00E+00	2.14E-05	2.58E-09	2.14E-05	2.58E-09
Fe-55	0.00E+00	0.00E+00	4.27E+00	1.94E-03	4.27E+00	1.94E-03
Co-60	0.00E+00	0.00E+00	8.84E-01	8.04E-04	8.84E-01	8.04E-04
Ni-59	0.00E+00	0.00E+00	3.30E-01	4.07E+00	3.30E-01	4.07E+00
Ni-63	0.00E+00	0.00E+00	3.75E+01	8.16E-01	3.75E+01	8.16E-01
Kr-85	5.57E+00	1.39E-02	3.55E+01	8.88E-02	4.11E+01	1.03E-01
Sr-90	1.27E+02	8.45E-01	6.32E+02	4.21E+00	7.59E+02	5.06E+00
Y-90	1.27E+02	6.10E-04	6.32E+02	2.53E-03	7.59E+02	3.14E-03
Tc-99	0.00E+00	0.00E+00	1.40E-01	8.24E+00	1.40E-01	8.24E+00
Ru-106	0.00E+00	0.00E+00	3.43E-03	1.01E-06	3.43E-03	1.01E-06
Sb-125	1.64E-02	1.17E-05	5.97E-01	4.27E-04	6.14E-01	4.38E-04
Cs-134	2.70E-02	2.25E-05	1.71E+00	1.42E-03	1.73E+00	1.44E-03
Cs-137	1.31E+02	1.34E+00	6.91E+02	7.05E+00	8.22E+02	8.39E+00
Pm-147	3.34E-01	3.56E-04	1.22E+01	1.30E-02	1.25E+01	1.33E-02
Sm-151	1.20E+00	4.63E-02	3.16E+00	1.22E-01	4.36E+00	1.68E-01
Eu-154	8.11E-01	5.41E-03	1.28E+01	8.52E-02	1.36E+01	9.06E-02
Eu-155	4.78E-02	3.42E-05	2.15E+00	1.53E-03	2.20E+00	1.57E-03
Th-232	2.10E-04	1.91E+03	0.00E+00	0.00E+00	2.10E-04	1.91E+03
U-233	2.92E-01	3.07E+01	1.71E-07	1.80E-05	2.92E-01	3.07E+01
U-234	3.13E-02	5.05E+00	3.91E-04	6.30E-02	3.17E-02	5.11E+00
U-235	2.27E-04	1.08E+02	7.39E-04	3.52E+02	9.66E-04	4.60E+02
U-236	1.04E-03	1.66E+01	5.61E-03	8.90E+01	6.65E-03	1.06E+02
U-238	3.84E-06	1.16E+01	8.58E-04	2.60E+03	8.62E-04	2.61E+03
Np-237	0.00E+00	0.00E+00	2.48E-03	3.60E+00	2.48E-03	3.60E+00
Pu-238	2.74E+00	1.61E-01	0.00E+00	0.00E+00	2.74E+00	1.61E-01
Pu-239	1.70E-02	2.75E-01	1.30E+00	2.10E+01	1.32E+00	2.13E+01
Pu-240	1.91E-02	8.31E-02	1.35E+00	5.86E+00	1.37E+00	5.94E+00
Pu-241	2.16E+00	1.96E-02	1.95E+02	1.78E+00	1.98E+02	1.80E+00
Pu-242	1.08E-04	2.77E-02	3.35E-03	8.60E-01	3.46E-03	8.88E-01
Total	3.98E+02		2.27E+03		2.66E+03	

3.3 DESCRIPTION OF IFM CANISTERS

The HTGR and RERTR IFM were both separately packaged in a right-circular cylindrical welded Primary Enclosure, encased inside a welded Secondary Enclosure, with integral lifting bail. The construction material of the Primary and Secondary Enclosures is Type 304 Stainless-Steel (SS) seamless tubing, with end caps of Type 6600 Inconel plate. The lifting bail on each of the Secondary Enclosures is constructed of 0.125" (0.32 cm) diameter Type 304 SS wire rope, the ends of which are threaded through Type 304 SS 0.50" (1.27 cm) square blocks, and affixed with crimped copper stop sleeves.

The HTGR IFM canister external dimensions (i.e., HTGR Secondary Enclosure) are 39.05 in. height x 5.25 in. diameter (99.2 cm h x 13.3 cm dia). The gross weight of the HTGR IFM canister is the combined weight of HTGR fuel materials, 23.52 lb (10.668 kg), plus the weight of the Enclosures, 47.98 lb (21.764 kg), for a total loaded canister weight of 71.50 lb (32.432 kg).

The RERTR IFM canister external dimensions (i.e., RERTR Secondary Enclosure) are 37.25 in. height x 4.75 in. diameter (94.6 cm h x 12.1 cm dia). The gross weight of the RERTR IFM canister is the combined weight of RERTR fuel materials, 23.73 lb (10.766 kg), plus the weight of the Enclosures and non-fuel element components, 52.27 lb (23.708 kg), for a total loaded canister weight of 76.00 lb (34.474 kg).

Design details of the HTGR and RERTR Primary and Secondary Enclosures are provided in GA Engineering Drawings No. 032237/B, 032231/A, 032236/B, and 032230/A, (Refs. 6, 7, 8, and 9).

4.0 IFM SHIPMENT PROJECT ACTIONS

The specific actions taken by GA and its subcontractors to complete this shipment included the following steps (tasks are listed in general chronological order; approximate action dates are shown in parentheses):

- Preparation, issuance, and submittal to DOE/OAK of monthly progress reports, describing project activities and financial details, (ongoing, from inception through Dec/03).
- Preparation and issuance of a report describing the radiological characteristics and containment packaging details concerning the GA IFM, entitled "HTGR/RERTR Fuel Materials Characterization and Packaging Report", GA Doc. No. PC-000384/2 (Ref. 4), (Apr/02).
- Travel by GA personnel to Idaho Falls, ID, to meet with INEEL, DOE/ID, and DOE/OAK representatives for initial project planning, and to establish interface communications between GA, NAC, INEEL, DOE/OAK and DOE/ID to coordinate shipment activities, (Jun/02).
- Preparation of a comprehensive Statement of Work (SOW) for the shipment of the IFM, and the distribution of this SOW as a Request for Proposal (RFP) to qualified candidate shipping cask service vendors, (Jul/02).
- Completion and submittal to the INEEL of Fuel and Packaging Required Shipper Data (RSD) forms for receipt and storage of the GA IFM, in accordance with the INEEL criteria entitled "Standard for Receipt of Spent Nuclear Fuel", INEEL Doc. No. STD-1120, Revision ID:0, dated 8/31/01 (Ref. 2), (initial GA submittals sent Aug/02; final INEEL approvals received Sep/03), **[see Appendix A]**.
- Collection and submittal to INEEL of comprehensive reference documentation package including all GA Pre-irradiation and Post-irradiation Reports covering the numerous Fuel Test Irradiation Capsules represented in the HTGR and RERTR IFM. (Oct/02).
- Travel to and audit of NAC International, Norcross, GA, (NAC) in order to qualify NAC as an "Approved Vendor", in accordance with the GA Quality Assurance program, (Oct/02).
- Preparation and submittal to INEEL of videotape footage depicting the November, 1995 remote loading of the HTGR and RERTR IFM canisters in the GA HCF, (Nov/02).
- Awarding of a shipping cask services contract by GA to NAC International (NAC), Norcross, GA; NAC was the low-cost bidder among the qualified vendor candidates; (Nov/02).

Start

- Preparation and issuance of a GA Site Assessment Report by NAC for the physical IFM transfer and shipping cask loading operations at the GA site, (Feb/03).
- Physical driving of the proposed IFM highway transport route from San Diego, CA to Idaho Falls, ID, by NAC engineers, in order to develop a detailed Transportation Plan, (Feb/03).
- Design, review by GA and INEEL, and fabrication of special IFM Basket (Top Module) and Spacer hardware for the containment of the GA IFM in the NAC-LWT shipping cask, (initial design Feb/03, final approval Jun/03), **[see Appendix B]**.

- Application for and approval acquisition of specific amendment to the NAC shipping cask U.S. Nuclear Regulatory Commission (NRC) Radioactive Material Package License to allow the transport of the GA IFM in the NAC-LWT shipping cask as "Approved Contents", (initial application Feb/03, final approval Jun/03), **[see Appendix C]**.

- Application for GA designation as "NRC Authorized User" status for the NAC-LWT shipping cask, (request submitted Feb/03, NRC approval Apr/03).

- Preparation, submittal, and acquisition of specific Route Approval from the U.S. NRC for the highway transport of the NAC-LWT shipping cask containing the GA IFM from GA, San Diego, CA, to the INEEL, Idaho Falls, ID, (initial application Feb/03, final approval Jul/03).

- Development and issuance of shipment-specific Shielding/Source Term Calculations, Criticality Safety Evaluation, Structural/Thermal/Containment Calculations, Security Plan, and Transportation Plan for the shipment of GA IFM, (Feb/03).

- Preparation and finalization of a Quality Assurance Interface Agreement between GA and INEEL, (Mar/03).

- Travel by GA personnel to Idaho Falls, ID, to meet with INEEL, DOE/ID, and DOE/OAK representatives for mid-course project planning, and to establish interface communications between GA, NAC, INEEL, DOE/OAK and DOE/ID to coordinate shipment activities, (Mar/03).

- Travel by NAC engineers to San Diego, CA, to discuss specific methodologies to be utilized for the free-air transfer of the IFM canisters to the NAC Basket, and the loading of the NAC-LWT cask, (Apr/03).

- Development and issuance of specific radiation dose calculations for the free air transfer of the IFM canisters in the Dry Pit and the handling of the loaded NAC Intermediate Transfer System (ITS) Inner Shield, (Apr/03).

- Travel by INEEL personnel to San Diego, CA, to review records and perform audit of GA Quality Assurance Program. GA QA Program certified as acceptable by INEEL, (Apr/03).

- Preparation and approval by INEEL, DOE/ID, DOE/OAK, and GA of a Material Control and Accountability Shipper/Receiver Agreement for shipment of the GA IFM, (May/03).

- Collection and submittal to INEEL of additional IFM-related supporting and reference documentation, which had been identified and requested by INEEL during their April, 2003 QA Program audit at GA, (May/03).

- Preparation and submittal, by NAC to GA and INEEL, of fabrication procedures and travelers to be utilized for the machine fabrication of the NAC Basket (Top Module) and Spacers to be used for NAC-LWT transport of the GA IFM, (May/03).

- Finalization and approval of a Shipper/Receiver Agreement between GA, INEEL, DOE/OAK, and DOE/ID, (Jun/03).

- Development and issuance of specific procedures for the loading and shipment of the IFM from GA to the INEEL, (Jul/03), **[see Appendix D]**.

- Specification and procurement of required hardware and equipment to complete IFM transfer operations, e.g., man-bridge, special high-capacity lifting slings, ladder, CCTV cameras and monitors, etc., (Jul/03).
- Design and fabrication of specialty equipment required for the transfer of IFM canisters from GA storage casks to the NAC-LWT shipping cask, e.g., IFM canister grapples, reach tools, etc., (Jul/03).
- Design and fabrication of surrogate storage cask and mock-up IFM canister, in preparation for operational dry runs, (Jul/03).
- Development, issuance, and periodic revision of an overall Project Schedule, including detailed Work Breakdown Structure (WBS) for all project tasks, (Jul/03), **[see Appendix E]**.
- Application for and acquisition of a specific amendment to the GA site-wide U.S. NRC Radioactive Material License to allow for the handling and high lift of the GA IFM storage casks, (application submitted Aug/03, final NRC approval Sep/03).
- Preparation and set up of the empty TRIGA Mark III reactor (MkIII) Dry Pit for the free air transfer of the two highly radioactive canisters containing the IFM from the GA storage casks to the IFM Basket and NAC Interim Transfer System (ITS) Inner Shield. These actions included the assembly and installation of OSHA-approved man-bridge to span the Dry Pit, and a vertical ladder for Dry Pit entry/egress, (Aug/03).
- Preparation and issuance of a GA Project Training Plan, and identification of GA and subcontractor project personnel assignments, (Aug/03).
- Identification of personnel and establishment of project-specific GA Readiness Review Committee, (Aug/03).
- Preparation and issuance of a job-specific nuclear safety evaluation of the IFM loading operations, and a pre-job radiological safety study to determine expected personnel radiological doses and permitted stay times for individual tasks, (Aug/03).
- Travel by GA, NAC, and INEEL Engineers to subcontractor machine shop, i.e., Columbiana/Hi-Tech, Greensboro, NC, to inspect the recently fabricated NAC Basket (Top Module), and Spacer to be utilized in the NAC-LWT cask for IFM transport, (8/20/03).
- Travel by GA Engineers to subcontractor shipping cask storage/maintenance facility, i.e., Alaron Corporation, Wampum, PA, to witness fit-up of the NAC Basket (Top Module) and Spacer inside the NAC-LWT cask cavity, and the operational checkout of auxiliary NAC transfer equipment, (8/27/03).
- Travel by NAC to GA to participate in the formal GA Readiness Review for IFM shipment project, (9/3/03).
- Performance of operational IFM transfer practice runs, utilizing the mock-up IFM canister and storage cask, and specialty reach tools and grapples, (9/4/03 thru 9/11/03).

- Notification of state governor's designees of the planned shipment through the states of California, Nevada, Arizona, Utah, and Idaho, (as required by 10CFR73), and arrangement for continuous highway escort of the shipment by state and/or local law enforcement throughout the transport route, (9/12/03).
- Lease arrangement for specific equipment required for IFM transfer and shipping operations, e.g., mobile truck crane, man-lifts, air compressor, etc., (9/12/03).
- Mobilization and training of GA project and subcontractor personnel to be involved in the IFM transfer and shipping activities, (9/16/03).
- Receipt, inspection, set up, and operational checkout of the NAC-LWT shipping cask and auxiliary equipment at GA, (9/16/03). — 8
- Performance of final operational dry runs utilizing mock-up IFM canister and storage cask, (9/17/03).
- Final GA Readiness Review Committee review and approval to proceed with actual IFM transfer/cask loading actions, (9/18/03).
- Preparation and issuance of a final job-specific GA Health Physics Radiological Work Permit (RWP) for IFM transfer/cask loading operations, (9/19/03).
- Safe handling, lift, and positioning in the TRIGA Mark III Dry Pit of the two GA storage casks containing the IFM, (9/19/03). 9
- Free-air transfer of highly radioactive IFM canisters from GA storage casks to the NAC Basket/NAC Interim Transfer System (ITS), and the subsequent loading of the IFM-loaded NAC Basket into the NAC-LWT shipping cask utilizing the NAC Dry Transfer System (DTS), (9/20/03). 10
- Closure, assembly, radiological survey, leak testing, and packaging of the loaded NAC-LWT shipping cask for transport, (9/20/03).
- Preparation of necessary shipping documentation for the transport of the NAC-LWT cask containing the GA IFM from San Diego, CA to Idaho Falls, ID, (9/22/03), **[see Appendix F]**.
- Inspection and repair, as necessary, of the Tri-State Motor Transit (TSMT) trailer and tractor utilized for transport of the NAC-LWT, in accordance with the standards of the Commercial Vehicle Safety Alliance (CVSA), (9/22/03).
- Dispatch of the loaded NAC-LWT shipping cask from GA, San Diego, CA, for transport to the INEEL, Idaho Falls, ID, (9/23/03). 11
- Continuous armed vehicle escort over the entire highway transport route by State law enforcement personnel through the affected states of California, Nevada, Arizona, Utah, and Idaho, (9/23/03 – 9/24/03).
- Continuous, real-time tracking of the NAC-LWT shipping cask during entire highway transport duration, utilizing the satellite-based Transportation Tracking and Communications

(TRANSCOM) system, as required by NRC Interim Compensatory Measures (ICM) guidelines, (9/23/03 – 9/24/03)

- Radiological survey, characterization, packaging, and shipment of the two empty GA storage casks and project-related low-level radioactive waste from GA, San Diego, CA, to Alaron Corporation, Wampum, PA, for decontamination and disposal (9/23/03). 12
- Receipt at INEEL INTEC facility of the loaded NAC-LWT shipping cask transport vehicle, (9/24/03).
- Preparation and transmittal, by GA Licensing, Safety, and Nuclear Compliance (LSNC), of required U.S. DOE and U.S. NRC Nuclear Material Transaction Report, DOE/NRC Form DP-741, to document the transfer of the Special Nuclear Material (SNM) contained in the IFM between GA and INEEL, (9/25/03).
- Off-loading of NAC Basket, containing GA IFM, from the NAC-LWT cask into the IFSF at the INEEL INTEC facility, (10/7/03).
- Dispatch of empty NAC-LWT cask from INEEL for transport to Alaron Corporation, Wampum, PA, (10/8/03).
- Preparation of photographic documentation of IFM transfer and NAC-LWT cask loading operations, (Oct/03), [see **Appendix G**].
- Preparation and issuance of this project Final Report, (Nov/03).

End

5.0 PROJECT COST

The GA Hot Cell IFM Disposal Project was initially budgeted at a cost of \$ 2,377,617. The total aggregate of funds provided to GA by DOE/OAK, by the issuance of Contract Modifications through A042, was \$ 2,592,803. Including the actual billed cost from inception of Project through October, 2003, and the estimated cost that will be billed from November, 2003 through the end of the Project, GA expects the Hot Cell IFM Disposal Project cost to be \$ 2,335,825. The Project will therefore be completed within budget.

GA Hot Cell IFM Disposal Project activities will be completed by December 31, 2003. However, GA estimates that the last Project-related bill will not be received until March, 2004.

The total Project costs, including actual costs billed through October, 2003, plus those expected costs to be billed from November, 2003 through March, 2004, are presented in Table 2.

TABLE 2: TOTAL PROJECT COST BY REPORTING ELEMENT

(Note: Expected Costs are engineering summary estimates)
Project: Hot Cell IFM Disposal; Task: 07340.800.20000 (in dollars)

Reporting Element	Actual Billed Inception to Date (thru Oct/03)	Estimated Costs (Nov/03 thru Mar/04)	Note	Total Estimated Project Costs
Labor Related	429,085	50,025	(1)	479,110
Materials/Supplies	25,713			25,713
Services/Subcontracts	1,013,192	65,100	(2)	1,078,292
Other Costs	638,242	114,468	(3)(4)	752,710
Total Costs	2,106,232	229,593		2,335,825
Total Budget				2,377,617
Total Funding				2,592,803

Notes:

(1) Labor Related Estimated Costs include Final Report preparation, QA records storage, and project closeout activities.

(2) Services/Subcontracts Estimated Costs include decontamination and disposal costs for GA Storage Casks and disposal processing of project-related low-level radioactive waste at Alaron Corporation, Wampum, PA.

(3) Other Costs include Other Employee Expenses, Travel, Legal & Consulting Fees, Intercompany Transfers, Compliance, Reproduction, Occupancy, Records Management, Laboratory Allocation, Burdens, and License Fees.

(4) Estimated License Fees yet to be billed include U.S. NRC fees for NAC-LWT Cask License Amendment and fees associated with on-site surveillance by NRC inspectors during actual IFM transfer and cask loading operations.

6.0 SCHEDULE

The project schedule objective established by the U.S. DOE was to complete the loading and off-site shipment of the GA IFM, and to disposition the empty GA storage casks and all of the low-level radioactive waste generated as a result of these actions, by the end of FY2003 (i.e., 9/30/03). This goal was accomplished, i.e., on 9/23/03, the NAC-LWT cask containing the GA IFM was dispatched from GA, San Diego, CA, for transport to the INEEL, Idaho Falls, ID, and the empty storage casks and project-related low level radioactive waste were dispatched from GA for transport to Alaron Corporation, Wampum, PA.

The overall project schedule, which was developed by GA with the assistance and input from DOE/OAK, NAC, and INEEL, is provided herein as Appendix E. This schedule, which has been updated through 11/20/03, lists all major tasks and milestones, and shows actual Start and Completion dates for each task item, where appropriate.

7.0 REFERENCES

1. U.S. Department of Energy Contract No. DE-AC03-95SF20798, issued to General Atomics, San Diego, CA, by the U.S. DOE Oakland Operations Office (DOE/OAK), Oakland, CA; Revision 1/12/06.
2. "Standard for Receipt of Spent Nuclear Fuel", Document No. STD-1120, Idaho National Engineering and Environmental Laboratory (INEEL), Idaho Falls, ID; Revision ID:0, 8/31/01.
3. "General Atomics Hot Cell Facility Decommissioning Project, Final Report", Document No. PC-000499, General Atomics, San Diego, CA; Revision 0, September, 2001.
4. "HTGR/RERTR Fuel Materials Characterization and Packaging Report", Document No. PC-000384, General Atomics, San Diego, CA; Revision 2, April, 2002.
5. "Safeguards and Security Measures for the Irradiated Fuel Material Temporary Storage Facility at General Atomics", Document No. PC-000457, General Atomics, San Diego, CA; Revision 1, January, 2000.
6. "HTGR Primary Enclosure", GA Engineering Drawing No. 032237, General Atomics, San Diego, CA; Revision B, January, 1996.
7. "HTGR Secondary Enclosure", GA Engineering Drawing No. 032231, General Atomics, San Diego, CA; Revision A, January, 1996.
8. "RERTR Primary Enclosure", GA Engineering Drawing No. 032236, General Atomics, San Diego, CA; Revision B, January, 1996.
9. "RERTR Secondary Enclosure", GA Engineering Drawing No. 032230, General Atomics, San Diego, CA; Revision A, January, 1996.

APPENDIX A
COMPLETED FUEL AND PACKAGING REQUIRED SHIPPER DATA (RSD) FORMS SUBMITTED BY GA TO INEEL

FUEL RSD FORM
for HTGR ASSEMBLED CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 2)
I. SHIPPER AND SHIPMENT IDENTIFICATION INFORMATION		
1. Shipper	RIS: **LAW**	N/A
	Name: General Atomics	N/A
	Address: PO Box 85608, San Diego, CA 92138	N/A
	Telephone: (858) 455-2823; or (858) 455-2010 (24 hrs)	N/A
	Attention: Dr. Keith Asmussen	N/A
2. Shipping Agent	Name: NAC International	N/A
	Location: 3930 East Jones Bridge Road, Norcross, GA 30092	N/A
	Name of authorized person for shipping agent: Michael J. Mosley	N/A
3 License number	SNM 696	N/A
4. Transfer authority - contact, NM draft or order number	DOE Contract No: DE-AC03-95SF20798	Reference 1
5. License number for import (foreign receipts)	N/A, (SNF is shipped from USA)	N/A
6. U.S. port of entry (foreign receipts)	N/A, (SNF is shipped from USA)	N/A
7. Batch identification number	From Project No. 7340	Reference 2

FUEL RSD FORM
for HTGR ASSEMBLED CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary		References (See Table 2)
8. Country control number	Fuel ID	Country Control Number	Reference 15
	FSV/FTE-2	USUS0000,US00US00	
	FSV/SURV	USUS0000,US00US00	
	PB/FTE-3	USUS0000,US00US00	
	PB/FTE-4	USUS0000,US00US00	
	PB/FTE-6	USUS0000,US00US00	
	PB/FTE-14	USUS0000,US00US00	
	PB/FTE-15	USUS0000,US00US00	
	PB/FTE-16	USUS0000,US00US00	
	AVR/1&2	USUS0000,US00EU00	
	P13P	USUS0000,US00US00	
	P13Q	USUS0000,US00US00	
	P13R/S	USUS0000,US00US00	
	P13T	USUS0000,US00US00	
	P13V	USUS0000,US00US00	
	HB-2	USUS0000,US00US00	
	HRB-14/15A	USUS0000,US00US00	
	GF-3	USUS0000,US00EU00	
	DR-GB2	USUS0000,US00EU00	
	Thorium	US000000	
	^{238}Pu (US)	USUSUS00	
	^{238}Pu (AVR/GM)	USUSEU00	
	^{238}Pu (SILOE/FR)	USUSEU00	
	^{238}Pu (DRAGON/UK)	USUSEU00	
9. Ownership of accountable nuclear material	The accountable nuclear material, contained in the subject HTGR Can is Government-Owned.		Reference 2

FUEL RSD FORM
for HTGR ASSEMBLED CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary		References (See Table 2)
II. SPENT NUCLEAR FUEL DATA			
1. Reactor name and reactor name abbreviation or fuel description and component	Note: The fuel being shipped is not reactor core fuel. The fuel being shipped was research and development test fuel. The reactors identified below were used solely as a source of neutrons in order to test the integrity/performance of the various fuel items.		Reference 2
	Fuel Test Capsule/ Element Identity	Reactor Name/Abbreviation	
	FSV/FTE-2	Fort St. Vrain/FSV	
	FSV/SURV	Fort St. Vrain/FSV	
	PB/FTE-3	Peach Bottom/PB	
	PB/FTE-4	Peach Bottom/PB	
	PB/FTE-6	Peach Bottom/PB	
	PB/FTE-14	Peach Bottom/PB	
	PB/FTE-15	Peach Bottom/PB	
	PB/FTE-16	Peach Bottom/PB	
	AVR/1&2	Arbeisgemeinshaft Versuchs-Reaktor (Germany/AVR	
	P13P	Engineering Test Reactor at Idaho Falls/ETR	
	P13Q	Oak Ridge 30MW(t) Research Reactor/ORR	
	P13R/S	General Electric Test Reactor /GETR	
	P13T	Oak Ridge 30 MW(t) Research Reactor/ORR	
	P13V	General Electric Test Reactor/GETR	
	HB-2	General Electric Test Reactor/GETR	
	HRB-14/15A	High-Flux Irradiation Reactor (ORNL)/HFIR	
	GF-3	Siloe Test Reactor Grenoble France/SILOE	
	DR-GB2	Dragon Test Reactor (England) /DRAGON	

FUEL RSD FORM
for HTGR ASSEMBLED CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 2)
2. Core information	Core identification or designation: See Above	N/A
	Initial criticality date: The core data is not related to the fuel being shipped	N/A
	Shutdown date: The core data is not related to the fuel being shipped	N/A
	Number of elements or assemblies: The core data is not related to the fuel being shipped	N/A
	Neutron flux: The core data is not related to the fuel being shipped	N/A
	Neutron lifetime: Unknown	Not Available
	Significant events in its operating history: There were no significant events reported in the referenced documents for each of the irradiated fuel units.	References 5-1 through 5-28 located in Reference 2
3. General fuel description with a description of how the fuel was fabricated	Note: The fuel being shipped is not reactor core fuel. The fuel being shipped was test fuel. The identified reactor was used solely as a source of neutrons in order to test the integrity/ performance of the various fuel items. **Fuel Particles**: 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. These kernels were used in the Fuel Particles (Coatings) which are 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). **Fuel Compacts**: 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, and graphite powder. **Fuel Pebbles:** 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.	Table 2-1 of Reference 2

Description	Shipper Summary	References (See Table 2)
4. Fuel unit (rod, plate, assembly, etc.)	Fuel units consist of rods, particles, and pebbles.	Table 2-1 of Reference 2
5. Fuel unit specific data each fuel unit, include as Table 1 or equivalent	Unique identification number: Because the fuel pieces are test capsules, element sections, fuel particles and fuel pebbles, there is no unique identification number stamped on each fuel unit. The reference numbers for each fuel unit is shown in Table 1. The unique identification for the assembled HTGR Can, (which contains the fuel pieces, test capsules, element sections, fuel particles, compacts, and pebbles), is #032231.	Tables 6.1 through 6.19 of Reference 2
	Pre-irradiation isotopic composition or reportable NM (element and isotope weight): Eighteen fuel units with a range of U isotopes from 2.4E-01 g to 5.01E+01 g and a range of U-235 of from 2.24E-01 g to 4.67E+01 g. Exact data for each fuel unit is given in Table 1.	Section 5 of Reference 2
	Post-irradiation isotopic composition or reportable NM (element and isotope weight): Eighteen fuel units with combined isotopes of U-233 (3.07E+01 g), U-234 (5.05E+00 g), U-235 (1.08E+02 g), U-236 (1.66E+01 g), & U-238 (1.16E+01 g) and Pu-238 (1.61E-01 g), Pu-239 (2.75E-01 g), Pu-240 (8.31E-02 g), Pu-241 (1.96E-02 g), Pu-242 (2.77E-02 g). See Table 1 for individual fuel unit isotopic composition.	Tables 6.1 through 6.19 of Reference 2 updated to 09/30/03 and included in Table 1
	Time in reactor: Various. From 133 EFPD to 985 EFPD. See Table 1 for details on each fuel unit.	Section 5 of Reference 2
	Reactor power level: Various. From 8 MW(th) to 842 MW(th). See Table 1 for details on each fuel unit.	References 5-1 through 5-28 of Reference 2
	Cooling time (discharge date from reactor or reactor shutdown date): Various: From 19 yrs. 9 mos. to 30 yrs. 3 mos. as of 09/30/03. See Table 1 for details on each fuel unit.	Reference 2 updated to 09/30/03 and included in Table 1
	Percent estimated burnup: Various: From 10.7% to 95.9%. See Table 1 for details on each fuel unit.	Reference 2

Description	Shipper Summary	References (See Table 2)
	Heat decay (watts): Various: As of 09/30/03, it will range from 0.0035 W to 0.440 W. See Table 1 for details on each fuel unit.	Reference 2 updated to 09/30/03 and included in Table 1
	Date of post-irradiation data: 9/30/03	Reference 2 updated to 09/30/03 and included in Table 1
6. Description of the distribution of fissile material:		
a. Description of fissile material at beginning-of-life (BOL) over the length of the fuel.	The distribution of the fissile material within the HTGR Can is unknown. The HTGR Can is packed solid with the different fuel units described in Table 1. A schematic of the fuel loading is shown in Fig. 6.3 of Reference 2.	Reference 2, Figure 6.3
b. As shipped (g/linear foot). Provide sufficient information to support criticality evaluation.	There is approximately 172 g U within the HTGR Can (including U-238) and 0.57 g Pu . The g/linear foot is unknown. It is assumed that the distribution of fissile material is homogeneous.	Reference 2 updated to 09/30/03 and included in Table 1
7. Detailed drawings (most current drawings, indicating revision number and date) and list of materials. Describe any deviations from the drawings. Include documentation providing traceability from the drawings to the fuel elements to be shipped. The following information shall be obtainable from the drawings and listed below:	GA did not receive drawings of the DOE-owned individual fuel units within the HTGR Can therefore GA cannot provide the drawings. Most of these fuel units are in the DOE inventory and the INEEL may have access to or may have the drawings for these fuels.	Reference 2

Description	Shipper Summary	References (See Table 2)
a. Physical description of each type of fuel unit:	The fuel particle kernels consist of high-temperature sintered fully-densified ceramic spherical kernel substrate, composed of UC_2, UCO, UO_2, $(Th,U)C_2$, or $(Th,U)O_2$. The fuel particle coatings consist of solid, spherical, isotropic, discrete multi-layered coatings around the fuel kernel. The chemical composition of the coatings includes pyrolytic-carbon (PyC) & silicon carbide (SiC). The fuel compacts or rods consist of multi-coated, ceramic fuel particles, bound in solid, cylindrical, injection-molded, high-temperature heat-treated compacts. The fuel compact bonding matrix is composed of carbonized graphite shim, coke, & graphite powder. The fuel pebbles consist of 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.	Tables 2.1, 2.2, & 2.6 of Reference 2
b. Total length:	The fuel kernels are of spherical geometry Approx. 200 to 1000 m kernel dia. Approx. 720 m dia. (typ) The fuel particle coatings are of spherical geometry; Approx. 250 to 1000 m dia. Approx. 20 to 80 m coating thickness. 3 to 5 coatings/particle (typ). The fuel compacts are of right-circular cylindrical geometry; Approx. 0.490" (1.25 cm) dia. Approx. 1.94" (4.93 cm) long. The fuel pebbles are of spherical geometry; Approx. 2.36" (6.00 cm) dia. Approx. 20,000 fuel particles/sphere.	Table 2.1 of Reference 2
c. Length of fueled portion:	The fuel particles range from 230 to 920 m in dia. The fuel compacts are homogeneous throughout the 4.93 cm length. The fuel pebbles are homogeneous throughout the 6.00 cm dia.	Table 2.1 of Reference 2

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Description	Shipper Summary	References (See Table 2)
d. Position of fueled portion with respect to a permanent reference point on the fuel unit:	The distribution of the fueled portion within the HTGR Can is unknown. The HTGR Can is packed solid with the different fuel units described in Table 1. A schematic of the fuel loading is shown in Fig. 6.3 of Reference 2. It is assumed that the fuel distribution is linear throughout the HTGR Can.	Fig. 6.3 of Reference 2
e. Cross-sectional dimensions	The fuel kernels are of spherical geometry Approx. 200 to 1000 m kernel dia. Approx. 720 m dia. (typ) The fuel particle coatings are of spherical geometry; Approx. 250 to 1000 m dia. Approx. 20 to 80 m coating thickness. 3 to 5 coatings/particle (typ). The fuel compacts are of right-circular cylindrical geometry; Approx. 0.490" (1.25 cm) dia. Approx. 1.94" (4.93 cm) long. The fuel pebbles are of spherical geometry; Approx. 2.36" (6.00 cm) dia. Approx. 20000 fuel particles/sphere.	Table 2.1 of Reference 2
f. Shape (plates, rod, etc.):	Various; coated fuel particles and pebbles are spheres; fuel compacts are right circular cylindrical rods.	Table 2.1 of Reference 2
g. Plenum spacers or springs:	None	N/A
h. Configuration of FHU as shipped, including cuts made to prepare for shipping, shipment configuration (assembly, loose plates, subassembly, etc.):	The FHU will consist of two (2) Cans. One consisting of HTGR fuel and the other RERTR fuel. This section will describe the HTGR Can which consists of particles, compacts, and pebbles, packed solid inside a welded Primary Enclosure, which is in turn contained inside a welded Secondary Enclosure. No cuts were made for shipping. The Fuel RSD form for the RERTR Can will describe the configuration for the RERTR fuel.	Figs. 6.1 thru 6.3 of Reference 2

Description	Shipper Summary	References (See Table 2)
i. Fuel particle size and composition:	The fuel kernels are of spherical geometry, and are composed of UC_2, UCO, UO_2, $(Th,U)C_2$, or $(Th,U)O_2$. Approx. 200 to 1000 m kernel dia. Approx. 720 m dia. (typ) The fuel particle coatings are of spherical geometry and are composed of pyrolytic carbon or silicon carbide. Approx. 250 to 1000 m dia. Approx. 20 to 80 m coating thickness. 3 to 5 coatings/particle (typ). The fuel compacts are of right-circular cylindrical geometry, and are composed of coated particles bonded with carbonized graphite matrix. Approx. 0.490" (1.25 cm) dia. Approx. 1.94" (4.93 cm) long. The fuel pebbles are of spherical geometry, and are composed of coated fuel particles bonded with carbonized graphite matrix. Approx. 2.36" (6.00 cm) dia. Approx. 20,000 fuel particles/sphere.	Table 2-1 of Reference 2
j. Fuel matrix composition:	The fuel matrix is composed of $(Th,U)C_2$, $(Th,U)O_2$ UC_2, UCO, UO_2, $(Th,Pu)C_2$, $(Th,Pu)O_2$ PuC_2, PuCO, PuO_2. The binding matrix for both fuel compacts and fuel pebbles is composed of carbonized graphite shim, coke, and graphite powder.	Table 2-1 of Reference 2
8. As shipped dimensions (cm):		
a. Cut length:	Not applicable; the fuel as shipped is not cut. Dimensions same as 7b, above.	Table 2-1 of Reference 2
b. Cross-sectional dimensions:	Cross-sectional dimensions same as 7e, above.	Table 2-1 of Reference 2

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Description	Shipper Summary	References (See Table 2)
9. Composition and weight (g/fuel unit) of material in the fuel as shipped:		
a. Total:	The total weight of material in the fuel being shipped in the HTGR Can is 10,668 g. This consists of: 7.08E+03 g Carbon 1.91E+03 g Thorium 1.41E+03 g Silicon 1.72E+02 g Uranium Isotopes consisting of 1.16E+01 g U-238 3.07E+01 g U-233 5.05E+00 g U-234 1.08E+02 g U-235 1.66E+01 g U-236 2.24E+01 g Oxygen 5.66E-01 g Plutonium Isotopes consisting of 1.61E-01 g Pu-238 2.75E-01 g Pu-239 8.31E-02 g Pu-240 1.96E-02 g Pu-241 2.77E-02 g Pu-242 8.06E+01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the FSV/FTE-2 is 9.75E+01 g. This consists of: 5.38E+01 g Carbon 3.04E+01 g Thorium 1.07E+01 g Silicon 1.37E+00 g Uranium Isotopes consisting of 8.00E-02 g U-238 5.00E-01 g U-233 5.19E-02 g U-234 5.71E-01 g U-235 1.63E-01 g U-236 1.70E-01 g Oxygen 3.64E-03 g Plutonium Isotopes consisting of 1.41E-03 g Pu-238 1.30E-03 g Pu-239 6.00E-04 g Pu-240 1.86E-04 g Pu-241 1.40E-04 g Pu-242	Reference 2 updated to 09/30/03 and included in Table 1

Description	Shipper Summary	References (See Table 2)
	1.02E+00 g fission products	
	The total weight of material being shipped in the FSV/SURV is 8.39E+02 g. This consists of: 4.15E+02 g Carbon 3.26E+02 g Thorium 8.25E+01 g Silicon 1.19E+01 g Uranium Isotopes consisting of 6.80E-01 g U-238 3.26E+00 g U-233 2.40E-01 g U-234 6.88E+00 g U-235 8.10E-01 g U-236 1.38E-02 g Plutonium Isotopes consisting of 1.50E-03 g Pu-238 8.90E-03 g Pu-239 2.80E-03 g Pu-240 4.27E-04 g Pu-241 1.70E-04 g Pu-242 4.12E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the PB/FTE-3 is 3.49E+02 g. This consists of: 2.54E+02 g Carbon 3.54E+01 g Thorium 5.05E+01 g Silicon 6.98E+00 g Uranium Isotopes consisting of 4.25E-01 g U-238 2.03E-01 g U-233 5.50E-02 g U-234 6.11E+00 g U-235 1.83E-01 g U-236 1.80E+00 g Oxygen 6.43E-03 g Plutonium Isotopes consisting of 6.11E-05 g Pu-238 5.90E-03 g Pu-239 4.40E-04 g Pu-240 2.34E-05 g Pu-241 2.30E-06 g Pu-242 5.39E-01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	The total weight of material being shipped in the PB/FTE-4 is 4.34E+02 g. This consists of: 3.11E+02 g Carbon 5.00E+01 g Thorium 6.18E+01 g Silicon 7.42E+00 g Uranium Isotopes consisting of 4.11E-01 g U-238 8.45E-01 g U-233 1.53E-01 g U-234 5.34E+00 g U-235 6.71E-01 g U-236 1.98E+00 g Oxygen 1.75E-02 g Plutonium Isotopes consisting of 3.10E-03 g Pu-238 1.10E-02 g Pu-239 2.70E-03 g Pu-240 5.10E-04 g Pu-241 2.30E-04 g Pu-242 2.66E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the PB/FTE-6 is 1.25E+03 g. This consists of: 9.25E+02 g Carbon 1.04E+02 g Thorium 1.84E+02 g Silicon 1.96E+01 g Uranium Isotopes consisting of 1.42E+00 g U-238 2.60E+00 g U-233 3.44E-01 g U-234 1.26E+01 g U-235 2.59E+00 g U-236 3.93E+00 g Oxygen 9.58E-02 g Plutonium Isotopes consisting of 2.82E-02 g Pu-238 4.80E-02 g Pu-239 1.30E-02 g Pu-240 3.86E-03 g Pu-241 2.70E-03 g Pu-242 1.13E+01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for HTGR ASSEMBLED CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 2)
	The total weight of material being shipped in the PB/FTE-14 is 2.20E+03 g. This consists of: 1.45E+03 g Carbon 4.15E+02 g Thorium 2.88E+02 g Silicon 3.87E+01 g Uranium Isotopes consisting of 1.93E+00 g U-238 4.54E+00 g U-233 7.20E-01 g U-234 2.92E+01 g U-235 2.31E+00 g U-236 6.08E+00 g Oxygen 6.10E-02 g Plutonium Isotopes consisting of 4.89E-03 g Pu-238 4.70E-02 g Pu-239 7.79E-03 g Pu-240 1.03E-03 g Pu-241 2.80E-04 g Pu-242 8.34E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the PB/FTE-15 is 1.94E+03 g. This consists of: 1.28E+03 g Carbon 3.55E+02 g Thorium 2.55E+02 g Silicon 3.24E+01 g Uranium Isotopes consisting of 1.66E+00 g U-238 5.75E+00 g U-233 7.62E-01 g U-234 2.13E+01 g U-235 2.93E+00 g U-236 5.56E+00 g Oxygen 8.17E-02 g Plutonium Isotopes consisting of 1.69E-02 g Pu-238 4.90E-02 g Pu-239 1.20E-02 g Pu-240 2.55E-03 g Pu-241 1.20E-03 g Pu-242 1.18E+01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	The total weight of material being shipped in the PB/FTE-16 is 2.49E+03 g. This consists of: 1.73E+03 g Carbon 3.51E+02 g Thorium 3.44E+02 g Silicon 3.82E+01 g Uranium Isotopes consisting of 2.23E+00 g U-238 6.86E+00 g U-233 1.06E+00 g U-234 2.33E+01 g U-235 4.76E+00 g U-236 1.17E-01 g Plutonium Isotopes consisting of 3.76E-02 g Pu-238 5.40E-02 g Pu-239 1.80E-02 g Pu-240 4.20E-03 g Pu-241 3.30E-03 g Pu-242 2.18E+01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the AVR/1&2 is 1.02E+02 g. This consists of: 7.42E+01 g Carbon 9.73E+00 g Thorium 1.48E+01 g Silicon 1.36E+00 g Uranium Isotopes consisting of 1.19E-01 g U-238 1.63E-01 g U-233 2.20E-02 g U-234 8.54E-01 g U-235 2.07E-01 g U-236 1.02E+00 g Oxygen 3.48E-03 g Plutonium Isotopes consisting of 1.03E-03 g Pu-238 1.40E-03 g Pu-239 7.99E-04 g Pu-240 1.17E-04 g Pu-241 1.30E-04 g Pu-242 1.05E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	The total weight of material being shipped in the P13P is 7.90E+01g. This consists of: 3.71E+01 g Carbon 3.15E+01 g Thorium 7.38E+00 g Silicon 1.48E+00 g Uranium Isotopes consisting of 5.40E-02 g U-238 9.00E-01 g U-233 2.32E-01 g U-234 1.43E-01 g U-235 1.48E-01 g U-236 1.17E-01 g Oxygen 7.48E-03 g Plutonium Isotopes consisting of 4.70E-03 g Pu-238 1.50E-03 g Pu-239 6.79E-04 g Pu-240 1.45E-04 g Pu-241 4.50E-04 g Pu-242 1.46E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the P13Q is 1.34E+02 g. This consists of: 7.12E+01 g Carbon 4.31E+01 g Thorium 1.42E+01 g Silicon 1.91E+00 g Uranium Isotopes consisting of 1.03E-01 g U-238 1.08E+00 g U-233 2.78E-01 g U-234 1.66E-01 g U-235 2.88E-01 g U-236 2.25E-01 g Oxygen 1.70E-02 g Plutonium Isotopes consisting of 1.13E-02 g Pu-238 2.80E-03 g Pu-239 1.50E-03 g Pu-240 2.89E-04 g Pu-241 1.10E-03 g Pu-242 3.03E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	The total weight of material being shipped in the P13R/S is 6.28E+01 g. This consists of: 3.59E+01 g Carbon 1.75E+01 g Thorium 7.15E+00 g Silicon 8.67E-01 g Uranium Isotopes consisting of 5.29E-02 g U-238 4.65E-01 g U-233 1.27E-01 g U-234 7.56E-02 g U-235 1.47E-01 g U-236 1.14E-01 g Oxygen 7.30E-03 g Plutonium Isotopes consisting of 4.70E-03 g Pu-238 1.20E-03 g Pu-239 7.29E-04 g Pu-240 1.24E-04 g Pu-241 5.40E-04 g Pu-242 1.28E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the P13T is 3.29E+02 g. This consists of: 2.06E+02 g Carbon 6.90E+01 g Thorium 4.11E+01 g Silicon 3.83E+00 g Uranium Isotopes consisting of 3.00E-01 g U-238 1.86E+00 g U-233 5.92E-01 g U-234 2.28E-01 g U-235 8.50E-01 g U-236 6.53E-01 g Oxygen 5.07E-02 g Plutonium Isotopes consisting of 3.29E-02 g Pu-238 7.30E-03 g Pu-239 5.30E-03 g Pu-240 8.27E-04 g Pu-241 4.40E-03 g Pu-242 8.02E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	The total weight of material being shipped in the P13V is 1.30E+02 g. This consists of: 7.42E+01 g Carbon 3.61E+01 g Thorium 1.48E+01 g Silicon 1.83E+00 g Uranium Isotopes consisting of 1.10E-01 g U-238 9.79E-01 g U-233 2.56E-01 g U-234 1.79E-01 g U-235 3.02E-01 g U-236 2.35E-01 g Oxygen 1.35E-02 g Plutonium Isotopes consisting of 8.47E-03 g Pu-238 2.40E-03 g Pu-239 1.40E-03 g Pu-240 2.62E-04 g Pu-241 9.70E-04 g Pu-242 2.61E+00 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the HB-2 is 1.39E+01 g. This consists of: 9.27E+00 g Carbon 2.39E+00 g Thorium 1.85E+00 g Silicon 1.79E-01 g Uranium Isotopes consisting of 1.50E-02 g U-238 5.10E-02 g U-233 8.00E-03 g U-234 7.40E-02 g U-235 3.10E-02 g U-236 2.94E-02 g Oxygen 5.70E-04 g Plutonium Isotopes consisting of 2.26E-04 g Pu-238 1.80E-04 g Pu-239 1.10E-04 g Pu-240 2.07E-05 g Pu-241 3.40E-05 g Pu-242 1.71E-01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	The total weight of material being shipped in the HRB-14/15A is 1.32E+02 g. This consists of: 9.67E+01 g Carbon 1.20E+01 g Thorium 1.93E+01 g Silicon 2.59E+00 g Uranium Isotopes consisting of 1.96E+00 g U-238 3.75E-01 g U-233 1.01E-01 g U-234 7.40E-02 g U-235 8.20E-02 g U-236 3.06E-01 g Oxygen 6.70E-02 g Plutonium Isotopes consisting of 3.01E-03 g Pu-238 3.20E-03 g Pu-239 1.50E-02 g Pu-240 5.03E-03 g Pu-241 1.20E-02 g Pu-242 9.41E-01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the GF-3 is 1.86E+01 g. This consists of: 8.31E+00 g Carbon 7.98E+00 g Thorium 1.65E+00 g Silicon 3.25E-01 g Uranium Isotopes consisting of 1.20E-02 g U-238 2.01E-01 g U-233 3.80E-02 g U-234 4.20E-02 g U-235 3.20E-02 g U-236 2.63E-02 g Oxygen 1.51E-03 g Plutonium Isotopes consisting of 9.12E-04 g Pu-238 3.40E-04 g Pu-239 1.40E-04 g Pu-240 3.65E-05 g Pu-241 8.00E-05 g Pu-242 3.29E-01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	The total weight of material being shipped in the DR-GB2 is 6.99E+01 g. This consists of: 4.67E+01 g Carbon 1.24E+01 g Thorium 9.29E+00 g Silicon 1.20E+00 g Uranium Isotopes consisting of 7.70E-02 g U-238 8.60E-02 g U-233 1.10E-02 g U-234 9.75E-01 g U-235 5.60E-02 g U-236 1.48E-01 g Oxygen 9.83E-04 g Plutonium Isotopes consisting of 3.01E-05 g Pu-238 7.90E-04 g Pu-239 1.50E-04 g Pu-240 1.10E-05 g Pu-241 2.50E-06 g Pu-242 1.45E-01 g fission products	Reference 2 updated to 09/30/03 and included in Table 1
b. Fuel (chemical form of uranium and plutonium):	Various: Physical forms are: $(Th,U)C_2$, $(Th,U)O_2$ UC_2, UCO, UO_2, $(Th,Pu)C_2$, $(Th,Pu)O_2$ PuC_2, PuCO, PuO_2..	Table 2.6 of Reference 2
	FSV/FTE-2 consists of: $(Th,U)C_2$, $(Th,Pu)C_2$ 1.08E+00 g Carbon 1.37E+00 g Uranium Isotopes 3.64E-03 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	FSV/SURV consists of: $(Th,U)C_2$, $(Th,Pu)C_2$ 2.49E+01 g Carbon 1.19E+01 g Uranium Isotopes 1.38E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	PB/FTE-3 consists of: $(Th,U)C_2$, UC_2, UO_2, $(Th,Pu)C_2$, PuC_2, PuO_2 7.62E+01 g Carbon 6.98E+00 g Uranium Isotopes 4.04E-01 g Oxygen 6.43E-03 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	PB/FTE-4 consists of: $(Th,U)C_2$, UC_2, UO_2, $(Th,Pu)C_2$, PuC_2, PuO_2 9.33E+01 g Carbon 7.42E+00 g Uranium Isotopes 4.93E-01 g Oxygen 1.75E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	PB/FTE-6 consists of: $(Th,U)C_2$, UC_2, UO_2, $(Th,Pu)C_2$, PuC_2, PuO_2 3.33E+02 g Carbon 1.96E+01 g Uranium Isotopes 1.50E+00 g Oxygen 9.58E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	PB/FTE-14 consists of: $(Th,U)O_2$, UC_2, UO_2, $(Th,Pu)C_2$, PuC_2, PuO_2 1.74E+02 g Carbon 3.87E+01 g Uranium Isotopes 2.29E+00 g Oxygen 6.10E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	PB/FTE-15 consists of: $(Th,U)O_2$, UC_2, UO_2, $(Th,Pu)C_2$, PuC_2, PuO_2 1.54E+02 g Carbon 3.24E+01 g Uranium Isotopes 2.06E+00 g Oxygen 8.17E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	PB/FTE-16 consists of: $(Th,U)C_2$, $(Th,Pu)C_2$ 2.60E+02 g Carbon 3.82E+01 g Uranium Isotopes 1.17E-01 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References (See Table 2)
	AVR/1&2 consists of: $(Th,U)C_2$, $(Th,U)O_2$, $(Th,Pu)C_2$, $(Th,Pu)O_2$ 4.45E+01 g Carbon 1.37E+00 g Uranium Isotopes 2.35E-01 g Oxygen 3.48E-03 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	P13P consists of: $(Th,U)O_2$, UC_2, UO_2, $(Th,Pu)O_2$, PuC_2, PuO_2 2.35E+00 g Carbon 1.48E+00 g Uranium Isotopes 1.17E-01 g Oxygen 7.48E-03 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	P13Q consists of: UC_2, PuC_2 6.76E+01 g Carbon 1.92E+00 g Uranium Isotopes 1.70E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	P13R/S consists of: UC_2, PuC_2 3.41E+01 g Carbon 8.68E-01 g Uranium Isotopes 7.30E-03 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	P13T consists of: UC_2, UCO, PuC_2, PuCO 1.96E+02 g Carbon 3.83E+00 g Uranium Isotopes 3.53E-01 g Oxygen 5.07E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	P13V consists of: UCO, PuCO 7.05E+01 g Carbon 1.83E+00 g Uranium Isotopes 1.35E-01 g Oxygen 1.35E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1

Description	Shipper Summary	References (See Table 2)
	HB-2 consists of: UCO, PuCO 8.81E+00 g Carbon 1.79E-01 g Uranium Isotopes 9.40E-03 g Oxygen 5.70E-04 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	HRB-14/15A consists of: UO_2, PuO_2 2.59E+00 g Uranium Isotopes 1.06E-01 g Oxygen 6.70E-02 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	GF-3 consists of: $(Th,U)O_2$, $(Th,Pu)O_2$ 3.25E-01 g Uranium Isotopes 6.30E-03 g Oxygen 1.51E-03 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
	DR-GB2 consists of: UC_2, PuC_2 4.44E+01 g Carbon 1.21E+00 g Uranium Isotopes 9.83E-04 g Plutonium Isotopes	Table 2.6 of Reference 2 updated to 09/30/03 and included in Table 1
c. Alloy or diluent in the matrix:	Fuel particles: P13P, P13Q, P13R/S, P13T, P13V, HB-2, HRB-14/15A, and GF-3, DR-GB2: Covered with various coatings which include pyrolytic-carbon and silicon carbide. Fuel compacts: FSV/FTE-2, FSV/SURV, PB/FTE-3, PB/FTE-4, PB/FTE-6, PB/FTE-14, PB/FTE-15, and PB/FTE-16: Contain fuel particles bound with carbonized graphite shim, coke, and graphite powder. Fuel pebbles: AVR/1&2: Contain fuel particles and bound with carbonized graphite shim, coke, and graphite powder.	Table 2.1 of Reference 2
	FSV/FTE-2 2.15E+01 g Carbon	Table 2.1 of Reference 2

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Description	Shipper Summary	References (See Table 2)
	1.07E+01 g Silicon	
	FSV/SURV	Table 2.1 of Reference 2
	1.66E+02 g Carbon 8.25E+01 g Silicon	
	PB/FTE-3	Table 2.1 of Reference 2
	1.02E+02 g Carbon 5.05E+01 g Silicon	
	PB/FTE-4	Table 2.1 of Reference 2
	1.24E+02 g Carbon 6.18E+01 g Silicon	
	PB/FTE-6	Table 2.1 of Reference 2
	3.70E+02 g Carbon 1.84E+02 g Silicon	
	PB/FTE-14	Table 2.1 of Reference 2
	5.80E+02 g Carbon 2.88E+02 g Silicon	
	PB/FTE-15	Table 2.1 of Reference 2
	5.12E+02 g Carbon 2.55E+02 g Silicon	
	PB/FTE-16	Table 2.1 of Reference 2
	6.92E+02 g Carbon 3.44E+02 g Silicon	
	AVR/1&2	Table 2.1 of Reference 2
	2.97E+01 g Carbon 1.48E+01 g Silicon	
	P13P	Table 2.1 of Reference 2
	1.51E+01 g Carbon 7.38E+00 g Silicon	
	P13Q	Table 2.1 of Reference 2
	2.91E+01 g Carbon 1.42E+01 g Silicon	
	P13R/S	Table 2.1 of Reference 2
	1.47E+01 g Carbon 7.15E+00 g Silicon	
	P13T	Table 2.1 of Reference 2
	8.43E+01 g Carbon 4.11E+01 g Silicon	
	P13V	Table 2.1 of Reference 2

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Description	Shipper Summary	References (See Table 2)
	3.03E+01 g Carbon 1.48E+01 g Silicon	
	HB-2 3.80E+00 g Carbon 1.85E+00 g Silicon	Table 2.1 of Reference 2
	HRB-14/15A 3.96E+01 g Carbon 1.93E+01 g Silicon	Table 2.1 of Reference 2
	GF-3 3.38E+00 g Carbon 1.65E+00 g Silicon	Table 2.1 of Reference 2
	DR-GB2 1.90E+01 g Carbon 9.29E+00 g Silicon	Table 2.1 of Reference 2
d. Cladding:	None	Reference 2
e. Any external coatings applied to the cladding:	N/A	N/A
f. Thermal transfer material (e.g., sodium):	None	Reference 2
g. Organic materials:	None	Reference 2
h. Special additives:	None	Reference 2
i. Chemically reactive materials (e.g., sodium):	None	Reference 2
j. Neutron poisons, fixed or burnable:	None	Reference 2
k. Other (specify):	None	Reference 2
10. Radiation level curve, at 3 feet in air, as a function of time out of the reactor.	No radiation levels were measured on the individual fuel items. The radiation level was only measured on the assembled HTGR Can. As of 12/1/95 the radiation levels taken at approximately one meter varied from a measured low of 58 R/hr at the top to a high of 100 R/hr at the middle. The bottom measured 80 R/hr. Based on the decay rate from 12/1/95 to 09/30/03, the levels were **calculated** to be: 48 R/hr at the top, 82 R/hr at the middle, and 66 R/hr at the bottom. Based on the radiation levels calculated on the assembled RERTR Can (See Reference 16), the combined radiation level will be 200 R/hr at the top, 407 R/hr at the middle, and 326 R/hr at the bottom.	Table 6.21 of Reference 2 updated to 09/30/03 and Reference 16.
11. Description of fuel element/cladding degradation, including where and how it occurred:	There is no cladding on the enclosed HTGR type fuels. They are all graphite or ceramic coated particles, fuel compacts, or pebbles. There was no fuel element degradation.	Reference 2
a. Description of fuel element/cladding degradation, including where and how it occurred.	N/A	N/A

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Description	Shipper Summary	References (See Table 2)
b. Is there any reason to suspect the fuel cladding has been damaged to the extent that it would no longer retain fission products? If so, state reasons.	N/A	N/A
c. Reactor and storage history. Note only conditions that might affect storage or future cladding integrity.	N/A	N/A
d. Describe exposure to contaminants. (Hg, halides, etc.)	None	Reference 2
i. Was the contamination removed?	N/A	N/A
ii. How was the contamination removed?	N/A	N/A
12. Activation products and curies.	None	Tables 6.1 through 6.18 of Reference 2
13. Metallurgical state of the cladding (annealed, stressed, sensitized, estimated exposure [nvt], etc.)	There is no cladding on the fuel.	Reference 2
14. Describe measured or projected external contamination, particularly alpha contamination levels.	The external contamination of each of the fuel items was not measured. The contamination was only measured on the exterior of the HTGR Can. The results as of 12/1/95 were: 3000 dpm/100 cm^2 beta+gamma 14 dpm/100 cm^2 alpha	Reference 2

Prepared By:

Printed Name/Title	Signature	Date
J. S. Greenwood Manager, TRIGA Reactors Facility	J. S. Greenwood	7/7/03

Approved By:

(Shipper Management) Printed Name/Title	Signature	Date
R. I. De Velasco Manager, Decommissioning Projects	[signature]	7/7/03

FUEL RSD FORM
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Table 1 - Fuel Units and HTGR Can Specific Data
(U includes the isotope U-238)

Can	Fuel Unit	Pre-Irradiation		Post-Irradiation as of September 30, 2003															
Unique ID Number	Unique ID Number	U (g)	U-235 (g)	U (g)	U-233 (g)	U-234 (g)	U-235 (g)	U-236 (g)	Pu (g)	Pu-238 (g)	Pu-239 (g)	Pu-240 (g)	Pu-241 (g)	Pu-242 (g)	Time in Reactor	Power Level	Cooling Time/Date	Percent Burnup	Decay Heat (watts)
032231	FSV/FTE-2	1.56E+00	1.45E+00	1.37E+00	5.00E-01	5.19E-02	5.71E-01	1.63E-01	3.64E-03	1.41E-03	1.30E-03	6.00E-04	1.86E-04	1.40E-04	488.8 EFPD	842MW(th)	19yrs 9 mo 09/30/03	60.6%	2.46E-02
032231	FSV/SURV	1.20E+01	1.12E+01	1.19E+01	3.26E+00	2.40E-01	6.88E+00	8.10E-01	1.38E-02	1.50E-03	8.90E-03	2.80E-03	4.27E-04	1.70E-04	174.0 EFPD	842MW(th)	24rs 8mo 09/30/03	38.6%	8.96E-02
032231	PB/FTE-3	7.35E+00	6.84E+00	6.98E+00	2.03E-01	5.50E-02	6.11E+00	1.83E-01	6.43E-03	6.11E-05	5.90E-03	4.40E-04	2.34E-05	2.30E-06	133.0 EFPD	110 MW(th)	31 yrs 9mo 09/30/03	10.7%	1.05E-02
032231	PB/FTE-4	8.99E+00	8.37E+00	7.42E+00	8.45E-01	1.53E-01	5.34E+00	6.71E-01	1.75E-02	3.10E-03	1.10E-02	2.70E-03	5.10E-04	2.30E-04	448.8 EFPD	110 MW(th)	30 yrs 09/30/03	36.2%	5.09E-02
032231	PB/FTE-6	2.68E+01	2.49E+01	1.96E+01	2.60E+00	3.44E-01	1.26E+01	2.59E+00	9.58E-02	2.82E-02	4.80E-02	1.30E-02	3.86E-03	2.70E-03	644.9 EFPD	110 MW(th)	28 yrs 10 mo 09/30/03	49.4%	2.25E-01
032231	PB/FTE-14	4.19E+01	3.91E+01	3.87E+01	4.54E+00	7.20E-01	2.92E+01	2.31E+00	6.10E-02	4.89E-03	4.70E-02	7.79E-03	1.03E-03	2.80E-04	315.8 EFPD	110 MW(th)	30 yrs 09/30/03	25.3%	1.65E-01
032231	PB/FTE-15	3.71E+01	3.46E+01	3.24E+01	5.75E+00	7.62E-01	2.13E+01	2.93E+00	8.17E-02	1.69E-02	4.90E-02	1.20E-02	2.55E-03	1.20E-03	511.9 EFPD	110 MW(th)	28 yrs 11 mo 09/30/03	38.4%	2.47E-01
032231	PB/FTE-16	5.01E+01	4.67E+01	3.82E+01	6.86E+00	1.06E+00	2.33E+01	4.76E+00	1.17E-01	3.76E-02	5.40E-02	1.80E-02	4.20E-03	3.30E-03	511.9 EFPD	110 MW(th)	28 yrs 11 mo 09/30/03	50.1%	4.40E-01
032231	AVR/1&2	2.15E+00	2.00E+00	1.37E+00	1.63E-01	2.20E-02	8.54E-01	2.07E-01	3.48E-03	1.03E-03	1.40E-03	7.99E-04	1.17E-04	1.30E-04	985.0 EFPD	46 MwWth)	30 yrs 3 mo 09/30/03	57.3%	1.93E-02
032231	P13P	1.07E+00	1.00E+00	1.48E+00	9.00E-01	2.32E-01	1.43E-01	1.48E-01	7.48E-03	4.70E-03	1.50E-03	6.79E-04	1.45E-04	4.50E-04	181.0 EFPD	175MW(th)	29 yrs 3 mo 09/30/03	85.7%	3.31E-02
032231	P13Q	2.06E+00	1.92E+00	1.92E+00	1.08E+00	2.78E-01	1.66E-01	2.88E-01	1.70E-02	1.13E-02	2.80E-03	1.50E-03	2.89E-04	1.10E-03	391.7 EFPD	30 MW(th)	28 yrs 7 mo 09/30/03	91.4%	6.86E-02
032231	P13R/S	1.04E+00	9.69E-01	8.68E-01	4.65E-01	1.27E-01	7.56E-02	1.47E-01	7.30E-03	4.70E-03	1.20E-03	7.29E-04	1.24E-04	5.40E-04	258.0 EFPD	50 MW(th)	28 yrs 9 mo 09/30/03	92.2%	2.99E-02
032231	P13T	5.97E+00	5.57E+00	3.83E+00	1.86E+00	5.92E-01	2.28E-01	8.50E-01	5.07E-02	3.29E-02	7.30E-03	5.30E-03	8.27E-04	4.40E-03	363.4 EFPD	50 MW(th)	27 yrs 3 mo 09/30/03	95.9%	1.87E-01
032231	P13V	2.15E+00	2.00E+00	1.83E+00	9.79E-01	2.56E-01	1.79E-01	3.02E-01	1.35E-02	8.47E-03	2.40E-03	1.40E-03	2.62E-04	9.70E-04	205.0 EFPD	50 MW(th)	26 yrs 11 mo 09/30/03	91.1%	5.93E-02
032231	HB-2	2.68E-01	2.50E-01	1.79E-01	5.10E-02	8.00E-03	7.40E-02	3.10E-02	5.70E-04	2.26E-04	1.80E-04	1.10E-04	2.07E-05	3.40E-05	123.0 EFPD	50 MW(th)	27 yrs 9 mo 09/30/03	70.4%	3.53E-03
032231	HRB-14/15A	2.80E+00	5.70E-01	2.59E+00	3.75E-01	1.01E-01	7.40E-02	8.20E-02	6.70E-02	3.01E-03	3.20E-02	1.50E-02	5.03E-03	1.20E-02	173.8 EFPD	100 MW(th)	22 yrs 8 mo 09/30/03	87.0%	2.47E-02
032231	GF-3	2.40E-01	2.24E-01	3.25E-01	2.01E-01	3.80E-02	4.20E-02	3.20E-02	1.51E-03	9.12E-04	3.40E-04	1.40E-04	3.65E-05	8.00E-05	363.0 EFPD	8 MW(th)	28 yrs 2 mo 09/30/03	81.3%	7.75E-03
032231	DR-GB2	1.35E+00	1.26E+00	1.21E+00	8.60E-02	1.10E-02	9.75E-01	5.60E-02	9.83E-04	3.01E-05	7.90E-04	1.50E-04	1.10E-05	2.50E-06	140.0 EFPD	20 MW(th)	27 yrs 10 mo 09/30/03	22.6%	4.69E-03
032231	Total	2.05E+02	1.89E+02	1.72E+02	3.07E+01	5.05E+00	1.08E+02	1.66E+01	5.66E-01	1.61E-01	2.75E-01	8.31E-02	1.96E-02	2.77E-02	Various	Various	Various 09/30/03	Various	1.69E+00

EFPD = Effective Full Power Days

PC-000512/0

FUEL RSD FORM
for HTGR ASSEMBLED CAN # 032231 for FHU # 315-391-120-44-001

Table 2 - Shipper References

Reference Number	Type (Drawing, Letter, etc.)	Title	Number	Revision/Date	Quality Assurance [1]
1	DOE Contract	"General Atomics Hot Cell Decontamination & Decommissioning Project", Phases 2 & 3 Activities; Statement of Work, Task 16" , Attachment I to USDOE Contract DE-AC03-95SF20798	Contract DE-AC03-95SF20798 (Note: This is Reference 1-1 called out in Reference 2 below)		Non QA Record
2	General Atomics Report	HTGR/RERTR Fuel Materials Characterization and Packaging Report	PC-000384	Rev.2, 4/02	QA Record Level NA
3	General Atomics QA Inspection Report	GA Hot Cell D&D Project Final Assessment Report, Transfer of HTGR/RERTR Fuel from Hot Cell to Bldg. 30 (Note: This is Reference 1-3 called out in Reference 2 above.)	16A	3/28/96	QA Record Level 1
4	General Atomics Internal Correspondence	J. S. Greenwood to C. M. Miller, "Calculation of Pressure Buildup in Fuel Storage Enclosures", 1/15/96	HCI:015:JSG:96	1/15/96	QA Record Level NA
5	General Atomics Internal Correspondence	Malakhof, V., "Nuclear Safety Evaluation of the Irradiated Fuel Material Interim Storage Facility at General Atomics"	NS:94:VM:399 Nuclear Safety File No. 533.0	6/95	QA Record Level NA
6	General Atomics Quality Assurance Manual	Quality Assurance Manual, General Atomics, San Diego, CA		3rd Edition, Revision D, 8/12/96	QA Record Level NA
7	General Atomics Internal Correspondence	Nicolayeff, V., "Hot Cell Irradiated Fuel Materials, INEEL Audits of General Atomics"	123:VN:02:18	5/15/02	QA Record Level NA
8	INEEL Correspondence	INEEL Procurement Quality Manager to J. Razvi, GA TRIGA Director, "Supplier Evaluation (TRIGA International)"	MTW-009-01	11/29/00	QA Record Level NA
9	Drawing	NAC International, Top Module, General Atomics IFM, NAC-LWT Cask	315-391-120	Rev. 2/ 5/8/03	QA Record Level A
10	Calculation Package	NAC International, Criticality Evaluation for the NAC-LWT with GA Irradiated Fuel Material (IFM)	14661-600	Rev. 0/ 12/16/02	QA Record Level A
11	Drawing	NAC International, Spacer, General Atomics IFM, LWT Cask	315-391-123	Rev. 1/ 5/8/03	QA Record Level A
12	Calculation Package	NAC International, Structural and Thermal Evaluation of the Top Module and Spacer	14661-200	Rev. 0/ 12/18/02	QA Record Level A
13	DOE/NRC Form	Certificate of Compliance for Radioactive Material Packages	Form 618 Certificate # 9225	Rev. 31/ (Later)	QA Record Level A

FUEL RSD FORM
for HTGR ASSEMBLED CAN # 032231 for FHU # 315-391-120-44-001

Reference Number	Type (Drawing, Letter, etc.)	Title	Number	Revision/Date	Quality Assurance [1]
14	Report	Safety Analysis Report for the NAC Legal Weight Truck Cask	Docket No. 71-9225 T-88004	Rev. LWT-03A 02/03	QA Record Level A
15	DOE/NRC Form	USDOE & USNRC Nuclear Material Transaction Report	DP-741	04/03	QA Record Level 1
16	Form	Fuel RSD Form for the Assembled RERTR Can #32230 for FHU # 315-391-120-44-001	434.28	NR/ 06/09/03	QA Record Level 1
17	Memo	Hot Cell IFM - HTGR/RERTR Enclosure Pressure Calculation	123:VN:03:08	Rev. 0/ 03/19/03	QA Record Level 1
18	Procedure	NAC-LWT Cask Generic Operating Procedure	NAC 315-P-02	Rev.11/ 6/13/03	QA Record Level A

[1] Describe quality level of reference. State "Non-QA Record" if not a quality assurance record.

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
I. SHIPPER AND SHIPMENT IDENTIFICATION INFORMATION		
1. Applicable Fuel RSD Form for this Packaging RSD	Fuel RSD Form for Assembled HTGR Can #032231 for FHU # 315-391-120-44-001. This Packaging RSD Form includes a revised list of References from the HTGR Fuel RSD Forms and is included as Table 3 at the end of this document.	
III. SNF PACKAGING		
1. FHU description.	The FHU is the Top Module (NAC Drawing #315-391-120) supplied by NAC. It is a right circular cylinder in shape 43.7" (111 cm) long by 13.265" (33.69 cm) in diameter fabricated out of 304 SS pieces. The Top Module consists of three (3) support plates, two (2) tubes to contain the HTGR and RERTR Cans, four (4) guide bars (made from plate) to align the support plates, four (4) grapple lift plates to allow the Top Module to be lifted, two (2) spacer plates to space the grapple lift plates, and two (2) guide plates to orient the tubes (Ref. 9). Inserted in the FHU are two (2) Can Spacers and the two (2) Cans.	Reference 9
	Can Spacers: (NAC Drawing #315-391-120, part 97 [HTGR] and #315-391-120, part 98 [RERTR]), are fabricated out of 4" (10.16 cm) dia. 304 SS pipe. Welded at the top and bottom of the pipe are 1/4" (0.64 cm) thick 304SS plates. These Can Spacers are used to assure that the Cans are at the same elevation at the top of the Top Module to minimize movement during shipping.	References 23 and 24
	Cans: One for the HTGR SNF #032231 (the subject of this packaging RSD) and the other for the RERTR SNF #032230 (see RERTR Packaging RSD Form for details of this can). The visible portion of the HTGR Can is the Secondary Enclosure. It is a right circular cylinder constructed of 304 SS tubing; 38.80" (98.55 cm) tall, 0.12" (0.3 cm) wall thickness and 5.25" (13.34 cm) dia. End caps constructed of Inconel 600 rolled plate 0.5" (1.27 cm) thick are seal welded to the tube. Inconel 625 weld material was used to weld the anchors of the lifting bail onto the top end cap.	Fig. 6.3 of Reference 2

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
1. FHU description (Cont.).	Cans (cont.) A stainless steel lifting bail is permanently attached to the top of the Can for lifting into the FHU. The lifting bail is crimped together using a copper end sleeve. The overall height of the Can is 39.05" (99.15 cm), with the lifting bail in the down position. Sealed inside of the Secondary Enclosure Can is the Primary Enclosure Can consisting of another right circular cylinder constructed of 304 SS tubing, 37.5" (95.25 cm) tall, 0.12" (0.3 cm) wall thickness and 4.75" (12.07 cm) dia. End caps constructed of Inconel 600 rolled plate 0.5" (1.27 cm) thick are seal welded to both ends of the tube. Inside the Primary Enclosure are the collected HTGR fuel units.	Fig. 6.3 of Reference 2
2. Method for the identification of each FHU	The FHU is identified with steel stamped letters, 0.03" (0.08 cm) deep, filled with black weather resistant paint. The HTGR Can is identified with stamped ¼" (0.64 cm) high characters on the top end cap and stencil painted 1" (2.54 cm) high characters along the side of the Can.	Reference 9 Fig. 6.3 of Reference 2
3. FHU specific data for each FHU, include as Table 1, updated with unique FHU identification number that identifies which fuel unit(s) are in each FHU, or equivalent.	See Table 1 of the Fuel RSD Form for HTGR Can #032231 for details of the fuel units that are in the Can.	Reference 2
4. Method for fuel unit/FHU location identification in the SNF packaging if the fuel is shipped and stored in a can, liner, basket, or other container.	Individual fuel units are intermingled within the Primary Enclosure. Details are given in Table 1 of the Fuel RSD Form for HTGR Can #032231. There is no method of identification for each of the fuel units within the enclosure. They were identified as they were removed from their storage unit at the GA Hot Cell Facility and duly noted by the Material Custodian and Quality Assurance Inspector. The primary enclosure was inserted in the secondary enclosure which is identified with a unique number stamped on the top and painted on the side.	Reference 2
5. Detailed drawings (most current drawings, indicating revision number and date) and list of materials. Describe any deviations from the drawings.	NAC International Drawing No. 315-391-120, Rev. 2, dated 5/8/03 GA Drawing No. 032231, Rev. A, dated 1/9/96 (Secondary Enclosure) GA Drawing No. 032237, Rev. B, dated 1/9/96 (Primary Enclosure) Deviations from the drawings of the Can were minor and did not affect its form, fit, or function. Deviations are described in Nonconformance Reports presented in Reference 3.	Reference 9 Fig 6.3 of Reference 2 Reference 3

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
	Deviations from the drawings of the Top Module and Can Spacers were minor and did not affect its form, fit, or function. The fabrication and test packages for the Top Module and Can Spacers is given in Reference 22.	Reference 22
a. Physical description of each type of SNF packaging:	See Shipper Summary to III, 1 above. There is only one SNF package.	Fig. 6.3 of Reference 2 and Reference 9
b. Materials of construction:	304 SS, Inconel 600, Inconel 625, Copper.	Figs 6.1 & 6.2 of Reference 2 and Reference 9
c. Dimensions (cm):		
i. Total length:	111.0 cm	Reference 9
ii. Cross-section dimensions:	33.69 cm	Reference 9
d. Weight (g):		
i. Empty:	82,327 g for the Top Module.	Reference 9
ii. Maximum loaded:	32,432 g for the HTGR Can and 34,474 g for the RERTR Can plus the weight of the Top Module and Spacers for a total loaded weight of 149,233 g.	References 2 and 9
6. List the extraneous material and its mass in grams. associated with the SNF packaging or its contents:		
a. Pyrophoric or reactive material	None	Reference 2
b. Inert materials	Graphite is the moderator used in HTGR fuels. Graphite and Silicon were used in the fabrication of the fuel particles, rods, and spheres.	Tables 2.1, 2.2, & 2.6 of Reference 2
c. Organic materials	None	Reference 2
d. Water		
i. As free water	None	Reference 2
ii. As associated water		
A. Chemically bound	None	Reference 2
B. Physically bound	None	Reference 2
e. Other (specify)	None	Reference 2

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description		Shipper Summary	References (See Table 3)
7.	Describe the following, including chemical composition:		
	a. Exposure to contaminants (Hg, halides, etc.). If any, how was the contamination removed?	There was no exposure to Hg, halides, etc. Confirmation of this is given in the material certifications from the vendors and presented in Reference 3.	Reference 3
	b. Any external coatings applied to the SNF	None	N/A
	c. Special additives	None	N/A
8.	Activation product contents and curies.	There are no activation products in the Top Module or the Primary and Secondary Enclosures since they were never exposed to neutrons. The enclosed SNF packages were exposed to neutrons, however since there is no cladding and the non-fissile materials consist of carbon, silicon, and oxygen, the activation products, if any, were undetectable.	Reference 2
9.	Metallurgical state of the SNF packaging (annealed, stressed, sensitized, estimated exposure [nvt], etc.)	The loaded Top Module, Spacers, and Primary and Secondary enclosures were not exposed to neutrons. The exposure (nvt) of the enclosed SNF was: FSV/FTE-2 1.90 E+21 fast FSV/SURV 1.00 E+21 fast PB/FTE-3 0.60 E+21 fast PB/FTE-4 1.98 E+21 fast PB/FTE-6 2.90 E+21 fast PB/FTE-14 1.50 E+21 fast PB/FTE-15 2.00 E+21 fast PB/FTE-16 2.30 E+21 fast AVR/1&2 3.00 E+21 fast P13P 8.10 E+21 fast P13Q 9.60 E+21 fast P13R/S 1.25 E+22 fast P13T 8.00 E+21 fast P13V 9.00 E+21 fast HB-2 5.00 E+21 fast HRB-14/15A 6.50 E+21 fast GF-3 1.00 E+21 fast DR-GB2 (Not recorded) The fuel items are all sealed in a 304 SS tube that was annealed and pickled.	Reference 2

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
10. Describe external contamination, particularly alpha contamination levels.	On the surface of the HTGR Can, the contamination levels were, as of 12/1/95: 3000 dpm/100 cm^2 + ; 14 dpm/100 cm^2 . The FHU Top Module and associated Can Spacers are new and clean and thus will not be contaminated prior loading at GA. However, it might become contaminated during loading. The maximum contamination level would be if all of the contamination from the RERTR and HTGR Cans is transferred to the Top Module. This maximum level would be 5,700 dpm/100 cm^2 - ; 27 dpm/100 cm^2 .	Reference 2

NOTE: The following subsections of Section III are applicable to canned fuel (cans, tubes, containers, etc.) only. If canned fuel is stored in an overpack canister (e.g., dual purpose canister), this section must be completed separately for both the fuel can and the overpack can.)

The following data applies to the fuel can (Primary Enclosure) GA Dwg # 032237/B		Fig. 6.1 of Reference 2
11. Provide a design analysis of the can to include:		
a. Identify the purpose of all penetrations	There are no penetrations.	Reference 2
b. Thermal analysis of can and contents	A thermal analysis of the can is not applicable. The temperature of the fuel units in the Can are at room temperature as is the FHU. There is currently only 1.69 watts of decay heat generated by the fuel items as of 09/30/03. This is easily dissipated by the Can.	Reference 2 updated to 09/30/03 and included in Table 1.
c. Integrity test results agreed upon by the INEEL	A helium leak test of the bottom end cap of the Can assembly per QDI LTH-S-3801 with an acceptable leak rate of "non-detected" (at a detection limit of 4.4E-08 standard cc/sec) was satisfactory. A hydro test of the tube at the fabrication facility of 800 psi minimum was performed and was satisfactory. After loading of the SNF into the primary closure the top end cap was placed on the tube inside the hot cell facility. The tube was mounted on a turn table with a variable rotational speed capability. The tube was rotated at a predetermined speed and the top end cap was welded remotely using an in-cell weld-head. The weld integrity was inspected using an in-cell mirror and an out-cell telescope for visual inspection. The weld was judged satisfactory by the QA inspector.	Reference 3

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
d. State the maximum allowable pressure on sealed can (e.g., could the can build up pressure?)	The pressure at which the weld at the end caps will begin to yield is 683 psi. For a safety factor of 2, the yield point is 341 psi. The ultimate pressure at which the ends caps will pop off is 1,367 psi. Based on a buildup rate of 1.62 E-04 psi/year and assuming a linear rise in pressure (actual buildup will decrease exponentially), the end cap will pop off in 8,400,000 years.	Reference 17
	Can buildup pressure is negligible (calculated to be 1.62 E-04 psi/year).	Reference 4
e. Drop test results, including extent of damage	Drop calculations have been performed for the loaded Top Module by BBWI for the IFSF Plant Safety Documents. This report is Reference 19. The analyzed lift height for the fuel basket or an individual package is 45" (114.3 cm.) over the top surface of the transfer car's steel slabs. The end cap of the secondary enclosure is severely damaged during most of the end drops and has significantly less containment integrity but the end cap will remain in place and provide a lesser level of containment. The damage to the primary enclosure is significantly less than the damage to the secondary enclosure. There is some material failure of the primary enclosure end cap (impacting end). This may reduce the level of confinement of the primary enclosure by a small amount, which is not defined because material failure does not develop into a gap from which contents could readily escape. The damage to the packages during a seismic event will be less than any of the analyzed postulated drops.	Reference 19
f. Other test reports	o Welder certifications were all satisfactory. o Weld integrity was satisfactory. o Transverse tension, flattening and reverse flattening tests satisfactory. o Flange and reverse bend tests satisfactory. o Yield strength confirmed. o Factory chemical and mechanical tests on tube and end cap materials were satisfactory.	Reference 3
g. Material certification papers for can and seal	Certification papers for the can and seal are provided in Reference 3.	Reference 3
h. Gas generation rates within the can	Pressure buildup within the can is calculated to be 1.62E-04 psi/year. This is generated from the generation of helium gas resulting from the contained alpha-emitting radioactive species.	Reference 4
12. State the conditions under which the contents were canned (e.g., canned wet, canned dry, etc.)	Canned dry in atmospheric air at ambient temperature and pressure. The HVAC was functional during the packaging of the contents. It is expected that the relative humidity was about 50% during packaging.	Reference 3

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
13. If the can was subject to a wet environment and will be stored dry, describe the drying process to include the following:	Can not subjected to wet environment.	Reference 2
a. Drying process	N/A	N/A
b. Dryness criteria	N/A	N/A
c. Dryness criteria verification compliance method	N/A	N/A
14. List can back fill gas and pressure	No back fill gas.	Reference 3
15. Contents of the can verification method. The verification method shall be onsite inspection or approved alternate. Provide photographs of contents if available.	All purchase orders for material for the can were reviewed and approved by a GA QA Engineer. Copies of the purchase orders are included in Reference 3.	Reference 3
	Component parts for the Can were all receipt-inspected by GA QA department. Any testing performed was also witnessed by a GA QA inspector. This documentation is included in Reference 3.	Reference 3
	The loading of the Can was visually observed by a qualified GA QA Engineer, per procedure GA Document Number HCP-6-6, Issue A. As steps were completed in the procedure, a GA QA Engineer initialed them to confirm they were completed. This document is included with Reference 3. The fuel loading campaign was recorded on videotape, a copy of which has been transmitted to INEEL.	Reference 3
	The GA Hot Cell Facility was designated as an MBA (Material Balance Area). All Special Nuclear Material (SNM) coming into and out of the facility was documented and a log sheet of all SNM was maintained by the designated Material Custodian. Material receipt forms are on file at GA along with regularly scheduled inventory reports. Qualifications for the SNM Material Custodian are documented in GA Nuclear Material Custody & Control Manual. The SNM material that went into the Can was confirmed by the material custodian and a GA QA inspector and appropriate adjustments made to the MBA account of SNM.	
	Qualifications for a QA inspector are listed in the GA QA Manual (Reference 6).	Reference 6

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
15. (Cont.)	GA's QA program has been routinely audited by both the NRC and DOE. References 8 and 20 list audits of GA that were conducted or observed by INEEL personnel.	References 8 and 20
16. Describe the distribution of the fuel [fuel unit(s)] over the length of the can.	Distribution is assumed to be homogeneous. A schematic of the loading is shown in Fig. 6.3 of Reference 2.	Reference 2
The following data applies to the fuel overpack (Secondary Enclosure) GA Dwg # 032231/A		Fig. 6.2 of Reference 2
11. Provide a design analysis of the can to include:		
a. Identify the purpose of all penetrations	There are no penetrations.	Reference 2
b. Thermal analysis of can and contents	A thermal analysis of the can is not applicable. The temperature of the fuel units in the Can are at room temperature as is the Can. There is currently only 1.69 watts of decay (as of 09/30/03) heat generated by the fuel items. This is easily dissipated by the Can.	Reference 2 updated to 09/30/03 and included in Table 1
c. Integrity test results agreed upon by the INEEL	A helium leak test of the bottom end cap of the Can assembly per QDI LTH-S-3801 with an acceptable leak rate of "non-detected" (at a detection limit of 4.4E-08 standard cc/sec) was satisfactory. A hydro test of the tube at the fabrication facility of 800 psi minimum was performed and was satisfactory. After loading of the primary closure into the secondary closure, the top end cap was placed on the tube inside the hot cell facility. The tube was mounted on a turn table with a variable rotational speed capability. The tube was rotated at a predetermined speed and the top end cap was welded remotely using an in-cell weld-head. The weld integrity was inspected using an in-cell mirror and an out-cell telescope for visual inspection. The weld was judged satisfactory by the QA inspector.	Reference 3
d. State the maximum allowable pressure on sealed can (e.g., could the can build up pressure?)	There is no can pressure buildup. Pressure is contained within the Primary Enclosure. The Can was hydrotested to 800 psi.	Reference 3

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
e. Drop test results, including extent of damage	Drop calculations have been performed for the load Top Module by BBWI for the IFSF Plant Safety Documents. This report is Ref. 23. The analyzed lift height for the fuel basket or an individual package is 45" over the top surface of the transfer car's steel slabs. The end cap of the secondary enclosure is severely damaged during most of the end drops and has significantly less containment integrity but the end cap will remain in place and provide a lesser level of containment. The damage to the primary enclosure is significantly less than the damage to the secondary enclosure. There is some material failure of the primary enclosure end cap (impacting end). This may reduce the level of confinement of the primary enclosure by a small amount, which is not defined because material failure does not develop into a gap from which contents could readily escape. The damage to the packages during a seismic event will be less than any of the analyzed postulated drops.	Reference 19
f. Other test reports	o Welder certifications were all satisfactory. o Factory chemical and mechanical tests on tube and end cap materials were satisfactory. o Weld integrity was satisfactory. o Transverse tension, flattening and reverse flattening tests satisfactory. o Flange and reverse bend tests satisfactory. o Yield strength confirmed.. o Proof load test to 165 lbs. (74,843 gm) of lifting cable assembly were satisfactory.	Reference 3
g. Material certification papers for can and seal	Certification papers for the can and seal are provided in Reference 3.	Reference 3
h. Gas generation rates within the can	There is no method of gas generation within the Secondary Enclosure, since the fuel items are sealed within the Primary Enclosure.	Reference 3
12. State the conditions under which the contents were canned (e.g., canned wet, canned dry, etc.)	Canned dry in atmospheric air at ambient temperature and pressure. Since the HVAC was operational, it is expected that the relative humidity was 50%.	Reference 3
13. If the can was subject to a wet environment and will be stored dry, describe the drying process to include the following:	Can not subjected to wet environment.	Reference 3
a. Drying process	N/A	N/A
b. Dryness criteria	N/A	N/A
c. Dryness criteria verification compliance method	N/A	N/A

PACKAGING RSD FORM
for ASSEMBLED HTGR CAN # 032231 for FHU # 315-391-120-44-001

	Description	Shipper Summary	References (See Table 3)
14.	List can back fill gas and pressure	Air at atmospheric pressure.	Reference 3
15.	Contents of the can verification method. The verification method shall be onsite inspection or approved alternate. Provide photographs of contents if available.	All purchase orders for material for the can were reviewed and approved by a GA QA Engineer. Copies of the purchase orders are included in Reference 3.	Reference 3
		Component parts for the Can were all receipt-inspected by GA's QA department. Any testing performed was also witnessed by a GA QA inspector. This documentation is included in Reference 3.	Reference 3
		The loading of the Can was visually observed by a qualified GA QA Engineer, per procedure GA Document Number HCP-6-6, Issue A. As steps were completed in the procedure, a GA QA Engineer initialed them to confirm they were completed. This document is included with Reference 3. The fuel loading campaign was recorded on videotape, a copy of which has been transmitted to INEEL.	Reference 3
		The GA Hot Cell Facility was designated as an MBA (Material Balance Area) and all SNM coming into and out of the facility was documented and a log sheet of all SNM maintained by the designated material custodian. Material receipt forms are on file at GA along with regual inventories. Qualifications for the material custodian are documented in GA's QA manual. The SNM material that went into the can was confirmed by the material custodian and a GA QA inspector and appropriate adjustments made to the MBA account of SNM.	Reference 6
		Qualifications for a QA inspector are listed in the GA QA Manual (Reference 6).	Reference 6
		GA's QA program has been routinely audited by both the NRC and DOE. References 8 and 20 list the recent audits of GA that were conducted or observed by INEEL personnel.	References 8 and 20
16.	Describe the distribution of the fuel [fuel unit(s)] over the length of the can.	Distribution is assumed to be homogeneous. A schematic of the loading is shown in Fig. 6.3 of Reference 2.	Reference 2

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Description	Shipper Summary	References (See Table 3)
IV. HANDLING AND RIGGING FIXTURES (rigging, slings, spreader bars, tools, yokes fuel handing and storage fixtures, etc.) - To be filled out for the fixtures to be used at the INEEL only. If no shipper or carrier supplied fixtures are to be used at the INEEL, clearly state this.		
1. Detailed drawings (most current drawings, indicating revision number and date) and list of materials for each fixture. Describe any deviations from the drawings.	The grapple to be used for removal of the FHU is a modified version of NAC's standard TRIGA handling item (Drawing # 61102, Reference 25) that has been modified by Change Request DCR 61102-024-1A (Reference 26). The secondary yoke(Drawing # 315-390-29, Rev. 2) will be used for the transfer process (Reference 27). The load test confirming the load carrying capability is in Reference 28. The remaining items used for the transfer are part of the standard set of fixtures for TRIGA transfer and therefore are on INEEL's site.	References 25 and 26 Reference 27 Reference 28 Reference 9
a. Dimensions (cm):	N/A	
i. Total length	N/A	
ii. Cross-sectional dimensions	N/A	
b. Weight (g):	N/A	
c. Type of load bar (if applicable):	N/A	
d. Special fuel, lid, SNF packaging or transportation packaging removal tools. State size and description of lid bolts.	N/A	
2. Design life of the fixture. Provide analysis and certification to support the design life, including certification of the material of construction.	N/A	
3. Provide certification of the most recent load tests performed on all lifting hardware (package lifting trunnions, devices and fixtures) accompanying the shipment.	N/A	
4. State if the fixture expendable or reusable.	N/A	
5. Describe the use of each fixture. If any special tools are required, describe them in detail and provide drawings. Include precautions for use of the fixture and tools.	N/A	

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Description	Shipper Summary		References (See Table 3)
V. TRANSPORTATION PACKAGE (SHIPPING CASK) DATA			
1. Identify the transportation package (cask), its model number and its serial number and transport vehicle to be used.	Transportation package:	NAC-LWT Cask in a 20' ISO, mounted on a Tri-State drop deck trailer.	Reference 18
	Model number:	NAC-LWT	
	Serial number:	Will be either #1 through #8. The exact Cask has not been identified yet, but will be prior to shipping	
	Transport vehicle:	Tractor and trailer supplied by Tri-State Motor Transit Company.	
2. Carrier Identification	Tri-State Motor Transit Company, Joplin, MO		Reference 18
3. Furnish 1 copy of the current Department of Transportation (DOT) Certificate of Competent Authority or the DOE/NRC Certificate of Compliance (C of C), or equivalent, as applicable.	An amendment to the original C of C (Revision 34) has been approved by the NRC. This has been supplied to INEEL.		Reference 13
4. Furnish 1 copy of the Safety Analysis Report For Packaging (SARP) or equivalent, including the Shipping Package Transport Plan if applicable.	"Safety Analysis Report for the NAC Legal Weight Truck" has been supplied. INEEL is an approved user of the NAC-LWT cask.		Reference 14
5. Provide 1 copy of reproducible detailed drawings of the packaging which indicate:	This information is in the SAR listed above.		Reference 14
a. Dimensions	The overall length is 199.8" (507.5 cm) and the maximum diameter is 44.2" (112 cm).		Reference 14
b. Weight (gross and net)	The maximum weight of the package is 48,000 lbs and the maximum weight of the contents and basket is 4,000 lbs.		Reference 14
c. Surface finish (e.g., roughness, painting, coating, etc.)	125 rms or better on all surfaces. Surface electropolished 304 SS		Reference 14

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Description	Shipper Summary	References (See Table 3)
d. Materials of construction	The cask body consists of a 0.75 " (1.91 cm.) thick XM19 SS inner shell, a 5.75 " (14.61 cm.) thick lead gamma shield, a 1.2" (3.05 cm.) thick XM19 SS outer shell, and a neutron shield tank. The inner and outer shells are welded to a 4" (10.16 cm.) thick 304 SS steel bottom end forging. The cask bottom consists of a 3" (7.62 cm.) thick 20.75 " (52.71 cm.) dia. lead disk enclosed by a 3.5 " (8.89 cm.) thick 304 SS plate and bottom end forging. The cask lid is 11.3" (28.7 cm.) thick 304 SS stepped design	Reference 14
e. Size and description of lid bolts	Lid bolts are twelve (12) ea. Grade 1660, Class A 1-UNC x 9 long socket head cap screws.	Reference 14
f. Poison, poison inserts or spacing insert necessary for shipping	There are no poison inserts. The cask has a neutron shield tank containing an ethylene glycol/water solutions that is 1% boron by weight. The spacing insert is shown in NAC Drawing 315-391-123 (Reference 11).	Reference 14 Reference 11
6. Describe the poison, poison inserts or spacing insert necessary for shipping and furnish certification of integrity.	No poison inserts. The neutron shield tank contains ethylene glycol/water solution that is 1% boron by weight. There is a spacing insert that is loaded in the bottom of the shipping cask prior to loading of the FHU. The spacing insert is of right circular cylinder shape. It is constructed of 304 SS. Including the guide pins it is 135.25" (343.54 cm.) long by 13.265" (33.69 cm) in diameter. There are two (2) guide pins made from bar stock, one (1) spacer tube, five (5) support plates, two (2) grapple plates, four (4) spacer plates, and eight (8) guide bars. The structural and thermal evaluation of the spacer is included in Reference 12. The cask integrity is included in the SAR, Reference 14.	Reference 12 Reference 14
7. Provide a copy of loading, handling, and dry cask storage procedures (if applicable).	The NAC generic operating procedure is Reference 18. The NAC-LWT loading procedure at GA is Reference 21.	Reference 18 Reference 21
8. Furnish tie-down configuration of the package to the transport vehicle (Drawings or Sketches).	There are no drawings available. The cask ISO is mounted on the trailer using standard ISO twist locks.	
9. Furnish tie-down configuration of the package to the transport vehicle (Documents).	There are no drawings available. The cask ISO is mounted on the trailer using standard ISO twist locks.	

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Description	Shipper Summary	References (See Table 3)
VI. ENVIRONMENTAL SAFETY AND HEALTH		
1. Provide copies of any documents covering criticality safety evaluations and calculations which determine the minimum critical number of pieces of the subject fuel and which evaluate the criticality safety of shipping, handling and storing the fuel.	References 5 and 10.	References 5 and 10
2. Provide a copy of all existing NEPA documents for the storage and handling of this fuel. If none, state none.	None	
VIII. QUALITY ASSURANCE		
1. Reference attached documentation that the shipper's QA program has been approved by the INEEL.	At the time of packaging (1995), the Shipper's QA program was approved by INEEL (see Reference 7). The next INEEL audit of GA was conducted in November 2000, by Bechtel BWXT Idaho. This audit focused on TRIGA fuel design and fabrication (see Reference 8), and resulted in the approval of General Atomics TRIGA International Group as a Qualified Supplier. This approval status was restricted to the supply of TRIGA Reactor Fuel to BBWI, purchased through the INEEL University Fuels Program. The latest audit (Reference 20) of GA was conducted in April 2003 by J. H. Valentine for the purpose of approving GA for shipment of fuel to INEEL.	Reference 7 Reference 8 Reference 20

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Description		Shipper Summary	References (See Table 3)
X.	OTHERS		
1.	Provide any information not provided above that may have an impact on the shipper's SNF receipt and storage or INEEL Operations.	None	

Prepared By: J. S. Greenwood, Manager, TRIGA Reactors Facility — J. S. Greenwood (Signature) — 9/11/03 (Date)

Approved By: R. I. De Velasco, Manager, Decommissioning Projects — R. De Velasco (Signature) — 9/11/03 (Date)

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Table 3 - Shipper References

Reference Number	Type (Drawing, Letter, etc.)	Title	Number	Revision/Date	Quality Assurance [1]
1	DOE Contract	"General Atomics Hot Cell Decontamination & Decommissioning Project", Phases 2 & 3 Activities; Statement of Work, Task 16" , Attachment I to USDOE Contract DE-AC03-95SF20798	Contract DE-AC03-95SF20798 (Note: This is Reference 1-1 called out in Reference 2 below)		Non QA Record
2	General Atomics Report	HTGR/RERTR Fuel Materials Characterization and Packaging Report	PC-000384	Rev.2, 4/02	QA Record Level NA
3	General Atomics QA Inspection Report	GA Hot Cell D&D Project Final Assessment Report, Transfer of HTGR/RERTR Fuel from Hot Cell to Bldg. 30 (Note: This is Reference 1-3 called out in Reference 2 above.)	16A	3/28/96	QA Record Level 1
4	General Atomics Internal Correspondence	J. S. Greenwood to C. M. Miller, "Calculation of Pressure Buildup in Fuel Storage Enclosures", 1/15/96	HCI:015:JSG:96	1/15/96	QA Record Level NA
5	General Atomics Internal Correspondence	Malakhof, V., "Nuclear Safety Evaluation of the Irradiated Fuel Material Interim Storage Facility at General Atomics"	NS:94:VM:399 Nuclear Safety File No. 533.0	6/95	QA Record Level NA
6	General Atomics Quality Assurance Manual	Quality Assurance Manual, General Atomics, San Diego, CA		3rd Edition, Revision D, 8/12/96	QA Record Level NA
7	General Atomics Internal Correspondence	Nicolayeff, V., "Hot Cell Irradiated Fuel Materials, INEEL Audits of General Atomics"	123:VN:02:18	5/15/02	QA Record Level NA
8	INEEL Correspondence	INEEL Procurement Quality Manager to J. Razvi, GA TRIGA Director, "Supplier Evaluation (TRIGA International)"	MTW-009-01	11/29/00	QA Record Level NA
9	Drawing	NAC International, Top Module, General Atomics IFM, LWT Cask	315-391-120	Rev. 2/ 5/8/03	QA Record Level A
10	Calculation Package	NAC International, Criticality Evaluation for the NAC-LWT with GA Irradiated Fuel Material (IFM)	14661-600	Rev. 0/ 12/16/02	QA Record Level A
11	Drawing	NAC International, Spacer, General Atomics IFM, LWT Cask	315-391-123	Rev. 1/ 5/8/03	QA Record Level A
12	Calculation Package	NAC International, Structural and Thermal Evaluation of the Top Module and Spacer	14661-200	Rev. 0/ 12/18/02	QA Record Level A
13	DOE/NRC Form	Certificate of Compliance for Radioactive Material Packages	Form 618 Certificate # 9225	Rev. 34/ June 30, 2003	QA Record Level A

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Reference Number	Type (Drawing, Letter, etc.)	Title	Number	Revision/Date	Quality Assurance [1]
14	Report	Safety Analysis Report for the NAC Legal Weight Truck Cask	Docket No. 71-9225 T-88004	Rev. 34 11/02	QA Record Level A
15	DOE/NRC Form	USDOE & USNRC Nuclear Material Transaction Report	DP-741	04/03	QA Record Level 1
16	Form	Fuel RSD Form for the Assembled RERTR Can #32230 for FHU # 315-391-120-44-001	434.28	NR/ 06/09/03	QA Record Level 1
17	Memo	Hot Cell IFM - HTGR/RERTR Enclosure Pressure Calculation	123:VN:03:08	Rev. 0/ 03/19/03	QA Record Level 1
18	Procedure	NAC-LWT Cask Generic Operating Procedure	NAC 315-P-02	Rev.11/ 6/13/03	QA Record Level A
19	Calculation Package	BBWI, Drop Analysis of the General Atomics HTGR/RERTR Fuel Packaing at IFSF	EDF-3446	Initial Issue March 3, 2003	QA Record Level A
20	INEEL Letter	From J. H. Valentine to J. S, Greenwood, "Approval of the General Atomics Quality Assurance Program for Shipment of the General Atomics Spent Nuclear Fuel to the INEEL"	CCN 41779	April 24, 2003	QA Record Level NA
21	Procedure	GA Hot Cell D&D Project: Irradiated Fuel Materials Shipment	DDP-1.12	Issue A 9/2/03	QA Record Level 1
22	Fabrication Package	Columbiana Hi Tech Data Package, Top Module and Spacer, NAC PO #03-0157.	Columbiana Hi Tech WO#03-003	August 21, 2003	QA Record Level 1
23	Drawing	NAC International, HTGR Spacer, General Atomics IFM,LWT Cask	315-391-120-97	Rev. 2/ 5/8/03	QA Record Level A
24	Drawing	NAC International, RERTR Spacer, General Atomics IFM,LWT Cask	315-391-120-98	Rev. 2/ 5/8/03	QA Record Level A
25	Drawing	NAC International, Grapple Assembly, INEEL, CPP	61102-024	Rev. 2 9/2/03	QA Record Level A
26	Design Change Request	Grapple Assembly, INEEL, CPP	61102-024-1A	Original Issue 8/29/03	QA Record Level A
27	Drawing	NAC International, Secondary Yoke, LWT Cask	315-390-29	Rev. 2	QA Record Level A
28	Certification	Load Certification for Secondary Yoke			

[1] Describe quality level of reference. State "Non-QA Record" if not a quality assurance record.

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
I. SHIPPER AND SHIPMENT IDENTIFICATION INFORMATION		
1. Shipper	RIS: **LAW**	N/A
	Name: General Atomics	N/A
	Address: PO Box 85608, San Diego, CA 92138	N/A
	Telephone: (858) 455-2823 or (858) 455-2010 (24 hrs.)	N/A
	Attention: Dr. Keith Asmussen	N/A
2. Shipping Agent	Name: NAC International	N/A
	Location: 3930 East Jones Bridge Road, Norcross, GA 30092	N/A
	Name of authorized person for shipping agent: Michael J. Mosley	N/A
3 License number	SNM 696	N/A
4. Transfer authority - contact, NM draft or order number	DOE Contract No. DE-AC03-95SF20798	Reference 1
5. License number for import (foreign receipts)	N/A, (SNF is shipped from USA)	N/A
6. U.S. port of entry (foreign receipts)	N/A, (SNF is shipped from USA)	N/A
7. Batch identification number	From Project No. 7340	Reference 2
8. Country control number	USUS0000, USUSUS00	Reference 19
9. Ownership of accountable nuclear material	The accountable nuclear material contained in the subject RERTR Can is Government-Owned.	Reference 2

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
II. SPENT NUCLEAR FUEL DATA		
1. Reactor name and reactor name abbreviation or fuel description and component	Note: The fuel being shipped is not reactor core fuel. The fuel being shipped was research and development test fuel. The reactor identified below was used solely as a source of neutrons in order to test the performance/integrity of the subject fuel units. Oak Ridge 30MW(t) Research Reactor/ORR	Reference 2
2. Core information	Core identification or designation: See Above	Reference 2
	Initial criticality date: The core data is not related to fuel being shipped.	N/A
	Shutdown date: The core data is not related to fuel being shipped.	N/A
	Number of elements or assemblies: The core data is not related to fuel being shipped.	N/A
	Neutron flux: The core data is not related to fuel being shipped	N/A
	Neutron lifetime: Unknown	Not Available
	Significant events in its operating history: There were no significant events reported in the referenced document for each of the irradiated fuel units	Reference 5

FUEL RSD FORM
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Description	Shipper Summary	References From Table 2
3. General fuel description with a description of how the fuel was fabricated	Note: The fuel being shipped is not reactor core fuel. The fuel being shipped is test fuel. The identified reactor was used solely as a source of neutrons in order to test the performance/integrity of the subject fuel units. The fuel being shipped is a homogeneous erbium-uranium-zirconium hydride alloy metal. The metal alloy is first created by melting erbium-uranium-zirconium together in the proper proportions at about the same size as the required fuel meat. The cooled metal alloy is then placed into an oven where the oven is evacuated. Hydrogen gas is then added and the oven heated. At a particular temperature, the hydrogen goes into solution with the fuel meat (detected by a noticeable drop in hydrogen pressure) and bonds interstitially to the Zr to create $Er(U,Zr)H_{1.65}$. The hydrogen remains in solution with the fuel meat even at normal reactor operating temperatures. The fuel meats are then machined to the required diameter and length and then inserted into an Incoloy 800H tube cladding. Spacers (Mo) and springs (Inconel 600) are added. Special top and bottom end fittings (304SS) are then affixed to the tubing ends by seal weldment.	Table 2.3 of Reference 2
4. Fuel unit (rod, plate, assembly, etc.)	13 intact RERTR fuel rods and 7 sectioned RERTR fuel rods. The pieces of the sectioned rods were placed into separate aluminum tubes. The ends of these aluminum tubes were crimped to keep the loose pieces contained. These twenty items were loaded dry, into an A-36 carbon steel basket and loaded into the Primary Enclosure.	Table 2.3 of Reference 2
5. Fuel unit specific data each fuel unit, include as Table 1 or equivalent	The unique identification number for the units are: For 45 wt-%U units: 1086[1], 1087[1], 1088[1], 1089, 1090, 1091TC, 1092TC, 1093[1], and 1094. For 30 wt-%U units: 1080, 1081, 1082[1], 1083, 1084TC, 1085TC. For 20 wt-%U units: 1096, 1097, 1098[1], 1099TC, 1100TC[1]. TC implies a fuel unit with a built-in thermocouple that was used to monitor the fuel unit temperature. The thermocouples are: three each 0.060" (0.152 cm) dia. sheathed, grounded, chromel-alumel (Type K). [1] Indicates fuel units were physically sectioned as part of the Post Irradiation Examination program.	Reference 5

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FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	Pre-irradiation isotopic composition or reportable NM (element and isotope weight): From 8.90E+01 to 9.46E+01 g U for each of the five 20 wt-%U units. From 1.47E+02 to 1.546E+02 g U for the six 30 wt-%U units. From 2.65E+02 to 2.79E+02 g U for the nine 45 wt-%U units.	Reference 5
	Post-irradiation isotopic composition or reportable NM (element and isotope weight): Twenty fuel units with combined isotopes of U-233 (1.80E-05 g), U-234 (6.30E-02 g), U-235 (3.52E+02 g), U-236 (8.90E+01 g) and Pu-239 (2.10E+01 g), Pu-240 (5.86E+00 g), Pu-241 (1.78E+00 g), Pu-242 (8.60E-01 g). See Table 1 for individual fuel unit isotopes for the 13 intact fuel units, and the composite fuel unit isotopes for the 7 sectioned units.	Table 6.22 of Reference 2
	Time in reactor: Various. From 295 EFPD to 920 EFPD. See Table 1 for details on each of the 13 intact fuel units and the composite details for the 7 sectioned fuel units.	Reference 5
	Reactor power level: 30 MW(th)	Reference 5
	Cooling time (discharge date from reactor or reactor shutdown date): Various. As of 9/30/03, from 19 years, 4 months to 21 years, 1 month. See Table 1 for details on each of the 13 intact fuel units and the composite details for the 7 sectioned fuel units.	Reference 2 updated to 09/30/03 and included in Table 1
	Percent estimated burnup: Various. From 24% to 82%. See Table 1 for details on each of the 13 intact fuel units and the composite details for the 7 sectioned fuel units.	Reference 5
	Heat decay (watts): Various. From 5.40E-02 to 5.98E-01 watts or a total of 8.57E+00 watts for all fuel units. See Table 1 for details on each of the 13 intact fuel units and the composite details for the 7 sectioned fuel units.	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
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Description	Shipper Summary	References From Table 2
	Date of post-irradiation data: 9/30/03	Reference 2 updated to 09/30/03 and included in Table 1
6. Description of the distribution of fissile material:		
a. Description of fissile material at beginning-of-life (BOL) over the length of the fuel.	The RERTR element fuel meats were manufactured so as to contain a homogeneous distribution of fissile material over the length of the fueled portion of the fuel unit assembly.	Table 2.3 of Reference 2
b. As shipped (g/linear foot). Provide sufficient information to support criticality evaluation.	The distribution of fissile material over the length of the fueled portion of the RERTR units, as shipped, is assumed to be homogeneous.	Reference 2
7. Detailed drawings (most current drawings, indicating revision number and date) and list of materials. Describe any deviations from the drawings. Include documentation providing traceability from the drawings to the fuel elements to be shipped. The following information shall be obtainable from the drawings and listed below:	The detail drawings are T14R210D210, Rev. E, 6/11/91, (for the standard RERTR fuel element configuration), and T14R210E220, Rev. C, 11/15/82, (for the instrumented or thermocoupled RERTR fuel element configuration).	References 3 and 4
a. Physical description of each type of fuel unit:	Typical intact fuel unit is 30.13" (76.53 cm) long by 0.542" (1.377 cm) OD. The cladding is 0.016" (0.041 cm) thick Incoloy 800H. There are four each active fuel meats inserted in the cladding, each of which are 5.51" (14.00 cm) long by 0.512" (1.300 cm) diameter. A Molybdenum spacer is inserted to assure proper fit. An Inconel 600 compression spring holds the fuel meats in place. Special end fittings of 304SS are seal welded to the top and bottom of the elements. The fuel meats start 2.01" (5.11 cm) from the bottom of the fuel unit.	References 2, 3, 4, & 5
b. Total length:	30.13" (76.53 cm)	References 3, 4, & 5
c. Length of fueled portion:	22.00" (55.88 cm)	References 3, 4, & 5
d. Position of fueled portion with respect to a permanent reference point on the fuel unit:	2.01" (5.11 cm) from the bottom.	References 3, 4, & 5
e. Cross-sectional dimensions	0.542" (1.377 cm) OD	References 3, 4, & 5
f. Shape (plates, rod, etc.):	Rod	References 3, 4, & 5
g. Plenum spacers or springs:	Spacer (Molybdenum), Spring (Inconel 600)	References 3, 4, & 5

Description	Shipper Summary	References From Table 2
h. Configuration of FHU as shipped, including cuts made to prepare for shipping, shipment configuration (assembly, loose plates, subassembly, etc.):	The FHU will consist of two (2) Cans. One consisting of HTGR fuel and the other RERTR fuel. This section will describe the assembled RERTR Can which consists of 13 intact RERTR fuel units, and 7 sectioned RERTR fuel units packaged inside a welded Primary Enclosure. The component pieces of the sectioned units were placed in separate aluminum tubes. The ends of these tubes were crimped to contain the loose parts. The Primary Enclosure is contained inside a welded Secondary Enclosure. No cuts were made for shipping. The Fuel RSD form for the HTGR Can will describe the configuration for the HTGR fuel..	Reference 2
i. Fuel particle size and composition:	Fuel meats are 5.51" (14.00 cm) long by 0.512" (1.300 cm) diameter each.	Reference 5.
j. Fuel matrix composition:	$Er(UZr)H_{1.65}$	Table 2.3 of Reference 2.
8. As shipped dimensions (cm):		
a. Cut length:	Fuel units 1080, 1081, 1083, 1084TC, 1085TC, 1089, 1090, 1091TC, 1092TC, 1094, 1096, 1097, 1099TC each have four each fuel meats that are 5.51" (14.00 cm) long. Fuel units 1082, 1086, 1087, 1088, 1093, 1098, 1100TC have fuel sections which were cut up into various lengths at the ORNL; GA does not know details of the sectioning process.	Table 2.3 Reference 2
b. Cross-sectional dimensions:	All fuel units are 0.512" (1.300 cm) diameter	Table 2.3 Reference 2

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Description	Shipper Summary	References From Table 2
9. Composition and weight (g/fuel unit) of material in the fuel as shipped:		
a. Total:	Total weight of material in the fuel being shipped in the RERTR Can is 10,767 g. This consists of: 6.72E+03 g Zirconium 3.04E+03 g Uranium Isotopes consisting of: 2.60E+03 g U-238 6.30E-02 g U-234 3.52E+02 g U-235 8.90E+01 g U-236 1.80E-05 g U-233 1.16E+02 g Hydrogen 6.33E+01 g Erbium 1.54E+01 g Carbon 2.95E+01 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 2.10E+01 g Pu-239 5.86E+00 g Pu-240 1.78E+00 g Pu-241 8.60E-01 g Pu-242 7.84E+02 g Fission Products	Reference 2 and Table 1
	The total weight of material being shipped in the 1080 fuel unit is 5.13E+02 g. This consists of: 3.49E+02 g Zirconium 1.37E+02 g Uranium Isotopes consisting of: 1.17E+02 g U-238 2.86E-03 g U-234 1.60E+01 g U-235 4.05E+00 g U-236 8.18E-07 g U-233 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.34E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 9.54E-01 g Pu-239 2.66E-01 g Pu-240 8.08E-02 g Pu-241 3.91E-02 g Pu-242	Reference 2 updated to 09/30/03 and included in Table 1

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Description	Shipper Summary	References From Table 2
	1.56E+01 g Fission Products	
	The total weight of material being shipped in the 1081 fuel unit is 5.13E+02 g. This consists of: 3.49E+02 g Zirconium 1.37E+02 g Uranium Isotopes consisting of: 1.18E+02 g U-238 2.68E-03 g U-234 1.50E+01 g U-235 3.79E+00 g U-236 7.67E-07 g U-233 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.26E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 8.95E-01 g Pu-239 2.50E-01 g Pu-240 7.57E-02 g Pu-241 3.66E-02 g Pu-242 1.58E+01 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the 1083 fuel unit is 5.13E+02 g. This consists of: 3.49E+02 g Zirconium 1.39E+02 g Uranium Isotopes consisting of: 1.18E+02 g U-238 3.04E-03 g U-234 1.70E+01 g U-235 4.30E+00 g U-236 8.69E-07 g U-233 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.42E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 1.01E+00 g Pu-239 2.83E-01 g Pu-240 8.58E-02 g Pu-241 4.15E-02 g Pu-242 1.36E+01 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	The total weight of material being shipped in the 1084TC fuel unit is 5.06E+02 g. This consists of: 3.49E+02 g Zirconium 1.40E+02 g Uranium Isotopes consisting of: 1.12E+02 g U-238 3.94E-03 g U-234 2.20E+01 g U-235 5.56E+00 g U-236 1.13E-06 g U-233 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.84E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 1.31E+00 g Pu-239 3.66E-01 g Pu-240 1.11E-01 g Pu-241 5.38E-02 g Pu-242 4.99E+00 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the 1085TC fuel unit is 5.06E+02 g. This consists of: 3.49E+02 g Zirconium 1.33E+02 g Uranium Isotopes consisting of: 1.12E+02 g U-238 3.04E-03 g U-234 1.70E+01 g U-235 4.30E+00 g U-236 8.69E-07 g U-233 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.42E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 1.01E+00 g Pu-239 2.83E+00 g Pu-240 8.58E-02 g Pu-241 4.15E-02 g Pu-242 1.23E+01 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	The total weight of material being shipped in the 1089 fuel unit is 6.19E+02 g. This consists of: 3.29E+02 g Zirconium 2.22E+02 g Uranium Isotopes consisting of: 2.09E+02 g U-238 1.79E-03 g U-234 1.00E+01 g U-235 2.53E+00 g U-236 5.11E-07 g U-233 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 8.38E-01 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 5.96E-01 g Pu-239 1.66E-01 g Pu-240 5.05E-02 g Pu-241 2.44E-02 g Pu-242 5.47E+01 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the 1090 fuel unit is 6.19E+02 g. This consists of: 3.29E+02 g Zirconium 2.39E+02 g Uranium Isotopes consisting of: 2.18E+02 g U-238 3.04E-03 g U-234 1.70E+01 g U-235 4.30E+00 g U-236 8.69E-07 g U-233 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 1.42E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 1.01E+00 g Pu-239 2.83E-01 g Pu-240 8.58E-02 g Pu-241 4.15E-02 g Pu-242 3.75E+01 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	The total weight of material being shipped in the 1091TC fuel unit is 6.06E+02 g. This consists of: 3.29E+02 g Zirconium 2.45E+02 g Uranium Isotopes consisting of: 2.04E+02 g U-238 5.91E-03 g U-234 3.30E+01 g U-235 8.34E+00 g U-236 1.69E-06 g U-233 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 2.76E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 1.97E+00 g Pu-239 5.49E-01 g Pu-240 1.67E-01 g Pu-241 8.06E-02 g Pu-242 1.78E+01 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the 1092TC fuel unit is 6.06E+02 g. This consists of: 3.29E+02 g Zirconium 2.22E+02 g Uranium Isotopes consisting of: 1.99E+02 g U-238 3.22E-03 g U-234 1.80E+01 g U-235 4.55E+00 g U-236 9.20E-07 g U-233 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 1.51E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 1.07E+00 g Pu-239 2.99E-01 g Pu-240 9.09E-02 g Pu-241 4.40E-02 g Pu-242 4.18E+01 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	The total weight of material being shipped in the 1094 fuel unit is 6.19E+02 g. This consists of: 3.29E+02 g Zirconium 2.45E+02 g Uranium Isotopes consisting of: 2.04E+02 g U-238 5.91E-03 g U-234 3.30E+01 g U-235 8.34E+00 g U-236 1.69E-06 g U-233 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 2.76E+00 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 1.97E+00 g Pu-239 5.49E-01 g Pu-240 1.67E-01 g Pu-241 8.06E-02 g Pu-242 2.99E+01 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the 1096 fuel unit is 4.73E+02 g. This consists of: 3.72E+02 g Zirconium 8.50E+01 g Uranium Isotopes consisting of: 7.25E+01 g U-238 1.79E-03 g U-234 1.00E+01 g U-235 2.53E+00 g U-236 5.11E-07 g U-233 6.39E+00 g Hydrogen 8.38E-01 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 5.96E-01 g Pu-239 1.66E-01 g Pu-240 5.05E-02 g Pu-241 2.44E-03 g Pu-242 8.78E+00 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	The total weight of material being shipped in the 1097 fuel unit is 4.73E+02 g. This consists of: 3.72E+02 g Zirconium 8.50E+01 g Uranium Isotopes consisting of: 7.25E+01 g U-238 1.79E-03 g U-234 1.00E+01 g U-235 2.53E+00 g U-236 5.11E-07 g U-233 6.39E+00 g Hydrogen 8.38E-01 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 5.96E-01 g Pu-239 1.66E-01 g Pu-240 5.05E-02 g Pu-241 2.44E-02 g Pu-242 8.44E+00 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1
	The total weight of material being shipped in the 1099TC fuel unit is 4.69E+02 g. This consists of: 3.72E+02 g Zirconium 8.00E+01 g Uranium Isotopes consisting of: 6.87E+01 g U-238 1.61E-03 g U-234 9.00E+00 g U-235 2.28E+00 g U-236 4.60E-07 g U-233 6.39E+00 g Hydrogen 7.54E-01 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 5.37E-01 g Pu-239 1.50E-01 g Pu-240 4.54E-02 g Pu-241 2.20E-02 g Pu-242 9.20E+00 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	The total weight of material being shipped in the 1082, 1086, 1087, 1088, 1093, 1098, and 1100TC fuel units is 3.93E+03 g. This consists of: 2.41E+03 g Zirconium 9.28E+02 g Uranium Isotopes consisting of: 7.71E+02 g U-238 2.24E-02 g U-234 1.25E+02 g U-235 3.16E+01 g U-236 6.39E-06 g U-233 4.14E+01 g Hydrogen 2.47E+01 g Erbium 4.17E+00 g Carbon 1.05E+01 g Plutonium Isotopes consisting of: 0.00E+00 g Pu-238 7.46E+00 g Pu-239 2.08E+00 g Pu-240 6.31E-01 g Pu-241 3.05E-01 g Pu-242 5.14E+02 g Fission Products	Reference 2 updated to 09/30/03 and included in Table 1
b. Fuel (chemical form of uranium and plutonium):	The chemical forms of the fuel are $Er(U,Zr)H_{1.65}$, $(U,Zr)H_{1.65}$, $Er(Pu,Zr)H_{1.65}$ and $(Pu,Zr)H_{1.65}$	Reference 2
	Fuel Unit 1080 consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.49E+02 g Zirconium 1.37E+02 g Uranium Isotopes 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.34E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1081 consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.49E+02 g Zirconium 1.37E+02 g Uranium Isotopes 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.26E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	Fuel Unit 1083 consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.49E+02 g Zirconium 1.39E+02 g Uranium Isotopes 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.42E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1084TC consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.49E+02 g Zirconium 1.40E+02 g Uranium Isotopes 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.84E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1085TC consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.49E+02 g Zirconium 1.33E+02 g Uranium Isotopes 5.90E+00 g Hydrogen 2.62E+00 g Erbium 1.69E+00 g Carbon 1.42E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1089 consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.29E+02 g Zirconium 2.22E+02 g Uranium Isotopes 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 8.38E-01 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1090 consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.29E+02 g Zirconium 2.39E+02 g Uranium Isotopes 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 1.42E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References From Table 2
	Fuel Unit 1091TC consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.29E+02 g Zirconium 2.45E+02 g Uranium Isotopes 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 2.76E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1092TC consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.29E+02 g Zirconium 2.22E+02 g Uranium Isotopes 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 1.51E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1094 consists of $Er(U,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 3.29E+02 g Zirconium 2.45E+02 g Uranium Isotopes 5.69E+00 g Hydrogen 5.51E+00 g Erbium 6.20E-01 g Carbon 2.76E+00 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1096 consists of $(U,Zr)H_{1.65}$ and $(Pu,Zr)H_{1.65}$ 3.72E+02 g Zirconium 8.50E+01 g Uranium Isotopes 6.39E+00 g Hydrogen 8.38E-01 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1097 consists of $(U,Zr)H_{1.65}$ and $(Pu,Zr)H_{1.65}$ 3.72E+02 g Zirconium 8.50E+01 g Uranium Isotopes 6.39E+00 g Hydrogen 8.38E-01 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
	Fuel Unit 1099TC consists of $(U,Zr)H_{1.65}$ and $(Pu,Zr)H_{1.65}$ 3.72E+02 g Zirconium 8.00E+01 g Uranium Isotopes 6.39E+00 g Carbon 7.74E-01 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1

FUEL RSD FORM
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Description	Shipper Summary	References From Table 2
	Fuel Units 1082, 1086, 1087, 1088, 1093, 1098, and 1100TC consist of $Er(U,Zr)H_{1.6}$ $(U,Zr)H_{1.65.5}$ $(Pu,Zr)H_{1.65}$ and $Er(Pu,Zr)H_{1.65}$ 2.41E+03 g Zirconium 9.28E+02 g Uranium Isotopes 4.14E+01 g Hydrogen 2.47E+01 g Erbium 4.17E+00 g Carbon 1.05E+01 g Plutonium Isotopes	Reference 2 updated to 09/30/03 and included in Table 1
c. Alloy or diluent in the matrix:	None	N/A
d. Cladding:	The cladding tubing is Incoloy 800H and the end fittings are 304SS. The total weight of cladding material is 3145.8 g, consisting of 1704.5 g Fe, 716.4 g Ni, 694.1 g. Cr, and 30.8 g of Mn.	Table 2.5 of Reference 2
e. Any external coatings applied to the cladding:	None	N/A
f. Thermal transfer material (e.g., sodium):	None	N/A
g. Organic materials:	None	N/A
h. Special additives:	None	N/A
i. Chemically reactive materials (e.g., sodium):	None	N/A
j. Neutron poisons, fixed or burnable:	Erbium, 65.31 g	Table 2.4 Reference 2
k. Other (specify):	Compression Springs, 20 ea., constructed of Inconel 600, collectively weighing a total of 270.3 g, consisting of 202.7 g Ni and 67.6 g Cr. Spacers, 20 ea., 1 per fuel unit constructed of Molybdenum, collectively weighing a total of 19.4 g. Containment sleeves 7 ea. to contain the sectioned fuel units, constructed of Aluminum, collectively weighing 1150 g total. Chromel-Alumel thermocouples, 6 sets of three each, collectively weighing a total of 90 g.	Table 2.5 Reference 2

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Description	Shipper Summary	References From Table 2
10. Radiation level curve, at 3 feet in air, as a function of time out of the reactor.	No radiation levels were measured on the individual fuel units, only on the assembled RERTR Can. As of December 1, 1995, the radiation levels, measured at one meter, ranged from a minimum of 208 R/hr at the top of the Can to a maximum of 418 R/hr at the middle. The bottom measured 381 R/hr. Based on the decay of the fission products from 12/1/95 to 09/30/03, the radiation levels were calculated to range from 162 R/hr at the top, 325 R/hr at the middle and 296 R/hr at the bottom. Based on the radiation levels calculated on the assembled HTGR Can (See Reference 20), the combined radiation level will be 200 R/hr at the top, 407 R/hr at the middle, and 326 R/hr at the bottom.	Table 6.24 Reference 2 updated to 09/30/03 and Reference 20.
11. Description of fuel element/cladding degradation, including where and how it occurred:		
a. Description of fuel element/cladding degradation, including where and how it occurred.	Seven fuel units were intentionally sectioned for Post-Irradiation Examination. The remaining 13 fuel units are intact.	Reference 2
b. Is there any reason to suspect the fuel cladding has been damaged to the extent that it would no longer retain fission products? If so, state reasons.	Seven fuel units were intentionally sectioned for Post-Irradiation Examination and no longer retain fission products. The remaining 13 fuel units are intact.	Reference 2
c. Reactor and storage history. Note only conditions that might affect storage or future cladding integrity.	None	N/A
d. Describe exposure to contaminants. (Hg, halides, etc.)	None	N/A
i. Was the contamination removed?	N/A	N/A
ii. How was the contamination removed?	N/A	N/A
12. Activation products and curies.	Total Activation Product radionuclide activities for entire Can as of 09/30/03 are: Mn-54: 2.14E-05 Ci; Fe-55: 4.27E+00 Ci; Co-60: 8.84E-01 Ci; Ni-59: 3.30E-01 Ci; Ni-63: 3.75E+01 Ci	Table 6.22 Reference 2 updated to 09/30/03.
13. Metallurgical state of the cladding (annealed, stressed, sensitized, estimated exposure [nvt], etc.)	The fuel element cladding Incoloy 800H tubing was solution annealed during fabrication. The estimated neutron exposure while in the reactor was 5.0E+21 n-cm^2 (fast) for all 20 fuel elements.	Reference 5

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Description	Shipper Summary	References From Table 2
14. Describe measured or projected external contamination, particularly alpha contamination levels.	No measurements were taken of the contamination levels on the fuel units. Seven units were segmented and contain significant levels of contamination. The contamination was only measured on the exterior surfaces (Secondary Enclosure) of the Can. The measured results for surface contamination as of 12/1/95 were: 2700 dpm/100 cm^2 (beta + gamma) and 13 dpm/100 cm^2 (alpha).	Reference 2

Prepared By:

J. S. Greenwood
Manager, TRIGA Reactor Facility
Printed Name/Title

J. S. Greenwood
Signature

7/7/03
Date

Approved By:

R. I. De Velasco
Manager, Decommissioning Projects
(Shipper Management) Printed Name/Title

Signature

7/7/03
Date

FUEL RSD FORM
for RERTR ASSEMBLED CAN #032230 for FHU # 315-391-120-44-001

Table 1 - Fuel Units and RERTR Can Specific Data

CAN	Fuel Unit	Pre-Irradiation		Post-Irradiation as of September 30, 2003															
Unique ID Number	Unique ID Number	U (g)	U-235 (g)	U (g)	U-234 (g)	U-235 (g)	U-236 (g)	U-233 (g)	Pu (g)	Pu-238 (g)	Pu-239 (g)	Pu-240 (g)	Pu-241 (g)	Pu-242 (g)	Time in Reactor	Power Level	Cooling Time/Date	Percent Burnup	Decay Heat (watts)
032230	1080	1.54E+02	3.03E+01	1.37E+02	2.86E-03	1.60E+01	4.05E+00	8.18E-07	1.34E+00	0.00E+00	9.54E-01	2.66E-01	8.08E-02	3.91E-02	573 EFPD	30 MW(th)	21 yrs 1 Mo 9/30/03	47%	1.70E-01
032230	1081	1.54E+02	3.03E+01	1.37E+02	2.68E-03	1.50E+01	3.79E+00	7.67E-07	1.26E+00	0.00E+00	8.95E-01	2.50E-01	7.57E-02	3.66E-02	573 EFPD	30 MW(th)	21 yrs 1 Mo 9/30/03	50%	1.73E-01
032230	1083	1.54E+02	3.03E+01	1.39E+02	3.04E-03	1.70E+01	4.30E+00	8.69E-07	1.42E+00	0.00E+00	1.01E+00	2.83E-01	8.58E-02	4.15E-02	573 EFPD	30 MW(th)	21 yrs 1 Mo 9/30/03	44%	1.48E-01
032230	1084TC	1.47E+02	2.89E+01	1.40E+02	3.94E-03	2.20E+01	5.56E+00	1.13E-06	1.84E+00	0.00E+00	1.31E+00	3.66E-01	1.11E-01	5.38E-02	573 EFPD	30 MW(th)	21 yrs 1 Mo 9/30/03	24%	5.40E-02
032230	1085TC	1.47E+02	2.88E+01	1.33E+02	3.04E-03	1.70E+01	4.30E+00	8.69E-07	1.42E+00	0.00E+00	1.01E+00	2.83E-01	8.58E-02	4.15E-02	573 EFPD	30 MW(th)	21 yrs 1 Mo 9/30/03	41%	1.34E-01
032230	1089	2.78E+02	5.51E+01	2.22E+02	1.79E-03	1.00E+01	2.53E+00	5.11E-07	8.38E-01	0.00E+00	5.96E-01	1.66E-01	5.05E-02	2.44E-02	920 EFPD	30 MW(th)	19 yrs 4 Mo 9/30/03	82%	5.98E-01
032230	1090	2.78E+02	5.52E+01	2.39E+02	3.04E-03	1.70E+01	4.30E+00	8.69E-07	1.42E+00	0.00E+00	1.01E+00	2.83E-01	8.58E-02	4.15E-02	920 EFPD	30 MW(th)	19 yrs 4 Mo 9/30/03	69%	4.09E-01
032230	1091TC	2.66E+02	5.27E+01	2.45E+02	5.91E-03	3.30E+01	8.34E+00	1.69E-06	2.76E+00	0.00E+00	1.97E+00	5.49E-01	1.67E-01	8.06E-02	320 EFPD	30 MW(th)	19 yrs 4 Mo 9/30/03	37%	1.94E-01
032230	1092TC	2.65E+02	5.27E+01	2.22E+02	3.22E-03	1.80E+01	4.55E+00	9.20E-07	1.51E+00	0.00E+00	1.07E+00	2.99E-01	9.09E-02	4.40E-02	625 EFPD	30 MW(th)	19 yrs 4 Mo 9/30/03	66%	4.57E-01
032230	1094	2.78E+02	5.52E+01	2.45E+02	5.91E-03	3.30E+01	8.34E+00	1.69E-06	2.76E+00	0.00E+00	1.97E+00	5.49E-01	1.67E-01	8.06E-02	340 EFPD	30 MW(th)	19 yrs 4 Mo 9/30/03	40%	3.27E-01
032230	1096	9.46E+01	1.86E+01	8.50E+01	1.79E-03	1.00E+01	2.53E+00	5.11E-07	8.38E-01	0.00E+00	5.96E-01	1.66E-01	5.05E-02	2.44E-02	295 EFPD	30 MW(th)	21 yrs 1 Mo 9/30/03	46%	9.59E-02
032230	1097	9.43E+01	1.85E+01	8.50E+01	1.79E-03	1.00E+01	2.53E+00	5.11E-07	8.38E-01	0.00E+00	5.96E-01	1.66E-01	5.05E-02	2.44E-02	295 EFPD	30 MW(th)	21 yrs 1 Mo 9/30/03	46%	9.21E-02
032230	1099TC	9.00E+01	1.77E+01	8.00E+01	1.61E-03	9.00E+00	2.28E+00	4.60E-07	7.54E-01	0.00E+00	5.37E-01	1.50E-01	4.54E-02	2.20E-02	295 EFPD	30 MW(th)	21 yrs 1 Mo 9/30/03	49%	1.00E-01
032230	Remainder [1]	1.45E+03	2.88E+02	9.28E+02	2.24E-02	1.25E+02	3.16E+01	6.39E-06	1.05E+01	0.00E+00	7.46E+00	2.08E+00	6.31E-01	3.05E-01	VARIOUS	30 MW(th)	Various 9/30/03	57%	5.62E+00
032230	Total	3.86E+03	7.62E+02	3.04E+03	6.30E-02	3.52E+02	8.90E+01	1.80E-05	2.95E+01	0.00E+00	2.10E+01	5.86E+00	1.78E+00	8.60E-01	VARIOUS	30 MW(th)	Various	Various	8.57E+00

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FUEL RSD FORM
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1 "Remainder" is the composite of the 7 sectioned RERTR fuel elements, ID Numbers 1082, 1086, 1087, 1088, 1093, 1098, and 1100TC.

EFPD = Effective Full Power Days.

Table 2 - Shipper References

Reference Number	Type (Drawing, Letter, etc.)	Title	Number	Revision/Date	Quality Assurance [1]
1	DOE Contract	"General Atomics Hot Cell Decontamination & Decommissioning Project", Phases 2 & 3 Activities; Statement of Work, Task 16" , Attachment I (Note: This is Reference 1-1 called out in Reference 2 below)	Contract DE-AC03-95SF20798		Non QA Record
2	General Atomics Report	HTGR/RERTR Fuel Materials Characterization and Packaging Report	PC-000384	Rev. 2 4/02	QA Record Level NA
3	General Atomics Drawing	Fuel Pin Assembly	T14R210E210	Rev. E, 6/11/91	QA Record Level 1
4	General Atomics Drawing	Temperature Sensing Fuel Pin Assembly and Installation	T14R210E220	Rev. C 11/15/82	QA Record Level 1
5	General Atomics Report	"Post-Irradiation Examination and Evaluation of TRIGA LEU Fuel Irradiated in the Oak Ridge Research Reactor" (Note: This is Reference 5-29 called out in Reference 2 above.)	GA-A18599	5/86	QA Record Level NA
6	General Atomics QA Inspection Report	GA Hot Cell D&D Project Final Assessment Report, "Transfer of HTGR/RERTR Fuel from Hot Cell to Bldg. 30" (Note: This is Reference 1-3 called out in Reference 2 above.)	16A	3/28/96	QA Record Level I
7	General Atomics Internal Correspondence	Greenwood, J. S. to Miller, C. M., "Calculation of Pressure Buildup in Fuel Storage Enclosures"	HCI:015:JSG:96	1/15/96	QA Record Level NA
8	General Atomics Report	"The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel"	E-117-833	2/80	Non QA Record
9	General Atomics Internal Correspondence	Malakhov, V., "Nuclear Safety Evaluation of the Irradiated Fuel Material Interim Storage Facility at General Atomics"	NS:94:VM:399 Nuclear Safety File No. 533.0	6/95	QA Record Level NA
10	General Atomics Quality Assurance Manual	Quality Assurance Manual, General Atomics, San Diego, CA		3rd Edition, Rev. D, 8/12/96	QA Record Level NA
11	General Atomics Internal Correspondence	Nicolayeff, V., "Hot Cell Irradiated Fuel Materials, INEEL Audits of General Atomics"	123:VN:02:18	5/15/02	QA Record Level NA
12	INEEL Correspondence	INEEL Procurement Quality Manager to Razvi, J., GA TRIGA Director, "Supplier Evaluation (TRIGA International)"	MTW-009-01	11/29/00	QA Record Level NA
13	Drawing	NAC International, Top Module, General Atomics IFM, LWT Cask	315-391-120	Rev. 2/ 05/08/03	QA Record Level A

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Reference Number	Type (Drawing, Letter, etc.)	Title	Number	Revision/Date	Quality Assurance [1]
14	Calculation Package	NAC International, Criticality Evaluation for the NAC-LWT with GA Irradiated Fuel Material (IFM)	14661-600	Rev. 0/ 12/16/02	QA Record Level A
15	Drawing	NAC International, Spacer, General Atomics IFM, LWT Cask	315-391-123	Rev.1/ 05/08/03	QA Record Level A
16	Calculation Package	NAC International, Structural and Thermal Evaluation of the Top Module and Spacer	14661-200	Rev. 0/ 12/18/02	QA Record Level A
17	DOE/NRC Form	Certificate of Compliance for Radioactive Material Packages	Form 618 Certificate # 9225	31/ (Later)	QA Record Level A
18	Report	Safety Analysis Report for the NAC Legal Weight Truck Cask	Docket No. 71-9225 T-88004	Rev. LWT-03A 02/03	QA Record Level A
19	DOE/NRC Form	USDOE & USNRC Nuclear Material Transaction Report	DP-741	04/03	QA Record Level 1
20	Form	Fuel RSD Form for the Assembled HTGR Can #32231 for FHU # 315-391-120-44-001	434.28	NR/ 06/09/03	QA Record Level 1
21	Memo	Hot Cell IFM - HTGR/RERTR Enclosure Pressure Calculation	123:VN:03:08	Rev. 0/ 03/19/03	QA Record Level 1
22	Procedure	NAC-LWT Cask Generic Operating Procedure	NAC 315-P-02	Rev.11/ 6/13/03	QA Record Level A

[1] Describe quality level of reference. State "Non-QA Record" if not a quality assurance record.

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for ASSEMBLED RERTR CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
I. SHIPPER AND SHIPMENT IDENTIFICATION INFORMATION		
1. Applicable Fuel RSD Form for this Packaging RSD	Fuel RSD Form for Assembled RERTR Can #032230 for FHU # 315-391-120-44-001. This Packaging RSD Form includes a revised list of References from the RERTR Fuel RSD Forms and is included as Table 3 at the end of this document.	
III. SNF PACKAGING		
1. FHU description.	The FHU is the Top Module (NAC Drawing #315-391-120) supplied by NAC. It is a right circular cylinder in shape 43.7" (111 cm) long by 13.265" (33.69 cm) in diameter fabricated out of 304 SS pieces. The Top Module consists of three (3) support plates, two (2) tubes to contain the HTGR and RERTR Cans, four (4) guide bars (made from plate) to align the support plates, four (4) grapple lift plates to allow the Top Module to be lifted, two (2) spacer plates to space the grapple lift plates, and two (2) guide plates to orient the tubes (Ref. 13). Inserted in the FHU are two (2) Can Spacers and the two (2) Cans.	Reference 13
	Can Spacers: (NAC Drawing #315-391-120, part 97 [HTGR] and #315-391-120, part 98 [RERTR]), are fabricated out of 4" (10.16 cm) dia. 304 SS pipe. Welded at the top and bottom of the pipe are 1/4" (0.64 cm) thick 304SS plates. These Can Spacers are used to assure that the Cans are at the same elevation at the top of the Top Module to minimize movement during shipping.	References 27 and 28
	Cans: One for the HTGR SNF #032231 (see HTGR Packaging RSD Form for details of this can) and the other for the RERTR SNF #032230 (the subject of this Packaging RSD). The visible portion of the RERTR Can is the Secondary Enclosure. It is a right circular cylinder constructed of 304 SS tubing; 37.0" (93.98 cm) tall, 0.12" (0.30 cm) thick wall and 4.75" (12.07 cm) dia. End caps constructed of Inconel 600 rolled plates and 0.5" (1.27 cm) thick are seal welded to the tube. Inconel 625 weld material was used to weld the anchors of the lifting bail onto the top end cap.	Fig. 6.4 of Reference 2

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Description	Shipper Summary	References (See Table 3)
1. FHU description (cont.).	Cans (cont.) A stainless steel lifting bail is permanently attached to the top of the Can for lifting into the FHU. The lifting bail is crimped together using a copper end sleeve. The overall height of the Can is 37.25" (94.62 cm.), with the lifting bail in the down position. Sealed inside of the secondary enclosure is the primary enclosure consisting of another right circular cylinder constructed of 304 SS tubing, 35.5" (90.27 cm) tall, 0.12" (0.30 cm) thick wall and 4.25" (10.80 cm) dia. End caps constructed of Inconel 600 rolled plate and 0.5" (1.27 cm) thick are seal welded to both ends of the tube. Inside the primary enclosure are the collected RERTR fuel units.	Fig. 6.4 of Reference 2
2. Method for the identification of each FHU	The FHU is identified with steel stamped letters, 0.03" (0.08 cm) deep, filled with black weather resistant paint. The RERTR Can is identified with stamped ¼" (0.64 cm) high characters on the top end cap and stencil painted 1" (2.54 cm) high characters along the side of the Can.	Reference 13 Fig. 6.4 of Reference 2
3. FHU specific data for each FHU, include as Table 1, updated with unique FHU identification number that identifies which fuel unit(s) are in each FHU, or equivalent.	See Table 1 of the Fuel RSD Form for RERTR Can #032230 for details of the fuel units that are in the Can.	Reference 2
4. Method for fuel unit/FHU location identification in the SNF packaging if the fuel is shipped and stored in a can, liner, basket, or other container.	Each fuel unit is identified with a unique serial number. However, they are all sealed in the primary enclosure, which in turn is sealed within the secondary enclosure. The secondary enclosure Can has a unique identifying number engraved on the top and painted on the side.	Fig. 6.4 of Reference 2
5. Detailed drawings (most current drawings, indicating revision number and date) and list of materials. Describe any deviations from the drawings.	NAC International Drawing No. 315-391-120, Rev. 2, dated 5/8/03	Reference 13
	GA Drawing No. 032230, Rev. A, Dated 1/9/96 (Secondary Enclosure). GA Drawing No. 322236, Rev. B. Dated 1/3/96 (Primary Enclosure).	Fig 6.5 of Reference 2
	Deviations from the drawings of the Can were minor and did not affect its form, fit, or function. Deviations are described in Nonconformance Reports presented in Ref. 6.	Reference 6

PACKAGING RSD FORM
for ASSEMBLED RERTR CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
	Deviations from the drawings of the Top Module and Can Spacers were minor and did not affect its form, fit, or function. The fabrication and test packages for the Top Module and Can Spacers is given in Reference 26.	Reference 26
a. Physical description of each type of SNF packaging:	There is only one SNF Package. See Shipper Summary to III, 1. above	Fig 6.4 of Reference 2 and Reference 13
b. Materials of construction:	304 SS, Inconel 600, Inconel 625, Copper	Figs. 6.5 & 6.6 of Reference 2 and Reference 13
c. Dimensions (cm):		
i. Total length:	111.0 cm	Reference 13
ii. Cross-section dimensions:	33.69 cm	Reference 13
d. Weight (g):		
i. Empty:	82,327 g for the Top Module,	Reference 13
ii. Maximum loader:	32,432 g for the HTGR Can and 34,474 g for the RERTR Can plus the weight of the Top Module and Spacers for a total loaded weight of 149,233 g.	References 2 and 13
6. List the extraneous material and its mass in grams associated with the SNF packaging or its contents:		
a. Pyrophoric or reactive material	none	Reference 2
b. Inert materials	Erbium, Zirconium, Incoloy 800H	Tables 2.3, 2.4 and 2.5 of Reference 2
c. Organic materials	none	Reference 2
d. Water		
i. As free water	none	Reference 2
ii. As associated water		
A. Chemically bound	none	Reference 2
B. Physically bound	none	Reference 2
e. Other (specify)	none	Reference 2
7. Describe the following, including chemical composition:		
a. Exposure to contaminants (Hg, halides, etc.). If any, how was the contamination removed?	There was no exposure to Hg, halides, etc. Confirmation of this is given in the material certifications from the vendors and presented in Reference. 6.	Reference 6
b. Any external coatings applied to the SNF	none	Reference 2

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Description	Shipper Summary	References (See Table 3)
c. Special additives	none	Reference 2
8. Activation product contents and curies.	There are no activation products in the Top Module or the Primary and Secondary Enclosures since there were never exposed to neutrons. The enclosed SNF packages were exposed to neutrons. The activation products of the Zr if any were undetectable. The only detectable activation products are those within the fuel cladding. These are: ^{54}Mn = 2.14 E-05 Ci ^{55}Fe = 4.27 E+00 Ci ^{60}Co = 8.84 E-01 Ci ^{59}Ni = 3.30 E-01 Ci ^{63}Ni = 3.75 E+01 Ci ^{99}Tc = 1.40 E-01 Ci	Reference 2 updated to 09/30/03
9. Metallurgical state of the SNF packaging (annealed, stressed, sensitized, estimated exposure [nvt], etc.)	The loaded Top Module and Primary and Secondary Enclosures were not exposed to neutrons. The exposure (nvt) of the enclosed SNF and cladding was 5.0E+21 n-cm^2 (fast). The fuel items are all sealed in a 304 SS tube that was annealed and pickled.	Reference 2
10. Describe external contamination, particularly alpha contamination levels.	On the surface of the RERTR Can, the contamination levels were, as of 12/1/95: 2700 dpm/100 cm^2 - ; 13 dpm/100 cm^2 . The FHU Top Module and associated Can Spacers are new and clean and thus will not be contaminated prior loading at GA. However, it might become contaminated during loading. The maximum contamination level would be if all of the contamination from the RERTR and HTGR Cans is transferred to the Top Module. This maximum level would be 5,700 dpm/100 cm^2 - ; 27 dpm/100 cm^2	Reference 2

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Description			Shipper Summary	References (See Table 3)
NOTE: The following subsections of Section III are applicable to canned fuel (cans, tubes, containers, etc.) only. If canned fuel is stored in an overpack canister (e.g., dual purpose canister), this section must be completed separately for both the fuel can and the overpack can.)				
The data below is for the fuel can, (Primary Enclosure) GA Dwg. # 032236/B				Fig. 6.5 of Reference 2
11.	Provide a design analysis of the can to include:			
	a.	Identify the purpose of all penetrations	There are no penetrations.	Reference 2
	b.	Thermal analysis of can and contents	A thermal analysis of the can is not applicable. The temperature of the fuel units in the Can are at room temperature as is the Can. There is currently only 8.57 watts of decay heat generated. This is easily dissipated by the Can.	Reference 2 updated to 09/30/03 and included in Table 1
	c.	Integrity test results agreed upon by the INEEL	A helium leak test of the bottom end cap of the Can assembly per QDI LTH-S-3801 with an acceptable leak rate of "non-detected" (at a detection limit of 4.4 E-08 standard cc/sec was satisfactory. A hydro test of the tube at the fabrication facility of 450 psi minimum was performed and was satisfactory. After loading of the SNF into the primary closure the top end cap was placed on the tube inside the hot cell facility. The tube was mounted on a turn table with a variable rotational speed capability. The tube was rotated at a predetermined speed and the top end cap was welded remotely using an in-cell weld head. The weld integrity was inspected using an in-cell mirror and an out-cell telescope for visual inspection. The weld was judged satisfactory by the QA inspector.	Reference 6
	d.	State the maximum allowable pressure on sealed can (e.g., could the can build up pressure?)	The pressure at which the weld at the end caps will begin to yield is 683 psi. For a safety factor of 2, the yield point is 341 psi. The ultimate pressure at which the ends caps will pop off is 1,367 psi. Based on a buildup rate of 3.25 E-03 psi/year and assuming a linear rise in pressure (actual buildup will decrease exponentially), the end cap will pop off in 420,000 years.	Reference 21
			Can pressure buildup is negligible (calculated to be 3.25 E-03 psi/year.	Reference 7

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Description	Shipper Summary	References (See Table 3)
e. Drop test results, including extent of damage	Drop calculations have been performed for the load Top Module by BBWI for the IFSF Plant Safety Documents. This report is Ref. 23. The analyzed lift height for the fuel basket or an individual package is 45" over the top surface of the transfer car's steel slabs. The end cap of the secondary enclosure is severely damaged during most of the end drops and has significantly less containment integrity but the end cap will remain in place and provide a lesser level of containment. The damage to the primary enclosure is significantly less than the damage to the secondary enclosure. There is some material failure of the primary enclosure end cap (impacting end). This may reduce the level of confinement of the primary enclosure by a small amount, which is not defined because material failure does not develop into a gap from which contents could readily escape. The damage to the packages during a seismic event will be less than any of the analyzed postulated drops.	Reference 23
f. Other test reports	o Welder certifications were all satisfactory. o Weld integrity was satisfactory. o Transverse tension, flattening and reverse flattening tests satisfactory. o Flange and reverse bend tests satisfactory. o Yield strength confirmed. o Factory chemical and mechanical tests on tube and end cap materials were satisfactory.	Reference 6
g. Material certification papers for can and seal	Certification papers for the can and seal are provided in Reference 6.	Reference 6
h. Gas generation rates within the can	Pressure buildup within the can is calculated to be 3.25E-03 psi/year. This is generated from the generation of helium gas resulting from the contained -emitting radioactive species.	Reference 7
	There is no hydrogen disassociation from the hydrided fuel. The disassociation pressure of the hydrogen is 0.01 atm at 580° C. It basically zero at Standard Temperature and Pressure.	Reference 8
12. State the conditions under which the contents were canned (e.g., canned wet, canned dry, etc.)	Canned dry in atmospheric air at ambient temperature and pressure. The HVAC was functional during the packaging of the contents. It is expected that the relative humidity was about 50% during packaging.	Reference 6
13. If the can was subject to a wet environment and will be stored dry, describe the drying process to include the following:	Can not subjected to wet environment.	Reference 6
a. Drying process	n/a	
b. Dryness criteria	n/a	

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	Description	Shipper Summary	References (See Table 3)
	c. Dryness criteria verification compliance method	n/a	
14.	List can back fill gas and pressure	No back fill gas.	Reference 6
15.	Contents of the can verification method. The verification method shall be onsite inspection or approved alternate. Provide photographs of contents if available.	All purchase orders for material for the can were reviewed and approved by a GA QA inspector. Copies of the purchase orders are included in Reference 6.	Reference 6
		The parts for the can were all receipt inspected by GA's QA department. Any testing performed was also witnessed by a GA QA inspector. This documentation is included in Reference 6.	Reference 6
		The loading of the can was visually observed by a qualified GA QA Inspector per procedure GA Document Number HCP-6-6, Issue A. As steps were completed in the procedure, a GA QA inspector initialed them to confirm they were completed. This document is included with Reference 6.	Reference 6
		The GA Hot Cell Facility was designated as an MBA (Material Balance Area) and all SNM coming into and out of the facility was documented and a log sheet of all SNM maintained by the designated material custodian. Material receipt forms are on file at GA along with regual inventories. Qualifications for the material custodian are documented in GA's QA manual. The SNM material that went into the can was confirmed by the material custodian and a GA QA inspector and appropriate adjustments made to the MBA account of SNM.	
		Qualifications for a QA inspector are listed in the GA QA Manual (Reference 10.)	Reference 10
		GA's QA program has been routinely audited by both the NRC and DOE. References 12 and 24 list audits of GA that were conducted or observed by INEEL personnel.	References 12 and 24
16.	Describe the distribution of the fuel [fuel unit(s)] over the length of the can.	Distribution is shown schematically in Fig. 6.4 of Reference 2.	Reference 2

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Description	Shipper Summary	References (See Table 3)
The following data applies to the fuel overpack (Secondary Enclosure) GA Dwg #032230/A		See Fig. 6.6 of Reference 2
11. Provide a design analysis of the can to include:		
a. Identify the purpose of all penetrations	There are no penetrations.	Reference 2
b. Thermal analysis of can and contents	A thermal analysis of the can is not applicable. The temperature of the fuel units in the Can are at room temperature as is the Can. There is currently only 8.57 watts of decay (as of 09/30/03) heat generated by the fuel items. This is easily dissipated by the Can.	Reference 2 updated to 09/30/03 and included in Table 1
c. Integrity test results agreed upon by the INEEL	A helium leak test of the bottom end cap of the Can assembly per QDI LTH-S-3801 with an acceptable leak rate of "non-detected" (at a detection limit of 4.4E-08 standard cc/sec) was satisfactory. A hydro tests of the tube at the fabrication facility of 450 psi minimum was performed and was satisfactory. After loading of the primary closure into the secondary closure, the top end cap was placed on the tube inside the hot cell facility. The tube was mounted on a turn table with a variable rotational speed capability. The tube was rotated at a predetermined speed and the top end cap was welded remotely using an in-cell weld head. The weld integrity was inspected using an in-cell mirror and an out-cell telescope for visual inspection. The weld was judged satisfactory by the QA inspector.	Reference 6
d. State the maximum allowable pressure on sealed can (e.g., could the can build up pressure?)	There is no can pressure buildup. Pressure is contained within the Primary Enclosure. The Can was hydrotested to 450 psi.	Reference 6

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Description	Shipper Summary	References (See Table 3)
e. Drop test results, including extent of damage	Drop calculations have been performed for the load Top Module by BBWI for the IFSF Plant Safety Documents. This report is Ref. 23. The analyzed lift height for the fuel basket or an individual package is 45" over the top surface of the transfer car's steel slabs. The end cap of the secondary enclosure is severely damaged during most of the end drops and has significantly less containment integrity but the end cap will remain in place and provide a lesser level of containment. The damage to the primary enclosure is significantly less than the damage to the secondary enclosure. There is some material failure of the primary enclosure end cap (impacting end). This may reduce the level of confinement of the primary enclosure by a small amount, which is not defined because material failure does not develop into a gap from which contents could readily escape. The damage to the packages during a seismic event will be less than any of the analyzed postulated drops.	Reference 23
f. Other test reports	o Proof load test to 165 lbs. (74,843 gm) of lifting cable assembly were satisfactory. o Welder certifications were all satisfactory. o Weld integrity was satisfactory. o Transverse tension, flattening and reverse flattening tests satisfactory. o Flange and reverse bend tests satisfactory. o Yield strength confirmed..	Reference 6
g. Material certification papers for can and seal	Certification papers for the can and seal are provided in Reference 6.	Reference 6
h. Gas generation rates within the can	Since the fuel items are sealed within the Primary Enclosure, there is no method of gas generation within the Secondary Enclosure.	Reference 2
12. State the conditions under which the contents were canned (e.g., canned wet, canned dry, etc.)	Canned dry in atmospheric air at ambient temperature and pressure. Since the HVAC was operational, it is expected that the relative humidity was 50%.	Reference 6
13. If the can was subject to a wet environment and will be stored dry, describe the drying process to include the following:	Can not subjected to wet environment.	Reference 6
a. Drying process	n/a	
b. Dryness criteria	n/a	
c. Dryness criteria verification compliance method	n/a	
14. List can back fill gas and pressure	No back fill gas.	Reference 6

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	Description	Shipper Summary	References (See Table 3)
15.	Contents of the can verification method. The verification method shall be onsite inspection or approved alternate. Provide photographs of contents if available.	All purchase orders for material for the can were reviewed and approved by a GA QA inspector. Copies of the purchase orders are included in Reference 6. Component parts for the Can were all receipt inspected by GA's QA department. Any testing performed was also witnessed by a GA QA inspector. This documentation is included in Reference 6. The loading of the Can was visually observed by a qualified GA QA Inspector per procedure GA Document Number HCP-6-6, Issue A. As steps were completed in the procedure, a GA QA inspector initialed them to confirm they were completed. This document is included with Reference 6. The GA Hot Cell Facility was designated as an MBA (Material Balance Area) and all SNM coming into and out of the facility was documented and a log sheet of all SNM maintained by the designated material custodian. Material receipt forms are on file at GA along with regual inventories. Qualifications for the material custodian are documented in GA's QA manual. The SNM material that went into the can was confirmed by the material custodian and a GA QA inspector and appropriate adjustments made to the MBA account of SNM. Qualifications for a QA inspector are listed in the GA QA Manual (Reference 10.) GA's QA program has been routinely audited by both the NRC and DOE. References 12 and 24 list the recent audits of GA that were conducted or observed by INEEL personnel.	Reference 6 Reference 6 Reference 6 Reference 10 References 12, and 24
16.	Describe the distribution of the fuel [fuel unit(s)] over the length of the can.	Distribution is shown schematically in Fig. 6.4 of Reference 2.	Reference 2

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Description				Shipper Summary	References (See Table 3)
IV.	HANDLING AND RIGGING FIXTURES (rigging, slings, spreader bars, tools, yokes fuel handing and storage fixtures, etc.) - To be filled out for the fixtures to be used at the INEEL only. If no shipper or carrier supplied fixtures are to be used at the INEEL, clearly state this.				
	1.	Detailed drawings (most current drawings, indicating revision number and date) and list of materials for each fixture. Describe any deviations from the drawings.		The grapple to be used for removal of the FHU is a modified version of NAC's standard TRIGA handling item (Drawing # 61102, Reference 29) that has been modified by Change Request DCR 61102-024-1A (Reference 30). The secondary yoke (Drawing # 315-390-29, Rev. 2) will be used for the transfer process (Reference 31). The load test confirming the load carrying capability is in Reference 32. The remaining items used for the transfer are part of the standard set of fixtures for TRIGA transfer and therefore are on INEEL's site.	References 29 and 30 References 31 and 32 Reference 13
		a.	Dimensions (cm):		
			i. Total length	N/A	
			ii. Cross-sectional dimensions	N/A	
		b.	Weight (g):	N/A	
		c.	Type of load bar (if applicable):	N/A	
		d.	Special fuel, lid, SNF packaging or transportation packaging removal tools. State size and description of lid bolts.	N/A	
	2.	Design life of the fixture. Provide analysis and certification to support the design life, including certification of the material of construction.		N/A	
	3.	Provide certification of the most recent load tests performed on all lifting hardware (package lifting trunnions, devices and fixtures) accompanying the shipment.		N/A	
	4.	State if the fixture expendable or reusable.		N/A	
	5.	Describe the use of each fixture. If any special tools are required, describe them in detail and provide drawings. Include precautions for use of the fixture and tools.		N/A	

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Description	Shipper Summary	References (See Table 3)
V. TRANSPORTATION PACKAGE (SHIPPING CASK) DATA		
1. Identify the transportation package (cask), its model number and its serial number and transport vehicle to be used.	Transportation package: NAC-LWT Cask in a 20' ISO, mounted on a Tri-State drop deck trailer. Model number: NAC-LWT Serial number: Will be either #1 through #8. The exact Cask has not been identified yet, but will be prior to shipping. Transport vehicle: Tractor and trailer supplied by Tri-State Motor Transit Company.	Reference 22
2. Carrier Identification	Tri-State Motor Transit Company, Joplin, MO	Reference 22
3. Furnish 1 copy of the current Department of Transportation (DOT) Certificate of Competent Authority or the DOE/NRC Certificate of Compliance (C of C), or equivalent, as applicable.	An amendment to the original C of C (Revision 34) has been approved by the NRC. This has been supplied to INEEL.	Reference 17
4. Furnish 1 copy of the Safety Analysis Report For Packaging (SARP) or equivalent, including the Shipping Package Transport Plan if applicable.	"Safety Analysis Report for the NAC Legal Weight Truck" has been supplied. INEEL is an approved user of the NAC-LWT cask.	Reference 18
5. Provide 1 copy of reproducible detailed drawings of the packaging which indicate:	This information is in the SAR listed above.	Reference 18
a. Dimensions	The overall length is 199.8" (507.5 cm) and the maximum diameter is 44.2" (112 cm).	Reference 18
b. Weight (gross and net)	The maximum weight of the package is 48,000 lbs and the maximum weight of the contents and basket is 4,000 lbs.	Reference 18
c. Surface finish (e.g., roughness, painting, coating, etc.)	125 rms or better on all surfaces. Surface electropolished 304 SS	Reference 18
d. Materials of construction	The cask body consists of a 0.75 " (1.91 cm) thick XM19 SS inner shell, a 5.75 " (14.61 cm) thick lead gamma shield, a 1.2" (3.05 cm) thick XM19 SS outer shell, and a neutron shield tank. The inner and outer shells are welded to a 4" (10.16 cm) thick 304 SS steel bottom end forging. The cask bottom consists of a 3" (7.62 cm) thick 20.75 " (52.71 cm) dia. lead disk enclosed by a 3.5 " (8.89 cm) thick 304 SS plate and bottom end forging. The cask lid is 11.3" (28.7 cm) thick 304 SS stepped design	Reference 18

PACKAGING RSD FORM
for ASSEMBLED RERTR CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
e. Size and description of lid bolts	Lid bolts are twelve (12) ea. Grade 1660, Class A 1-UNC x 9 long socket head cap screws.	Reference 18
f. Poison, poison inserts or spacing insert necessary for shipping	There are no poison inserts. The cask has a neutron shield tank containing an ethylene glycol/water solution that is 1% boron by weight. The spacing insert is shown in NAC Drawing 315-391-123 (Reference 15).	Reference 18 Reference 15
6. Describe the poison, poison inserts or spacing insert necessary for shipping and furnish certification of integrity.	No poison inserts. The neutron shield tank contains ethylene glycol/water solution that is 1% boron by weight. There is a spacing insert that is loaded in the bottom of the shipping cask prior to loading of the FHU. The spacing insert is of right circular cylinder shape. It is constructed of 304 SS. Including the guide pins it is 135.25" (343.54 cm.) long by 13.265" (33.69 cm) in diameter. There are two (2) guide pins made from bar stock, one (1) spacer tube, five (5) support plates, two (2) grapple plates, four (4) spacer plates, and eight (8) guide bars. The structural and thermal evaluation of the spacer is included in Reference 16. The cask integrity is included in Reference 18.	Reference 16 Reference 18
7. Provide a copy of loading, handling, and dry cask storage procedures (if applicable).	The NAC generic operating procedure is Reference 22. The NAC-LWT loading procedure at GA is Reference 25.	Reference 22 Reference 25
8. Furnish tie-down configuration of the package to the transport vehicle.	There are no drawings available. The cask ISO is mounted on the trailer using standard ISO twist locks.	
9. Furnish tie-down configuration of the package to the transport vehicle.	There are no drawings available. The cask ISO is mounted on the trailer using standard ISO twist locks.	
VI. ENVIRONMENTAL SAFETY AND HEALTH		
1. Provide copies of any documents covering criticality safety evaluations and calculations which determine the minimum critical number of pieces of the subject fuel and which evaluate the criticality safety of shipping, handling and storing the fuel.	References 9 and 14	References 9 and 14
2. Provide a copy of all existing NEPA documents for the storage and handling of this fuel. If none, state none.	None	

PACKAGING RSD FORM
for ASSEMBLED RERTR CAN #032230 for FHU # 315-391-120-44-001

Description	Shipper Summary	References (See Table 3)
VIII. QUALITY ASSURANCE		
1. Reference attached documentation that the shipper's QA program has been approved by the INEEL.	At the time of packaging (1995), the Shipper's QA program was approved by INEEL (see Reference 11). The next INEEL audit of GA was conducted in November 2000, by Bechtel BWXT Idaho. This audit focused on TRIGA fuel design and fabrication (see Reference 12), and resulted in the approval of General Atomics TRIGA International Group as a Qualified Supplier. This approval status was restricted to the supply of TRIGA Reactor Fuel to BBWI, purchased through the INEEL University Fuels Program. The latest audit (Reference 24) of GA was conducted in April 2003 by J. H. Valentine for the purpose of approving GA for shipment of fuel to INEEL.	Reference 11 Reference 12 Reference 24
X. OTHERS		
1. Provide any information not provided above that may have an impact on the shipper's SNF receipt and storage or INEEL Operations.	None	

Prepared By:

J. S. Greenwood/Physicist-in-Charge, TRIGA Reactor Facility — Signature: J.S. Greenwood — Date: 9/11/03

Approved By:

R. I. De Velasco, Manager, Decommissioning Projects — Signature: De Velasco — Date: 9/11/03

PACKAGING RSD FORM
for ASSEMBLED RERTR CAN #032230 for FHU # 315-391-120-44-001

Table 3 - Shipper References

Reference Number	Type (Drawing, Letter, etc.)	Title	Number	Revision/Date	Quality Assurance [1]
1	DOE Contract	"General Atomics Hot Cell Decontamination & Decommissioning Project", Phases 2 & 3 Activities; Statement of Work, Task 16" , Attachment I. (Note: This is Reference 1-1 called out in Reference 2 below)	Contract DE-AC03-95SF20798		Non QA Record
2	General Atomics Report	HTGR/RERTR Fuel Materials Characterization and Packaging Report	PC-000384	Rev. 2 4/02	QA Record Level NA
3	General Atomics Drawing	Fuel Pin Assembly	T14R210E210	Rev. E, 6/11/91	QA Record Level 1
4	General Atomics Drawing	Temperature Sensing Fuel Pin Assembly and Installation	T14R210E220	Rev. C, 11/15/82	QA Record Level 1
5	General Atomics Report	"Post-Irradiation Examination and Evaluation of TRIGA LEU Fuel Irradiated in the Oak Ridge Research Reactor" (Note: This is Reference 5-29 called out in Reference 2 above.)	GA-A18599	5/86	QA Record Level NA
6	General Atomics QA Inspection Report	GA Hot Cell D&D Project Final Assessment Report, "Transfer of HTGR/RERTR Fuel from Hot Cell to Bldg. 30" (Note: This is Reference 1-3 called out in Reference 2 above.)	16A	3/28/96	QA Record Level I
7	General Atomics Internal Correspondence	Greenwood, J. S. to Miller, C. M., "Calculation of Pressure Buildup in Fuel Storage Enclosures"	HCI:015:JSG:96	1/15/96	QA Record Level NA
8	General Atomics Report	"The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel"	E-117-833	2/80	Non QA Record
9	General Atomics Internal Correspondence	Malakhov, V., "Nuclear Safety Evaluation of the Irradiated Fuel Material Interim Storage Facility at General Atomics"	NS:94:VM:399 Nuclear Safety File No. 533.0	6/95	QA Record Level NA
10	General Atomics Quality Assurance Manual	Quality Assurance Manual, General Atomics, San Diego, CA		3rd Edition, Rev. D, 8/12/96	QA Record Level NA
11	General Atomics Internal Correspondence	Nicolayeff, V., "Hot Cell Irradiated Fuel Materials, INEEL Audits of General Atomics"	123:VN:02:18	5/15/02	QA Record Level NA
12	INEEL Correspondence	INEEL Procurement Quality Manager to Razvi, J., GA TRIGA Director, "Supplier Evaluation (TRIGA International)"	MTW-009-01	11/29/00	QA Record Level NA
13	Drawing	NAC International, Top Module, General Atomics IFM, LWT Cask	315-391-120	Rev. 2/ 05/08/03	QA Record Level A
14	Calculation Package	NAC International, Criticality Evaluation for the NAC-LWT with GA Irradiated Fuel Material (IFM)	14661-600	Rev. 0/ 12/16/02	QA Record Level A

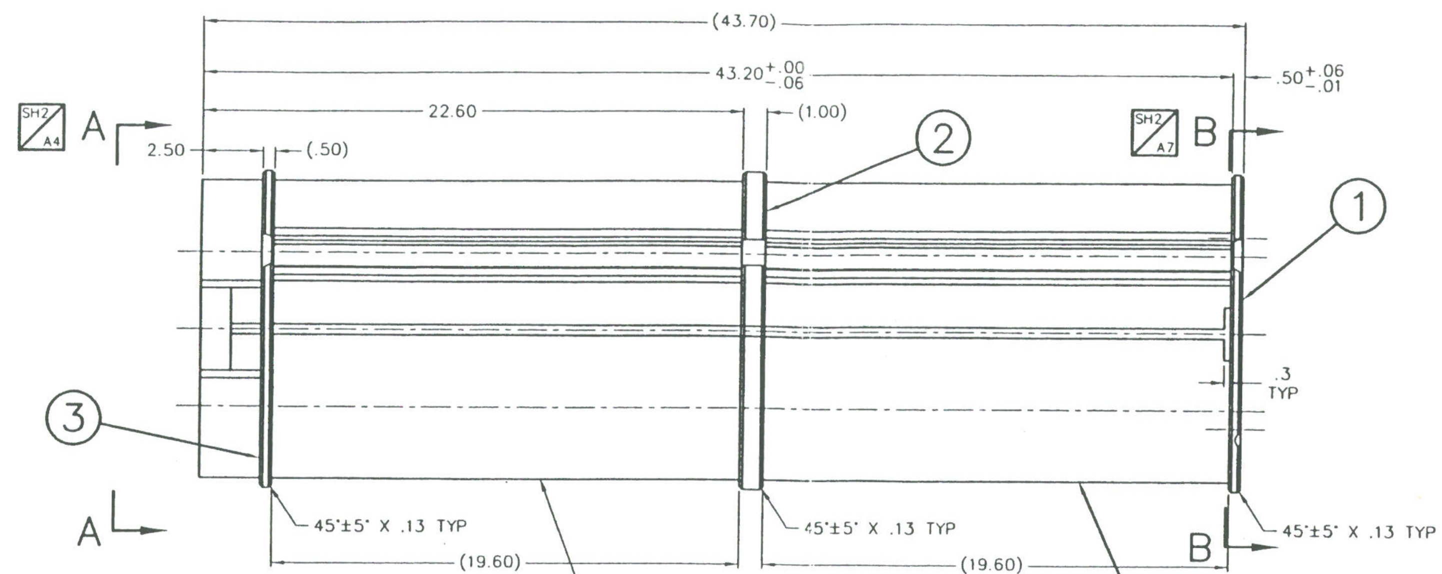
PACKAGING RSD FORM
for ASSEMBLED RERTR CAN #032230 for FHU # 315-391-120-44-001

Reference Number	Type (Drawing, Letter, etc.)	Title	Number	Revision/Date	Quality Assurance [1]
15	Drawing	NAC International, Spacer, General Atomics IFM, LWT Cask	315-391-123	Rev.1/ 05/08/03	QA Record Level A
16	Calculation Package	NAC International, Structural and Thermal Evaluation of the Top Module and Spacer	14661-200	Rev. 0/ 12/18/02	QA Record Level A
17	DOE/NRC Form	Certificate of Compliance for Radioactive Material Packages	Form 618 Certificate # 9225	Rev. 34/ June 30, 2003	QA Record Level A
18	Report	Safety Analysis Report for the NAC Legal Weight Truck Cask	Docket No. 71-9225, T-88004	Rev. 34 11/02	QA Record Level A
19	DOE/NRC Form	USDOE & USNRC Nuclear Material Transaction Report	DP-741	04/03	QA Record Level 1
20	Form	Fuel RSD Form for the Assembled HTGR Can #32231 for FHU # 315-391-120-44-001	434.28	NR/ 06/09/03	QA Record Level 1
21	Memo	Hot Cell IFM - HTGR/RERTR Enclosure Pressure Calculation	123:VN:03:08	Rev. 0/ 03/19/03	QA Record Level 1
22	Procedure	NAC-LWT Cask Generic Operating Procedure	NAC 315-P-02	Rev.11/ 6/13/03	QA Record Level A
23	Calculation Package	BBWI, Drop Analysis of the General Atomics HTGR/RERTR Fuel Packaing at IFSF	EDF-3446	Initial Issue March 3, 2003	QA Record Level A
24	INEEL Letter	From J. H. Valentine to J. S, Greenwood, "Approval of the General Atomics Quality Assurance Program for Shipment of the General Atomics Spent Nuclear Fuel to the INEEL"	CCN 41779	April 24, 2003	QA Record Level NA
25	Procedure	GA Hot Cell D&D Project: Irradiated Fuel Materials Shipment	DDP-1.12	Issue A 9/2/03	QA Record Level 1
26	Fabrication Package	Columbiana Hi Tech Data Package, Top Module and Spacer, NAC PO #03-0157.	Columbiana Hi Tech WO#03-003	August 21, 2003	QA Record Level 1
27	Drawing	NAC International, HTGR Spacer, General Atomics IFM,LWT Cask	315-391-120-97	Rev. 2/ 5/8/03	QA Record Level A
28	Drawing	NAC International, RERTR Spacer, General Atomics IFM,LWT Cask	315-391-120-98	Rev. 2/ 5/8/03	QA Record Level A
29	Drawing	NAC International, Grapple Assembly, INEEL, CPP	61102-024	Rev. 2 9/2/03	QA Record Level A
30	Design Change Request	Grapple Assembly, INEEL, CPP	61102-024-1A	Original Issue 8/29/03	QA Record Level A
31	Drawing	NAC International, Secondary Yoke, LWT Cask	315-390-29	Rev. 2	QA Record Level A
32	Certification	Load Certification for Secondary Yoke			

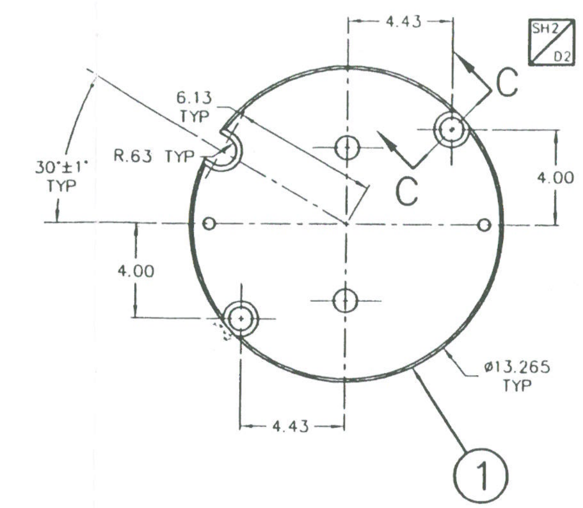
APPENDIX B
ENGINEERING DRAWINGS FOR NAC-LWT TOP MODULE (BASKET) AND SPACERS

L 152695

REV	CHANGE
0	INITIAL ISSUE
1	INC DCR 0A
2	INC DCR 1A



99 TOP MODULE
WEIGHT: 175#
(HIDDEN LINES REMOVED FOR CLARITY)



10. TOP MODULE (ASSY -99) HAS A RATED LOAD CAPACITY OF 152#.

9 LOAD BEARING WELD

8. TOP MODULE IS DESIGNED FOR USE WITH, AND COMPATIBLE WITH, THE NAC GRAPPLE(S) AND THE NAC-LWT CASK ASSEMBLY

7. UNLESS OTHERWISE NOTED, WELDING TO BE IN ACCORDANCE WITH SECTION IX OF THE ASME PRESSURE VESSEL CODE.

6. AFTER LOAD TESTING VISUALLY INSPECT (VT) ALL ACCESSIBLE WELDS IN ACCORDANCE WITH NOTE 4.

5. PENETRANT TEST (PT) FINAL PASS ALL WELDS EXCEPT SEAL WELDS. EXAMINE PER ASME SECTION V, ARTICLE 6, WITH ACCEPTANCE PER SECTION III, ARTICLE NG-5350.

4. VISUALLY INSPECT (VT) ALL WELDS. EXAMINE PER ASME SECTION V, ARTICLE 9, WITH ACCEPTANCE PER SECTION III, ARTICLE NG-5360.

3. BASKET SHALL BE LOAD TESTED AT 150% OF RATED LOAD CAPACITY WITH NO PERMANENT DEFORMATION ALLOWED.

2 EACH ITEM #4 (TUBE) TO ACCEPT A FULL LENGTH, ø5.40 GO GAGE.

1 STEEL STAMP/ENGRAVE APPROX. AS SHOWN. LETTERS .50 HIGH X .03 MAX. DEEP. FILL WITH KEELER AND LONG POLY-SILICON ENAMEL P SERIES. nnn REPRESENTS A UNIQUE IDENTIFICATION NUMBER IN ACCORDANCE WITH THE FABRICATION SPECIFICATION. "yyy" IS THE ACTUAL WEIGHT AT THE TIME OF FABRICATION.

NOTES:

FOR INFORMATION ONLY

97	98	99	ITEM	NAME	MATERIAL	SPEC	DRAWING No.	DESCRIPTION	Q CAT
1			11	HTGR SPACER TUBE	304 ST. STL.	ASTM A312		4" PIPE, SCH 40	-C-
	1		10	RERTR SPACER TUBE	304 ST. STL.	ASTM A312		4" PIPE, SCH 40	-C-
2	2		9	PLATE	304 ST. STL.	ASTM A276/A240		1/4 BAR/PLATE	-C-
		2	8	GUIDE PLATE	304 ST. STL.	ASTM A240		5/16 PLATE	-A-
		2	7	SPACER PLATE	304 ST. STL.	ASTM A240		5/16 PLATE	-A-
		4	6	GRAPPLE LIFT PLATE	304 ST. STL.	ASTM A240		5/16 PLATE	-A-
		4	5	GUIDE BAR	304 ST. STL.	ASTM A276/A240		1/2 BAR/PLATE	-C-
		2	4	TUBE	304 ST. STL.	ASTM A511		6.00 OD X .25 WALL	-A-
		1	3	UPPER SUPPORT PLATE	304 ST. STL.	ASME SA240		1/2 PLATE	-A-
		1	2	MIDDLE SUPPORT PLATE	304 ST. STL.	ASME SA240		1 PLATE	-A-
		1	1	LOWER SUPPORT PLATE	304 ST. STL.	ASME SA240		1/2 PLATE	-A-
ASSY	ASSY	ASSY							
QUANTITY									

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NAC INTERNATIONAL

DIMENSIONING AND TOLERANCING SHALL BE PER ASME Y14.5-94 UNSPECIFIED DIMENSIONS AND TOLERANCES SHOWN BELOW. DIMENSIONS ARE IN INCHES. FRACTIONAL TOLERANCE: ±1/8

SYM	GEOMETRY	.XXX	TOL	.XX	TOL
	FLATNESS	UNDER 3	±.003	UNDER 6	±.02
	STRAIGHTNESS	3-12	±.005	6-18	±.03
	ANGULARITY	OVER 12	±.010	OVER 18	±.06
	PERPENDICULARITY	X	±.1	ANGLES ±0.5°	
	PARALLELISM	ALL UNSPECIFIED TOOL RADII .015 - .030			
	CONCENTRICITY	BREAK ALL SHARP CORNERS .015 - .030			
	TRUE POSITION	ALL UNSPECIFIED MACHINED SURFACES SHALL BE 125 OR BETTER			

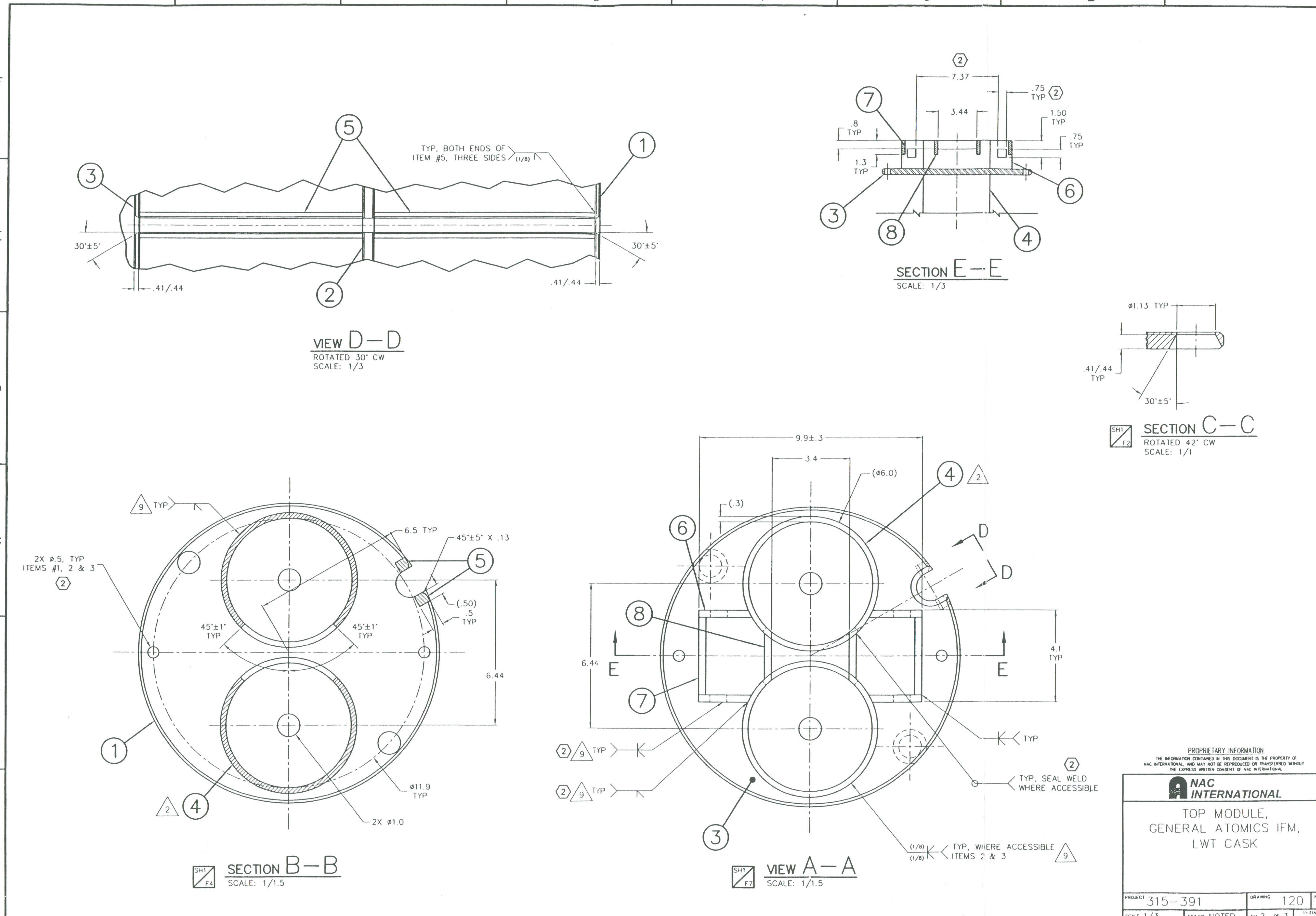
GROUP	NAME	DATE
PREPARER	R. Walker	4-29-03
CHECKER	[signature]	4-29-03
PROJECT MANAGER	[signature] for RVS	4/2/03
DIRECTOR DESIGN AND ANALYSIS	[signature]	4-30-03
DIRECTOR LICENSING	J C Thompson	5/1/03
VICE PRESIDENT QUALITY	[signature]	5/8/03

FABRICATION SPECIFICATION: 61102-S-02
DESIGN SPECIFICATION: 14661-S-01
NEXT ASSEMBLY: 315-391-124
DRAWING TYPE: DESIGN

TOP MODULE, GENERAL ATOMICS IFM, LWT CASK

PROJECT 315-391 | DRAWING 120 | REV 2
SCALE 1/3 | EST WT. NOTED | SH 1 OF 3 | 11:20AM 4-29-2003

NAC DCRM 01


VIEW D—D
ROTATED 30° CW
SCALE: 1/3
TYP, BOTH ENDS OF ITEM #5, THREE SIDES
30°±5°
.41/.44
SECTION E—E
SCALE: 1/3
7.37
3.44
.75 TYP
1.50 TYP
.8 TYP
1.3 TYP
SECTION C—C
ROTATED 42° CW
SCALE: 1/1
ø1.13 TYP
.41/.44 TYP
SECTION B—B
SCALE: 1/1.5
2X ø.5, TYP ITEMS #1, 2 & 3
6.5 TYP
45°±5° X .13
(.50)
.5 TYP
45°±1° TYP
6.44
ø11.9 TYP
2X ø1.0
VIEW A—A
SCALE: 1/1.5
9.9±.3
3.4
(ø6.0)
(.3)
4.1 TYP
TYP
TYP, SEAL WELD WHERE ACCESSIBLE
TYP, WHERE ACCESSIBLE ITEMS 2 & 3
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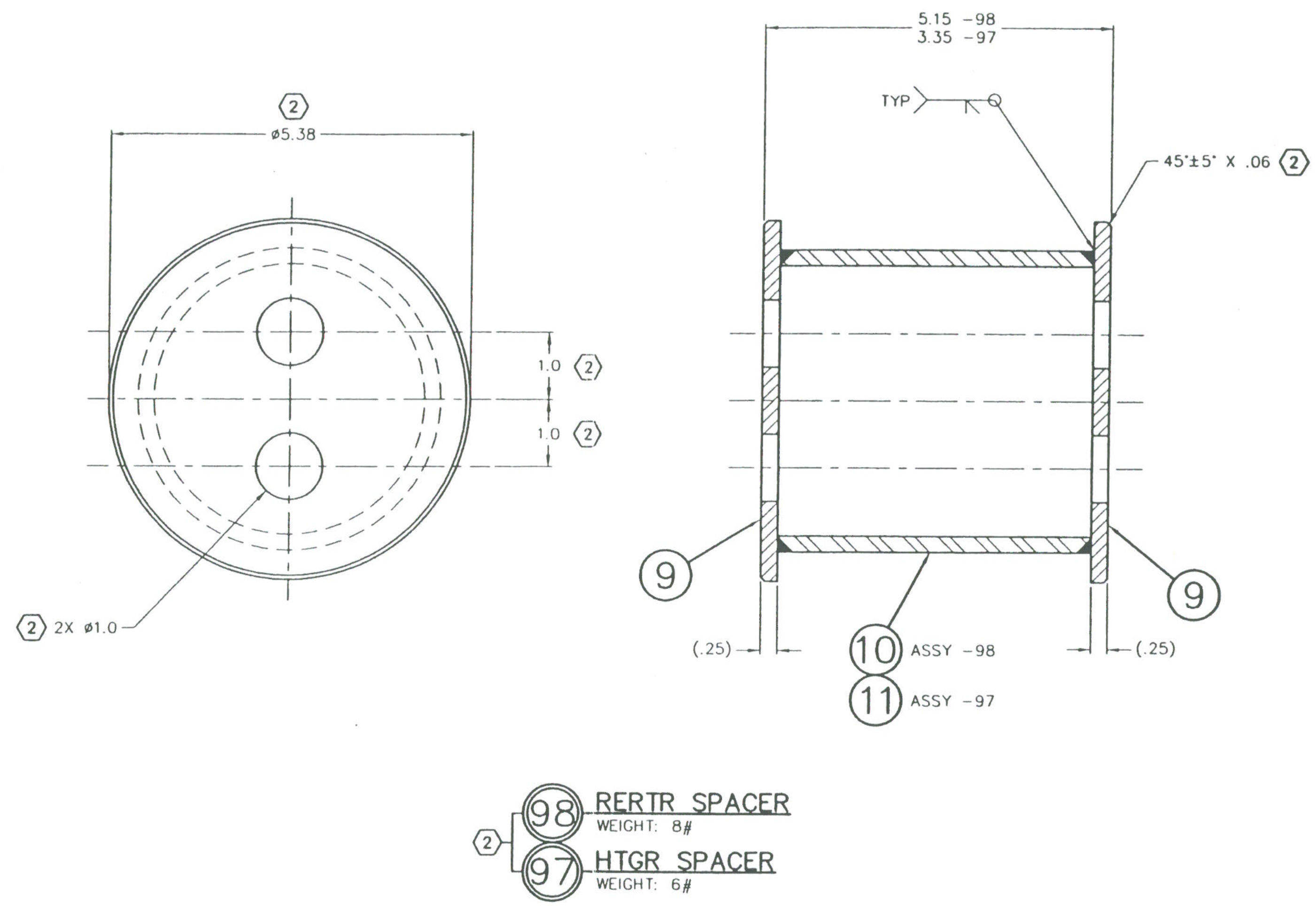
NAC INTERNATIONAL
TOP MODULE, GENERAL ATOMICS IFM, LWT CASK
PROJECT 315-391
DRAWING 120
REV 2
SCALE 1/3
EST.WT. NOTED
SH 2 OF 3



NAC INTERNATIONAL

TOP MODULE,
GENERAL ATOMICS IFM,
LWT CASK

PROJECT 315-391		DRAWING 120	REV 2
SCALE 1/1	EST.WT NOTED	SH 3 OF 3	11:21AM 4-29-2003

APPENDIX C
NAC-LWT CERTIFICATE OF COMPLIANCE AMENDMENT FOR TRANSPORT OF GA IFM

PC-000512/0

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

June 30, 2003

Mr. Thomas C. Thompson
Director, Licensing
Engineering and Design Services
NAC International
3930 East Jones Bridge Road
Norcross, Georgia 30092

SUBJECT: MODEL NO. NAC-LWT PACKAGE

Dear Mr. Thompson:

As requested by your application dated February 28, 2003, as supplemented March 19, May 29, June 23, and June 27, 2003, enclosed is Certificate of Compliance No. 9225, Revision No. 34, for the Model No. NAC-LWT package. This certificate supersedes, in its entirety, Certificate of Compliance No. 9225, Revision No. 33, dated November 14, 2002. Changes made to the enclosed certificate are indicated by vertical lines in the margin. The staff's Safety Evaluation Report is also enclosed.

The approval constitutes authority to use the package for shipment of radioactive material and for the package to be shipped in accordance with the provisions of 49 CFR 173.471. Those on the enclosed list have been registered as users of the package under the general license provisions of 10 CFR 71.12 or under the provisions of 49 CFR 173.471.

If you have any questions regarding this certificate, please contact me or Julia M. Barto of my staff at (301) 415-8500.

Sincerely,

John Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket No. 71-9225
TAC No. L23587

Enclosures:
1. CoC. No. 9225, Rev. 34
2. Safety Evaluation Report
3. Registered Users

cc w/encl:
R. Boyle, Department of Transportation
James M. Shuler, Department of Energy
RAMCERTS
Registered Users

REGISTERED USERS

Mr. John Claassen
A. J. Blotcky Reactor Facility
Department of Veterans Affairs
4101 Woolworth Ave.
Omaha, NE 68105

Mr. D. Gregg Simmons
Carolina Power & Light Company
5413 Shearon Harris Road
Mail Zone 5
New Hill, NC 27562

Mr. Don Allison
Duane Arnold Energy Center
3277 DAEC Road
Palo, IA 52324-9646

Mr. Richard C. Lemons
Duke Power Company
P. O. Box 1006
Charlotte, NC 28201-1007

Mr. Franchone Oshinowo
Edlow International Company
1666 Connecticut Avenue
Suite 201
Washington, DC 20009

Mr. Chuck W. Bassett
General Electric Company
6705 Vallecitos Road
Sunol, CA 94586

Mr. Mark J. Burzynski
Tennessee Valley Authority
1101 Market Street
Chattanooga, TN 37402-2801

Mr. Tim Debey
U. S. Department of the Interior
U.S. Geological Survey
Box 25046 M.S. 974
Denver, CO 80225

Mr. Thomas G. Hobbs
U. S. Department of Commerce
NIST
100 Bureau Drive, Stop 3540
Gaithersburg, MD 20899-3540

Manager, Nuclear License
Virginia Electric and Power Company
Nuclear Licensing and Operations
Support
Richmond, VA 23261

Mr. James M. Shuler
Office of Safety and Engineering
EM-5/CLV1081
U.S. Department of Energy
1000 Independence Avenue, S.W.
Washington, DC 20585-2040

PC-000512/0

NRC FORM 618 (8-2000) 10 CFR 71

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE FOR RADIOACTIVE MATERIAL PACKAGES

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9225	34	71-9225	USA/9225/B(U)F-85	1	OF	18

2. PREAMBLE

a. This certificate is issued to certify that the package (packaging and contents) described in Item 5 below meets the applicable safety standards set forth in Title 10, Code of Federal Regulations, Part 71, "Packaging and Transportation of Radioactive Material."

b. This certificate does not relieve the consignor from compliance with any requirement of the regulations of the U.S. Department of Transportation or other applicable regulatory agencies, including the government of any country through or into which the package will be transported.

3. THIS CERTIFICATE IS ISSUED ON THE BASIS OF A SAFETY ANALYSIS REPORT OF THE PACKAGE DESIGN OR APPLICATION

a. ISSUED TO *(Name and Address)*

NAC International, Inc.
3930 East Jones Bridge Road
Norcross, GA 30092

b. TITLE AND IDENTIFICATION OF REPORT OR APPLICATION

Nuclear Assurance Corporation application dated January 14, 2000, as supplemented.

4. CONDITIONS

This certificate is conditional upon fulfilling the requirements of 10 CFR Part 71, as applicable, and the conditions specified below.

5.

(a) Packaging

(1) Model No.: NAC-LWT

(2) Description

The LWT is a steel-encased, lead-shielded shipping cask. The cask is designed to transport one PWR assembly, two BWR assemblies, up to 15 metallic fuel rods, up to 42 MTR and DIDO fuel assemblies and plates, up to 25 individual PWR rods, up to 25 individual high burnup PWR or BWR rods, up to 140 TRIGA fuel elements, or up to 560 TRIGA fuel cluster rods. The overall dimensions of the package, with impact limiters, are 232 inches long by 65 inches in diameter. The cask body is approximately 200 inches in length and 44 inches in diameter. The cask cavity is 178 inches long and 13.4 inches in diameter. The volume of the cavity is approximately 14.5 cubic feet.

The cask body consists of a 0.75-inch-thick stainless steel inner shell, a 5.75-inch-thick lead gamma shield, a 1.2-inch-thick stainless steel outer shell, and a neutron shield tank. The inner and outer shells are welded to a 4-inch-thick stainless steel bottom end forging. The cask bottom consists of a 3-inch-thick, 20.75-inch-diameter lead disk enclosed by a 3.5-inch-thick stainless steel plate and bottom end forging. The cask lid is 11.3-inch-thick stainless steel stepped design, secured to a 14.25-inch-thick ring forging with twelve 1-inch diameter bolts. The cask seal is a metallic O-ring. A second teflon O-ring and a test port are provided to leak test the seal. Other penetrations in the cask cavity include the fill and drain ports, which are sealed with port covers and O-rings.

The neutron shield tank consists of a 0.24-inch-thick stainless steel shell with 0.50-inch-thick end plates. The neutron shield region is 164-inches long and 5-inches thick. The neutron shield tank contains an ethylene glycol/water solution that is 1% boron by weight.

1. a. CERTIFICATE NUMBER	b. REVISION NUMBER	c. DOCKET NUMBER	d. PACKAGE IDENTIFICATION NUMBER	PAGE		PAGES
9225	34	71-9225	USA/9225/B(U)F-85	2	OF	18

5.(a)(2) Description (continued)

The cask is equipped with aluminum honeycomb impact limiters. The top impact limiter has an outside diameter of 65.25 inches and a maximum thickness of 27.8 inches. The bottom impact limiter has an outside diameter of 60.25 inches and maximum thickness of 28.3 inches. Both impact limiters extend 12 inches along the side of the cask body.

The maximum weight of the package is 52,000 pounds and the maximum weight of the contents and basket is 4,000 pounds.

(3) Drawings

(i) The packaging is constructed in accordance with the following Nuclear Assurance Corporation Drawings:

LWT 315-40-01, Rev. 4	Cask Assembly
LWT 315-40-02, Rev. 14	Body Assembly
LWT 315-40-03, Rev. 16 (Sheets 1-6)*	Transport Cask Body
LWT 315-40-04, Rev. 10	Cask Lid Assembly
LWT 315-40-05, Rev. 9	Upper Impact Limiter
LWT 315-40-06, Rev. 9	Lower Impact Limiter
LWT 315-40-08, Rev. 14 (Sheets 1-4)	Cask Parts Detail

* Packaging Unit Nos. 1, 2, 3, 4, and 5 are constructed in accordance with Drawing No. LWT 315-40-03, Rev. 6 (Sheets 1-6).

(ii) The fuel assembly baskets are constructed in accordance with the following Nuclear Assurance Corporation and NAC International Drawings:

LWT 315-40-09, Rev. 2	PWR Basket Spacer
LWT 315-40-10, Rev. 4	PWR Basket
LWT 315-40-11, Rev. 2	BWR Basket Assembly
LWT 315-40-12, Rev. 3	Metal Fuel Basket Assembly
LWT 315-40-045, Rev. 4	42 MTR Element Base Module
LWT 315-40-046, Rev. 4	42 MTR Element Intermediate Module
LWT 315-40-047, Rev. 4	42 MTR Element Top Module
LWT 315-40-048, Rev. 1	42 MTR Element Cask Assembly
LWT 315-40-049, Rev. 4	28 MTR Element Base Module
LWT 315-40-050, Rev. 4	28 MTR Element Intermediate Module
LWT 315-40-051, Rev. 4	28 MTR Element Top Module
LWT 315-40-052, Rev. 1	28 MTR Element Cask Assembly
LWT 315-40-070, Rev. 3	7 Cell Basket TRIGA Base Module
LWT 315-40-071, Rev. 3	7 Cell Basket TRIGA Intermediate Module
LWT 315-40-072, Rev. 3	7 Cell Basket TRIGA Top Module
LWT 315-40-079, Rev. 1	TRIGA Fuel Cask Assembly
LWT 315-40-080, Rev. 2	7 Cell Poison Basket TRIGA Base Module

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5.(a)(3)(ii) Drawings (continued)

LWT 315-40-081, Rev. 2	7 Cell Poison Basket TRIGA Intermediate Module
LWT 315-40-082, Rev. 2	7 Cell Poison Basket TRIGA Top Module
LWT 315-40-083, Rev. 0	Spacer, LWT Cask Assembly TRIGA Fuel
LWT 315-40-084, Rev. 2	LWT Transport Cask Assy 140 TRIGA Elements
LWT 315-40-090, Rev. 2	35 MTR Element Base Module
LWT 315-40-091, Rev. 2	35 MTR Element Intermediate Module
LWT 315-40-092, Rev. 2	35 MTR Element Top Module
LWT 315-40-094, Rev. 2	35 MTR Element Cask Assembly
LWT 315-40-096, Rev. 2	Fuel Rod Insert, TRIGA Fuel
LWT 315-40-098, Rev. 1	Can Assembly, LWT Pin Shipment
LWT 315-40-099, Rev. 3 (Sheets 1-3)	Can Weldment, PWR/BWR Transport Canister
LWT 315-40-100, Rev. 1 (Sheets 1-2)	Lids, PWR/BWR Transport Canister
LWT 315-40-101, Rev. 0	4 x 4 Insert, PWR/BWR Transport Canister
LWT 315-40-102, Rev. 1	5 x 5 Insert, PWR/BWR Transport Canister
LWT 315-40-103, Rev. 0	Pin Spacer, PWR Transport Canister
LWT 315-40-104, Rev. 0	LWT Cask Assembly, PWR Transport Canister
LWT 315-40-105, Rev. 3 (Sheets 1-2)	PWR Insert, PWR/BWR Transport Canister
LWT 315-40-106, Rev. 1 (Sheets 1-3)	MTR Plate Canister, LWT Cask
LWT 315-40-108, Rev. 1 (Sheets 1-3)	7 Cell Basket, Top Module, DIDO Fuel
LWT 315-40-109, Rev. 1 (Sheets 1-3)	7 Cell Basket, Intermediate Module, DIDO Fuel
LWT 315-40-110, Rev.1 (Sheets 1-3)	7 Cell Basket, Bottom Module, DIDO Fuel
LWT 315-40-111, Rev. 0	LWT Transport Cask Assy DIDO Fuel
LWT 315-40-113, Rev. 0	Spacer, Top Module DIDO Fuel
LWT 315-40-120, Rev. 2 (Sheets 1-3)	Top Module, General Atomics IFM, LWT Cask
LWT 315-40-123, Rev. 1 (Sheets 1-2)	Spacer, General Atomics IFM, LWT Cask
LWT 315-40-124, Rev. 0	Transport Cask Assembly, General Atomics IFM, LWT Cask

5.(b) Contents

(1) Type and form of material

(i) Irradiated PWR fuel assemblies. The maximum fuel assembly weight is 1650 pounds, the maximum average burnup is 35,000 MWD/MTU, the minimum cool time is 2 years, and the maximum initial fuel pin pressure at 70°F is 565 psig. The fuel assemblies consist of uranium dioxide pellets within zircaloy or ZIRLO cladding, with the specifications listed below, and with fuel rod pitch, rod diameter, clad thickness, and pellet diameter as described in Table 1.2-5, of the application, as supplemented.

Fuel Type	No. Fuel Rods	Max. Initial Uranium Enrichment (w/o U-235)	Max. Initial Uranium Mass (MTU)	Max. Active Fuel Length (in.)
B&W 15x15	208	3.5	0.4750	144.0
B&W 17x17	264	3.5	0.4658	143.0
CE 14x14	176	3.7	0.4037	137.0
CE 16x16	236	3.7	0.4417	150.0
WE 14x14 Std	179	3.7	0.4144	145.2
WE 14x14 OFA	179	3.7	0.3612	144.0
WE 15x15	204	3.5	0.4646	144.0
WE 17x17 Std	264	3.5	0.4671	144.0
WE 17x17 OFA	264	3.5	0.4282	144.0
Ex/ANF 14x14 WE	179	3.7	0.3741	144.0
Ex/ANF 14x14 CE	176	3.7	0.3814	134.0
Ex/ANF 15x15 WE	204	3.7	0.4410	144.0
Ex/ANF 17x17 WE	264	3.5	0.4123	144.0

5.(b)(1) Type and form of material (continued)

(ii) Irradiated BWR fuel assemblies. The maximum fuel assembly weight is 750 pounds, the maximum average burnup is 30,000 MWD/MTU, the minimum cool time is 2 years, and the maximum initial fuel pin pressure at 70°F is 565 psig. The fuel assemblies consist of uranium dioxide pellets within zircaloy or ZIRLO cladding, with the specifications listed below, and with fuel rod pitch, rod diameter, clad thickness, and pellet diameter as described in Table 1.2-6, of the application, as supplemented.

Fuel Type	No. Fuel Rods	No. Water Rods	Max. Initial Uranium Enrichment (w/o U-235)	Max. Initial Uranium Mass (MTU)	Max. Active Fuel Length (in.)
GE 7x7	49	0	4.0	0.1923	146
GE 8x8-1	63	1	4.0	0.1880	146
GE 8x8-2	62	2	4.0	0.1847	150 (1)
GE 8x8-4	60	4	4.0	0.1787	150 (1,2)
GE 9x9	74	2	4.0	0.1854	150 (1,3,4)
	79	2	4.0	0.1979	150 (1,4)
Ex/ANF 7x7	49	0	4.0	0.1960	144
Ex/ANF 8x8-1	63	1	4.0	0.1764	145.2
Ex/ANF 8x8-2	62	2	4.0	0.1793	150
Ex/ANF 9x9	79	2	4.0	0.1779	150
	74	2	4.0	0.1666	150 (3)

(1) Six-inch natural uranium blankets on top and bottom.
(2) One large water hole - 3.2 cm ID, 0.1 cm thickness.
(3) Two large water holes occupying seven fuel rod locations - 2.5 cm ID, 0.07 cm thickness.
(4) Shortened active fuel length in some rods.

(iii) Irradiated PWR rods, consisting of uranium dioxide pellets within zircaloy or ZIRLO cladding. The maximum uranium enrichment is 5 weight percent U-235, the maximum active fuel length is 150 inches, and the maximum pellet diameter is 0.3765 inches. The maximum burnup is 60,000 MWD/MTU and the minimum cool time is 150 days. Up to two rods may have a maximum burnup of 65,000 MWD/MTU.

5.(b)(1) Type and form of material (continued)

(iv) Irradiated MTR fuel elements composed of U-Al, U_3O_8-Al, or U_3Si_2-Al positioned within the MTR fuel basket specified in 5.(a)(3)(ii). Loose fuel plates must meet the requirements of the MTR fuel element content tables and must be loaded into an MTR plate canister prior to shipment. The fuel elements are composed of aluminum clad plates, with initial uranium enrichment up to 94.0 weight percent U-235. The maximum burnup and the minimum cool time shall be consistent with the decay heat limits in Item 5.(b)(2)(iv) and shall be determined using the operating procedures in Section 7.1.5 of the application.

NISTR MTR fuel elements specifications are listed in Item 5.(b)(1)(iv)(a), generic MTR fuel elements are listed in Item 5.(b)(1)(iv)(b), and expanded fuel specifications applicable to LEU MTR fuel (up to 25.0 wt % ^{235}U) are listed in Item 5.(b)(1)(iv)(c).

(a) NISTR MTR Fuel Content Description

Parameter	Plate	Plate (cut in half)
Enrichment, wt % ^{235}U	≤94	
Number of fuel plates	≤17	≤34
^{235}U content per plate	≤22	≤11
Plate thickness (cm)	≥0.115	
Clad Thickness (cm)	≥0.02	
Active fuel width (cm)	≤6.6	
Active fuel height (cm)	≥54 cm	27 to 30
Maximum ^{235}U content per element (g)	≤380	

(b) Generic MTR Fuel Content Description

Parameter	Limiting Values[2]					
Enrichment, wt. % ^{235}U	≤94					
Number of fuel plates	≤23	≤19	≤23[1]	≤17	≤19	≤23
^{235}U content per plate	≤18	≤20	≤20[1]	≤21	≤21	≤16.5
Plate thickness (cm)	≥0.115	≥0.115	≥0.123[1]	≥0.115	≥.200	≥0.115
Clad Thickness (cm)	≥0.02					
Active fuel width (cm)	≤6.6	≤6.6	≤6.6	≤6.6	≤6.6	≤7.3
Active fuel height (cm)	≥56					
^{235}U content per element (g)	≤380[2]					

Notes:

1. HEU (>90 wt% ^{235}U enriched) MTR fuel having 23 plates with up to 20 g of ^{235}U per plate, with a minimum plate thickness of 0.123 cm, must have at least 2.0 cm of non-fuel material at the ends of each element. This fuel may also be loaded up to 460 g ^{235}U per plate.

2. At enrichments ≤25 wt% ^{235}U, MTR fuel elements with extended fuel characteristics may be loaded with the specifications defined in 5.(b)(1)(iv)(c)

(c) Expanded LEU MTR Fuel Content Description

<table>
<tr><th>Parameter</th><th>Base</th><th colspan="3">≤7.0 cm Active Fuel Width</th><th colspan="2">≤7.1 cm Active Fuel Width</th><th colspan="3">≤7.15 cm Active Fuel Width</th></tr>
<tr><td>Enrichment, wt. % ^{235}U</td><td>≤25</td><td colspan="3">≤25</td><td colspan="2">≤25</td><td colspan="3">≤25</td></tr>
<tr><td>Number of fuel plates</td><td>≤23</td><td colspan="3">≤23</td><td>≤17</td><td>≤23</td><td>≤22</td><td>≤23</td><td>≤23</td></tr>
<tr><td>^{235}U content per plate</td><td>≤22</td><td>≤22</td><td>≤22</td><td>≤21.5</td><td colspan="2">≤22</td><td>≤22</td><td>≤21.5</td><td>≤22</td></tr>
<tr><td>Plate thickness (cm)</td><td>≥0.115</td><td>≥0.119</td><td>≥0.115</td><td>≥0.115</td><td>≥0.115</td><td>≥0.200</td><td colspan="3">≥0.119</td></tr>
<tr><td>Clad Thickness (cm)</td><td colspan="9">≥0.02</td></tr>
<tr><td>Active fuel width (cm)</td><td>≤6.6</td><td colspan="3">≤7.0</td><td colspan="2">≤7.1</td><td colspan="3">≤7.15</td></tr>
<tr><td>Active fuel height (cm)</td><td>≥56</td><td>≥56</td><td>≥63</td><td>≥56</td><td colspan="2">≥56</td><td>≥56</td><td>≥56</td><td>≥61</td></tr>
<tr><td>^{235}U content per element (g)</td><td>≤420</td><td colspan="3">≤470</td><td colspan="2">≤470</td><td colspan="3">≤470</td></tr>
</table>

5.(b)(1) Type and form of material (continued)

(v) Metallic fuel rods containing natural enrichment uranium pellets with aluminum cladding 0.080-inches thick. The fuel pellet diameter is 1.36 inches and the maximum fuel rod length is 120.5 inches. The maximum weight of uranium per rod is 54.5 kg with a maximum average burnup of 1,600 MWD/MTU and a minimum cooling time of one year.

(vi) Irradiated TRIGA fuel elements with a 0.225" diameter zirconium rod in the center and meeting the following specifications:

	TRIGA HEU (Notes 1& 2)	TRIGA LEU (Notes 1& 2)	TRIGA LEU (Notes 1& 2)
Fuel Form	Clad U-ZrH rod	Clad U-ZrH rod	Clad U-ZrH rod
Maximum Element Weight, lbs	13.2	13.2	6.4
Maximum Element Length, in	45	45	28.4
Element Cladding	Stainless Steel	Stainless Steel	Aluminum
Clad Thickness, in	0.02	0.02	0.03
Active Fuel Length, in	15	15	14-15 (Note 4)
Element Diameter, in	1.478 max.	1.478 max.	1.47 max.
Fuel Diameter, in	1.435 max.	1.435 max.	1.41 max.
Maximum Initial U Content/Element, kilograms	0.196	0.845	0.205
Maximum Initial ^{235}U Mass, grams	137	169	41
Maximum Initial ^{235}U Enrichment, weight percent	70	20	20
Zirconium Mass, grams	2060	1886 - 2300	2300
Hydrogen to Zirconium Ratio, max.	1.6	1.7	1.0
Maximum Average Burnup, MWD/MTU	460,000 (80% ^{235}U)	151,100 (80% ^{235}U)	151,100 (80% ^{235}U)
Minimum Cooling Time	90 days (Note 3)	90 days (Note 3)	90 days (Note 3)

Notes:
1. Mixed TRIGA LEU and HEU contents authorized.
2. TRIGA Standard, instrumented and fuel follower control rod type elements authorized.
3. Maximum decay heat of any element is 7.5 watts.
4. Aluminum clad fuel with 14 inch active fuel is solid and has no central hole with a zirconium rod.

5.(b)(1) Type and form of material (continued)

(vii) Irradiated TRIGA fuel cluster rods with a maximum average burnup of 600,000 MWD/MTU (80% ^{235}U) and a minimum cooling time of 160 days meeting the following specifications prior to irradiation:

	TRIGA Fuel Cluster Rods
Fuel Form	Clad U-ZrH rod
Maximum Rod Weight, lbs	1.5
Maximum Rod Length, in	31
Rod Cladding	Incoloy 800
Minimum Clad Thickness, in	0.015
Maximum Active Fuel Length, in	22.5
Maximum Fuel Pellet Diameter, in	0.53
Maximum U Content/Rod, grams	48.6
Maximum ^{235}U Mass, grams	45.4
Maximum ^{235}U Enrichment, weight percent	93.3
Maximum Zirconium Mass, grams	421
Hydrogen to Zirconium Ratio, max.	1.6

(viii) Irradiated high burnup PWR rods, consisting of uranium dioxide pellets within zircaloy or ZIRLO cladding. The maximum uranium enrichment is 5 weight percent U-235, the maximum active fuel length is 150 inches, and the maximum pellet diameter is 0.3765 inches. The maximum burnup is 80,000 MWD/MTU and the minimum cool time is 150 days.

5.(b)(1) Type and form of Material (continued)

(ix) Irradiated high burnup BWR rods, consisting of uranium dioxide pellets within zircaloy or ZIRLO cladding. The maximum uranium enrichment is 5 weight percent U-235, the maximum active fuel length is 150 inches, and the maximum pellet diameter is 0.490 inches. The maximum burnup is 80,000 MWD/MTU and the minimum cool time is between 150 - 270 days, as specified in the table below:

BWR Fuel Type Array Size	Burnup, b (GWD/MTU)	Minimum Cool Time (days)
7 x 7	$b \leq 60$ $60 < b \leq 70$ $70 < b \leq 80$	210 240 270
8 x 8[1]	$b \leq 80$	150

Note 1: Includes rods from all larger BWR assembly arrays (e.g., 9 x 9, 10 x 10)

(x) Irradiated DIDO fuel elements composed of U-Al, U_3O_8-Al, or U_3Si_2-Al positioned within the DIDO fuel basket specified in 5.(a)(3)(ii). The fuel elements are composed of four concentric tubes of varying diameters. The fuel elements have an initial enrichment up to 94.0 weight percent U-235. The fuel elements shall have the specifications listed below:

Parameter	LEU[(1)]	MEU[(1)]	HEU[(1)]
Maximum ^{235}U content per Element	≤ 190 g	≤ 190 g	≤ 190 g
Maximum Uranium content per Element	≤ 1000 g	≤ 475.0 g	≤ 211.1g
Minimum Fuel Tube Thickness	0.130 cm	0.130 cm	0.130 cm
Minimum Clad Thickness	0.0325 cm	0.0325 cm	0.0325 cm
Maximum Outer Diameter	9.535 cm	9.535 cm	9.535 cm
Minimum Nominal Inner Diameter	6.08 cm	6.08 cm	6.08 cm
Minimum Initial Enrichment	19 wt% ^{235}U	40 wt% ^{235}U	90 wt% ^{235}U

1 The maximum burnup and minimum cool time shall be consistent with the decay heat limits in Item 5.(b)(2)(ix) and shall be determined using the operating procedures in Section 7.1.4 of the application.

5.(b)(1) Type and form of Material (continued)

(xi) General Atomics (GA) Irradiated Fuel Material (IFM) consisting of two separate types of fuel materials: (a) High Temperature Gas Cooled Reactor (HTGR); and (b) Reduced-Enrichment Research and Test Reactor (RERTR) type TRIGA fuel entities.

(a) GA HTGR IFM comprised of four forms: fuel particles (kernels), fuel particles (coatings), fuel compacts (rods), and fuel pebbles. Fuel particles (kernels) are solid, spheridized, high-temperature sintered fully-densified, ceramic kernel substrate, composed of UO_2, UCO_2, $(Th,U)C_2$, or $(Th,U)O_2$. Fuel particles (coatings) are solid, spheridized, isotropic, discrete multi-layered fuel particle coatings with chemical composition including pyrolitic-carbon (PyC) and silicon carbide (SiC). Fuel compacts (rods) are multi-coated ceramic fuel particles, bound in solid, cylindrical, injection molded, high-temperature heat-treated compacts which are composed of carbonized graphite shim, coke, and graphite powder. Fuel pebbles are multi-coated fuel particles, bound in solid, spherical injection-molded, high-temperature heat-treated pebbles composed of carbonized graphite shim, coke and graphite powder. Initial enrichment of the HTGR IFM varies from 10.0 to 93.15 wt% ^{235}U.

(b) GA RERTR IFM comprised of irradiated TRIGA fuel elements which contain three distinct mass loadings of uranium of 20, 30, and 45 wt% U. The average mass of the fuel portion of the elements is 551 g with a maximum initial enrichment of 19.7 wt% U-235.

GA IFM content description:

	GA HTGR IFM	GA RERTR IFM
Fuel material	UC_2, UCO, UO_2 $(Th,U)C_2$, $(Th,U)O_2$	U-ZrH metal alloy
Maximum fuel weight, lbs	23.52	23.73
Maximum overall length, in	n/a	29.92
Maximum active fuel length, in	n/a	22.05
Fuel rod cladding	n/a	Incoloy 800
Maximum Uranium, kg U	0.21	3.86
Maximum initial ^{235}U, wt%	93.15	19.7
Maximum Activity, Ci	483	2920

5.(b)(2) Maximum quantity of material per package

Not to exceed 4,000 pounds, including contents and fuel assembly basket.

(i) For the contents described in Item 5.(b)(1)(i): one PWR assembly positioned within the PWR fuel assembly basket. Maximum decay heat not to exceed 2.5 kilowatts per PWR assembly.

(ii) For the contents described in Item 5.(b)(1)(ii): two BWR assemblies positioned with the BWR fuel assembly basket. Maximum decay heat not to exceed 1.1 kilowatts per BWR assembly.

(iii) For PWR rods as described in Item 5.(b)(1)(iii): up to 25 intact individual rods in a Type 304 stainless steel spacer canister with a wall thickness of at least 0.12 inches positioned within the PWR or BWR basket. Maximum decay heat not to exceed 1.41 kilowatts per package.

(iv) For MTR fuel elements as described in Items 5.(b)(1)(iv):

Up to 42 fuel elements positioned within the MTR fuel assembly basket (7 fuel elements per basket module). Each of the MTR basket cell openings may contain a loose plate canister. The contents of each loose plate canister are limited to the number of fuel plates, dimensions, and masses that are equivalent to an intact MTR fuel element, as specified in Items 5.(b)(1)(iv).

(a) The maximum decay heat is not to exceed 1.26 kilowatts per package, with each MTR fuel assembly basket module not to exceed 210 watts.

(b) HEU, MEU, and LEU MTR fuel elements with decay heat not exceeding 30 watts per element may be loaded in any basket position.

(c) Mixed HEU, MEU, and LEU MTR contents, with decay heat limits as specified above, are authorized.

(d) MTR fuel elements with corrosion and/or mechanically damaged cladding are authorized, provided the total surface area of through-clad corrosion and/or mechanical damage does not exceed 2,775 cm^2 per package.

(e) For HEU-MTR fuel elements only, the center fuel element in any basket module is not to exceed 120 watts. The two exterior fuel elements vertically in-line with the center assembly for transport are not to exceed 70 watts.

(v) For the contents described in Item 5.(b)(1)(v): up to 15 intact metallic fuel rods positioned within the appropriate basket. Maximum decay heat not to exceed 0.036 kilowatts per rod. Total weight of all rods not to exceed 1,805 pounds.

5.(b)(2) Maximum quantity of material per package (continued)

(vi) For failed metallic fuel rods of the type described in Item 5.(b)(1)(v):

(a) Up to six canisters containing one defective metallic fuel rod per canister. The canisters are 2.75-inch I.D. failed fuel rod canisters as shown on Nuclear Assurance Corporation Drawing No. 340-108-D2, Rev. 10, and are placed in a six-hole liner as shown on Nuclear Assurance Corporation Drawing No. 315-040-43, Rev. 1. The maximum decay heat load for a defective metallic fuel rod is limited to 5 watts; or

(b) Up to three canisters containing either up to three defective metallic fuel rods per canister or up to 10 failed fuel filters per canister. The canisters are 4.00-inch I.D. failed fuel rod canisters as shown on Nuclear Assurance Corporation Drawing No. 340-108-D1, Rev. 10, and are placed in a three-hole basket as shown on Nuclear Assurance Corporation Drawing No. 315-40-12, Rev. 3. The weight of the filters is limited to 125 pounds per canister. For canisters containing fuel rods, the maximum decay heat load is 15 watts per canister; and for canisters containing filters, the maximum decay heat load is 5 watts per canister. The plutonium content of the filters shall not exceed 20 curies per package.

(vii) For TRIGA fuel elements as described in Item 5.(b)(1)(vi):

Maximum decay heat not to exceed 7.5 watts per TRIGA fuel element (or equivalent for failed fuel) and 1050 watts per package. TRIGA fuel elements must be positioned in either the non-poisoned TRIGA fuel basket or in the poisoned TRIGA fuel basket. Fuel may not be loaded in the center cell of the non-poisoned TRIGA fuel basket.

(a) Up to 120 fuel elements in the non-poisoned TRIGA fuel basket, and up to 140 fuel elements in the poisoned TRIGA fuel basket (4 fuel elements per basket cell).

(b) Up to 12 screened canisters in the non-poisoned TRIGA fuel basket, and up to 14 screened canisters in the poisoned TRIGA fuel basket. The screened canisters are in accordance with NAC International Drawing Nos. 315-40-074, Rev. 1, 315-40-075, Rev. 1, and 315-40-076, Rev. 1. Up to four intact TRIGA fuel elements per screened canister.

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5.(b)(2) Maximum quantity of material per package (continued)

(c) Up to 12 sealed canisters in the non-poisoned TRIGA fuel basket, and up to 14 sealed canisters in the poisoned TRIGA fuel basket. The sealed canisters are in accordance with NAC International Drawing Nos. 315-40-086, Rev. 0, 315-40-087, Rev. 3, and 315-40-088, Rev. 2. Up to a maximum equivalent of two fuel elements in the form of intact fuel, failed fuel or fuel debris per sealed canister. If the total failed fuel plutonium content of a package is greater than 20 Ci, all failed fuel containing plutonium must be enclosed in a sealed canister which is then leak tested to 3.2×10^{-7} std cm^3/sec (He) prior to shipment.

(d) Mixed intact and failed fuel contents are authorized. Base and top fuel basket modules may contain intact fuel elements, screened canisters, or sealed canisters. Intermediate fuel basket modules may contain only intact TRIGA fuel elements.

(viii) For TRIGA fuel cluster rods as described in Item 5.(b)(1)(vii):

Maximum decay heat not to exceed 1.875 watts per TRIGA fuel cluster rod (or equivalent for failed fuel) and 1050 watts per package. TRIGA fuel cluster rods must be positioned in either the non-poisoned TRIGA fuel basket or in the poisoned TRIGA fuel basket. Fuel may not be loaded in the center cell of the non-poisoned TRIGA fuel basket.

(a) Up to 480 rods in the non-poisoned TRIGA fuel basket, and up to 560 rods in the poisoned TRIGA fuel basket. TRIGA fuel cluster rods must be positioned within the fuel rod inserts as shown on NAC International Drawing No. 315-40-096, Rev. 2.

(b) Up to 12 sealed canisters in the non-poisoned TRIGA fuel basket, and up to 14 sealed canisters in the poisoned TRIGA fuel basket. The sealed canisters are in accordance with NAC International Drawing Nos. 315-40-086, Rev. 0, 315-40-087, Rev. 3, and 315-40-088, Rev. 2. Up to a maximum equivalent of six TRIGA fuel cluster rods in the form of intact fuel, failed fuel or fuel debris per sealed canister. If the total failed fuel plutonium content of a package is greater than 20 Ci, all failed fuel containing plutonium must be enclosed in a sealed canister which is then leak tested to 3.2×10^{-7} std cm^3/sec (He) prior to shipment.

(c) Mixed intact and failed fuel contents are authorized. Base and top fuel basket modules may contain intact fuel rods or sealed canisters. Intermediate fuel basket modules may contain only intact fuel rods.

(ix) For high burnup PWR rods as described in Item 5.(b)(1)(viii): up to 25 intact individual rods in the appropriate insert, placed within a sealed or free-flow canister, and positioned within the standard PWR basket. Maximum decay heat not to exceed 2.3 kilowatts per package.

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5.(b)(2) Maximum quantity of material per package (continued)

(x) For high burnup BWR rods as described in Item 5.(b)(1)(ix): up to 25 intact individual rods in the appropriate insert, placed within a sealed or free-flow canister, and positioned within the standard PWR basket. Maximum decay heat not to exceed 2.1 kilowatts per package.

(xi) For DIDO fuel as described in Item 5.(b)(1)(x)

Up to 42 DIDO fuel elements with a maximum decay heat not to exceed 25 watts per DIDO fuel element provided retention spacer is present for top basket. Maximum decay heat is 1.05 kilowatts per package. If retention spacer is not present, then maximum decay heat not to exceed 18 watts per DIDO fuel element and a total of 756 watts per package.

(xii) For GA IFM as described in Item 5.(b)(1)(xi):

(a) Mixture of fuel particles (kernels and coatings), fuel compacts (rods), and fuel pebbles, packaged in its own Fuel Handling Unit (FHU).

GA HTGR FHU consists of two redundant canisters. GA HTGR IFM is packaged inside a primary canister with welded closure, as shown in General Atomics Drawing No. 032237, Rev. B, "HTGR Primary Enclosure." The primary canister is packaged inside a secondary canister with welded closure, as shown in General Atomics Drawing No. 032231, Rev. A, "HTGR Secondary Enclosure."

GA HTGR FHU total maximum decay heat not to exceed 2.05 watts, and maximum loaded weight not to exceed 71.5 lbs.

(b) Twenty irradiated TRIGA fuel elements; 13 of the elements are intact, and the remaining 7 are sectioned. GA RERTR IFM is packaged in its own FHU.

GA RERTR FHU consists of two redundant canisters. GA RERTR IFM is packaged inside a primary canister with welded closure, as shown in General Atomics Drawing No. 032236, Rev. B, "RERTR Primary Enclosure." The GA RERTR IFM primary canister is packaged inside a secondary canister with welded closure, as shown in General Atomics Drawing No. 032230, Rev. A, "RERTR Secondary Enclosure."

GA RERTR FHU total maximum decay heat not to exceed 11 watts, and maximum loaded weight not to exceed 76.0 lbs.

5.(c) Transport Index for Criticality Control

Minimum transport index to be shown on label for nuclear criticality control:

(1) For TRIGA fuel elements, TRIGA fuel cluster rods, metallic fuel rods, MTR fuel assemblies, up to 25 PWR fuel rods, up to 25 high burnup PWR or BWR rods, and GA IFM: 0.0

(2) For PWR fuel assemblies: 100

(3) For BWR fuel assemblies: 5.0

(4) For DIDO fuel assemblies: 12.5

6. Known or suspected failed fuel assemblies (rods) or elements, and fuel with cladding defects greater than pin holes and hairline cracks are not authorized, except as described in Items 5.(b)(2)(iv)(d), 5.(b)(2)(vi), 5.(b)(2)(vii)(c), and 5.(b)(2)(viii)(b).

7. The cask must be dry (no free water) when delivered to a carrier for transport.

8. Bolt torque: The cask lids bolts must be torqued to 260 ft-lbs. The bolts used to secure the vent and drain port covers must be torqued to 100 inch-lbs.

9. Prior to each shipment, the package must be leak tested to 1×10^{-3} std cm^3/sec, except that replaced seals must be leak tested to 5.5×10^{-7} std cm^3/sec (He). Prior to first use, after third use, and at least once within the 12-month period prior to each subsequent use, the package must be leak tested to 5.5×10^{-7} std cm^3/sec (He).

10. In addition to the requirements of Subpart G of 10 CFR Part 71:

(a) The metallic O-ring seal must be replaced prior to each shipment; and

(b) Each package must meet the Acceptance Tests and Maintenance Program of Chapter 8 of the application, as supplemented; and

(c) The package shall be prepared for shipment and operated in accordance with the Operating Procedures of Chapter 7 of the application, as supplemented. If the cask is loaded under water or water is introduced into the cask cavity, the cask must be vacuum dried as described in Chapter 7 of the application. The cask cavity must be backfilled with 1.0 atm of helium when shipping PWR or BWR assemblies.

11. When shipping PWR, BWR, MTR, DIDO assemblies, TRIGA fuel elements, TRIGA fuel cluster rods, individual PWR rods, or high burnup PWR or BWR rods, and GA IFM, the neutron shield tank must be filled with a mixture of water and ethylene glycol which will not freeze or precipitate in a temperature range from -40 °F to 250 °F. The water and ethylene glycol mixture must contain at least 1% boron by weight.

12. A personnel barrier must be used when shipping PWR or BWR assemblies. Shipments of MTR, DIDO fuel assemblies, TRIGA fuel elements, TRIGA fuel cluster rods, individual PWR rods, or high burnup PWR or BWR rods must use the ISO container or a personnel barrier.

13. Packages used to ship metallic fuel rods may be shipped in a closed shipping container provided that the closed container, the cask tie-down and support system and transport vehicle (trailer) meet the applicable requirements of the Department of Transportation. When the cask is shipped in a closed shipping container, the center of gravity of the combined cask, closed shipping container and trailer must not exceed 75 inches.

14. The package authorized by this certificate is hereby approved for use under the general license provisions of 10 CFR 71.12.

15. Expiration Date: February 28, 2005.

REFERENCES

NAC International, Inc., application dated January 14, 2000.

Supplements dated: February 11 and 18; April 10 and 21; May 1, 22 and 26; June 5, 12 and 20; August 23 and 31, October 2, 6, and 16, November 14, and December 19 and 27, 2000. March 1, and 15; April 27, July 3 and 20; August 22, 2001; and September 12 and 13, 2001; February 28, April 12 and September 9, 2002; February 28, March 19, May 29, June 23, and June 27, 2003.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

John D. Monninger, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Date: June 30, 2003

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PC-000512/0

SAFETY EVALUATION REPORT
Docket No. 71-9225
Model No. NAC-LWT Package
Certificate of Compliance No. 9225
Revision No. 34

SUMMARY

By application dated February 28, 2003, as supplemented March 19, May 29, June 23, and June 27, 2003, NAC International, Inc., requested an amendment to Certificate of Compliance No. 9225, for the Model No. NAC-LWT package. NAC International, Inc., requested that the certificate be modified to include the addition of General Atomics Irradiated Fuel Material (GA IFM) contents. Based on the statements and representations in the application, the staff agrees that the changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

1.0 GENERAL INFORMATION

The applicant requested the addition of GA IFM contents. GA IFM contents consist of two separate types of fuel materials: (1) High-Temperature Gas-Cooled Reactor (HTGR); and (2) Reduced-Enrichment Research and Test Reactor (RERTR) type TRIGA fuel entities. Each type of IFM is packaged in its own unique Fuel Handling Unit (FHU).

The HTGR FHU and the RERTR FHU both consist of a stainless steel primary welded enclosure encased inside a stainless steel secondary welded enclosure. Both FHUs are filled and sealed with air at atmospheric pressure. The HTGR FHU is 39 inches long and 5.25 inches in diameter. The RERTR FHU is 37.25 inches long and 4.75 inches in diameter. Each end of the FHUs is comprised of a 0.25-inch thick plate welded to the container shell.

One HTGR FHU and one RERTR FHU can be loaded together inside the NAC-LWT packaging. The two FHUs are placed in the stainless steel "Top Module" basket, and loaded into the top of the NAC-LWT cavity. The Top Module is approximately 44 inches in length and 13 inches in diameter. Each FHU is positioned inside the Top Module basket with stainless steel spacer tubes. The remainder of the NAC-LWT cavity is filled with a stainless steel bottom spacer.

Content Description

The HTGR IFM is comprised of fuel in four forms: fuel particles (kernels), fuel particles (coatings), fuel compacts (rods), and fuel pebbles.

- Fuel particles (kernels) are solid, spheridized, high-temperature sintered fully-densified, ceramic kernel substrate, composed of UC_2, UCO, UO_2 (Th,U)C_2, (Th,U)O_2.

- Fuel particles (coatings) are solid, spheridized, isotropic, discrete multi-layered fuel particle coatings with chemical composition including pyrolitic-carbon (PyC) and silicon carbide (SiC).

- Fuel compacts (rods) are multi-coated ceramic fuel particles, bound in solid, cylindrical, injection molded, high-temperature heat-treated compacts which are composed of carbonized graphite shim, coke, and graphite powder.

- Fuel pebbles are multi-coated fuel particles, bound in solid, spherical injection-molded, high-temperature heat-treated pebbles composed of carbonized graphite shim, coke, and graphite powder.

The initial enrichment of the HTGR IFM varies from 10.0 to 93.15 wt% U-235.

The RERTR IFM is comprised of 20 irradiated TRIGA fuel elements; 13 of the elements are intact, and the remaining 7 are sectioned. The elements contain three distinct mass loadings of uranium of 20, 30, and 45 wt% U. The average mass of the fuel portion of the elements is 551 g with a maximum initial enrichment of 19.7 wt% U-235.

Drawings

The applicant provided packaging drawings, to include the GA IFM fuel contents. The new drawings for this amendment include NAC International Drawing Nos.:

315-40-120, Rev. 2, Sheets 1 - 3, Top Module, General Atomics IFM, LWT Cask
315-40-123, Rev. 1, Sheets 1 - 2, Spacer, General Atomics IFM, LWT Cask
315-40-124, Rev. 0, Transport Cask Assembly, General Atomics IFM, LWT Cask

New drawings for this amendment also include General Atomics Drawing Nos.:

032237, Rev. B, HTGR Primary Enclosure
032231, Rev. A, HTGR Secondary Enclosure
032236, Rev. B, RERTR Primary Enclosure
032230, Rev. A, RERTR Secondary Enclosure

Transport Index for Criticality Control (Criticality Safety Index)

Minimum criticality safety index for the NAC-LWT packaging containing the GA IFM	0.0

2.0 STRUCTURAL EVALUATION

The applicant introduced a new stainless steel basket assembly, which consists of a 43.7 inch-long top module and a 133 inch-long spacer assembly, to facilitate loading of GA IFM. The top module is made up of two 6-inch diameter fuel tubes and three support plates. The spacer assembly is comprised of an 8-inch Schedule 80S pipe and five support plates. Each type of IFM is packaged into its own unique canister-type stainless steel FHU before being placed in the top module. The two weld-encapsulated FHUs, which differ slightly in size from each other, are each comprised of an inner, primary enclosure and an outer, secondary enclosure. The outer enclosures are considered to be confinement boundaries for criticality control, as opposed to being pressure containment boundaries, and no structural credit is taken for the inner enclosure.

The applicant analyzed the structural performance of the package for meeting the 10 CFR Part 71 requirements for normal conditions of transport and hypothetical accident conditions. The staff notes that the cask body need not be re-analyzed because the weight of the new basket assembly and the cask internal pressure due to the GA IFM are bounded by those design basis conditions evaluated previously. Furthermore, since the outer FHU enclosures are only considered as confinement boundaries for which the design pressure limit does not apply, the staff agrees with the applicant that only inertia load effects need to be analyzed for the FHUs and the basket assembly.

Sections 2.6.12.9 and 2.7.7.11 of the application present structural analyses of the basket assembly and the FHUs under the 1-ft and the 30-ft drop tests, respectively. The analyses followed essentially the same classical hand calculations approach and structural acceptance criteria that had been acceptable to the staff in previous cask certificate amendment reviews. For the components subject to various failure modes, including bearing and weld stresses, the analysis results show that all design margins are positives. This demonstrates that the package with the GA IMF is structurally capable of meeting the requirements of 10 CFR Part 71 for normal conditions of transport and hypothetical accident conditions.

The applicant provided a general description of the new fuel contents to be shipped in the NAC-LWT in SAR Section 1.2.3.3 as well as in other sections of the SAR. The staff reviewed the information to verify that significantly adverse chemical or galvanic reaction between the packaging and the new fuel contents will not occur.

The new contents consist of HTGR and RERTR IFM, a basket and a spacer. The HTGR fuel material consists of loose uranium-oxide and uranium-carbide fuel components fully encapsulated in an all-welded, stainless steel FHU. The RERTR IFM consists of 20 irradiated Incoloy-clad TRIGA elements with 13 of which are intact and 7 of which have been previously sectioned. The RERTR IFM is also placed in a fully encapsulating, welded, stainless steel FHU. The two FHUs are placed in a stainless steel basket (Top Module) which is positioned at the top of the NAC-LWT cavity using a stainless steel spacer.

Since the fuel is encapsulated in fully welded FHUs, it is isolated from the environment (regardless of it being wet or dry), the basket, the spacer, and the shipping package. Therefore, the staff has reasonable assurance that there will be no chemical or galvanic reactions in accordance with 10 CFR 71.43(d).

3.0 THERMAL

The applicant provided supplemental information to the existing thermal analyses in order to support the requested design changes for the NAC-LWT transportation package: addition of GA HTGR IFM and GA RERTR IFM to the approved list of contents.

The review focused on the unique details of the requested amendment. The small heat load associated with the proposed GA IFM (13.05 watts), when compared to the previously approved maximum of 2.5 kW for PWR fuel assemblies, ensured that the thermal analysis would be bounded by a previously approved application. However, all specific details of the current amendment were investigated. The staff also verified that none of the proposed changes could affect the conclusions of the previously reviewed thermal analyses.

Two GA IFM FHUs are considered as contents in the NAC-LWT transportation cask. Each IFM FHU consists of stainless steel weld-encapsulated primary and secondary cylindrical enclosures, filled and sealed with air at atmospheric pressure. The two IFM FHUs are placed in the top of the NAC-LWT cavity through the use of a specially designed top module. The cask cavity is backfilled with helium or nitrogen gas to one atmosphere. Three stainless steel structural support plates help conduct heat to the transportation cask walls.

The side wall region of the transportation cask consists of concentric stainless steel shells with lead shielding between the shells. Neutron shielding is provided by a 5.0-inch thick ethylene glycol/water borated solution surrounding the outer cask shell. The neutron shield is covered by a 0.24-inch thick stainless steel shell.

Heat is released from the cask exterior to the environment through convection and radiation. An aluminum honeycomb material is used for the impact limiters, thermally insulating the ends of the transportation cask.

The HTGR IFM has a maximum decay heat of 2.05 watts, and the RERTR IFM has a maximum of 11.0 watts. Due to the small heat load, the applicant used the results from previous bounding calculations to determine maximum termperatures, as shown in the following summary table:

	Normal Conditions of Transport		Hypothetical Fire Accident	
Component	max. temperature	allowable limit	max. temperature	allowable limit
Neutron Shield	198°F [1]	350°F	*assumed lost*	-
Outer Shell	199°F [1]	800°F	480°F [1]	800°F
Gamma Shield	212°F [1]	600°F	578°F [3]	600°F
Inner Shell	214°F [1]	800°F	334°F [1]	800°F
Basket	250°F	800°F	370°F	800°F
FHU contents	326°F [2]	800°F	385°F [1]	800°F

1 Bounding values obtained from 1.26 kW MTR fuel
2 Bounding value obtained from 1.05 kW TRIGA fuel
3 Bounding value obtained from 2.5 kW PWR fuel

In addition, due to the significantly larger internal free volume, the maximum normal operating pressure (MNOP) that may result from GA IFM contents is enveloped by previous calculations and remains below the design pressure of 50 psig.

The applicant used material properties and component specifications already presented and approved in previous applications. As for the proposed new fuel materials, the cladding on the TRIGA fuel is an Inconel alloy, which withstands temperatures exceeding 1000°F, and the HTGR pellets are designed for operational exposure to core temperatures also exceeding 1000°F. Nevertheless, a conservative 800°F limit is assumed. The staff reviewed and confirmed that the maximum allowable temperatures for each safety component important for the proper function of cask containment, radiation shielding, and criticality control were specified.

The applicant derived an effective thermal resistance in order to calculate the temperature at the secondary enclosure wall. Conservative approaches included assuming air (instead of helium or nitrogen) as the cavity filler and using external boundary conditions (inner shell temperature) corresponding to a 1.26 kW heat load condition as opposed to the actual heat load of 13 watts. The resulting secondary enclosure temperature is 250°F. The fuel temperature is bounded by a previous TRIGA loading calculation which, with a heat load of 1.05 kW, resulted in the fuel cladding temperature of 326°F. Under normal conditions of transport, all of the cask materials remained well below their respective allowable temperatures.

Under normal conditions of transport and in the shade, the outside surface of the transportation package is barely above 100°F, due to the low internal heat load. A personnel barrier is, therefore, not required, and the package need not be shipped exclusive use.

The evaluation of the package under hypothetical accident conditions relies on the bounding MTR fuel calculations, presented for a previous application. The maximum basket temperature is determined by applying the steady-state temperature difference between the basket and the cask inner shell, to the inner shell temperature derived from the MTR fuel fire evaluation. This is an acceptable approach, reflecting the proportionality of the inner temperature gradients to the internal heat loads. Bounding values for the maximum temperatures of other components in the transportation package can be inferred from other fire evaluations presented in the application. During the fire scenario, all components remained well below their respective allowable temperatures, and the internal pressure remains below the design limit of 50 psig.

The confirmatory analyses performed by the staff involved verifying the outside surface temperature and the top module internal temperature distribution, under normal conditions of transport ambient temperature and insolation requirements. Analytical models, using a spreadsheet application, were developed to explore these simple 1-D bounding simulations. Results of the staff's evaluations and calculations are consistent with the applicant's results.

The staff reviewed the applicant's evaluation and agrees with the applicant's conclusion that the thermal performance of the package meets the requirements of 10 CFR Part 71.

4.0 CONTAINMENT

The applicant revised the containment analysis for the Model No. NAC-LWT to consider IFM contained in double cylindrical stainless steel weld-encapsulated canisters. The IFM consists of two types of fuel: 1) twenty irradiated TRIGA fuel rods, 7 of which may have been previously sectioned for examination purposes and placed in closed aluminum tubes, and 2) HTGR fuel as fuel particle kernels or coatings, fuel compacts, or fuel pebbles. The HTGR IFM consists of less than the 0.74 TBq (20 Ci) plutonium limit to require double containment per 10 CFR 71.63. The TRIGA IFM, although containing greater than 0.74 TBq (20 Ci) of plutonium, is exempt from the double containment requirement per §71.63(b)(2), due to its form as a metal alloy.

The combined payload of one canister of HTGR IFM and one canister of TRIGA IFM consists of 3403 Ci, the majority of which is contained in the TRIGA canister. The previously approved analysis for similar TRIGA fuel considered a payload of greater than 290,000 Ci. A containment analysis is not necessary for the IFM contents, since they are significantly less than the previously evaluated contents.

In Appendix 4.5.1, of the safety analysis report for the package, the applicant replaced "U.S. Military Specification MIL-R-8791D," with a document entitled, "SAE International Aerospace Standard AS8791." This document prescribes the physical and chemical properties of the tetrafluoroethylene O-rings. SAE AS8791 was taken directly from MIL-R-8791D, and contains only minor editorial format changes required to bring it into conformance with the publishing requirements of SAE technical standards.

The applicant has shown and the staff agrees that the package meets the containment requirements of 10 CFR 71.51 for HTGR and TRIGA IFM contained in double cylindrical stainless steel weld-encapsulated canisters.

5.0 SHIELDING

The applicant performed a shielding analysis for the inclusion of the GA IFM contents in the NAC-LWT package. The applicant determined the source terms for each IFM FHU using activity inventories from January 1, 1996. The activities used in the shielding analysis were 2920 Ci for the RERTR IFM, and 483 Ci for the HGTR IFM. Activated hardware components are included in the RERTR IFM; however, the applicant stated that hardware activation for the HTGR IFM is not significant. The activity inventory was input into ORIGEN-S to determine the gamma and neutron spectra in the SCALE 27-group neutron and 18-group gamma structures.

The gamma and neutron spectra were used in a SAS1 one-dimensional shielding analysis for each type of fuel. The applicant's SAS1 radial model used the following assumptions: homogenized fuel centered in the cask cavity, no credit for the NAC-LWT basket materials, minimum shield dimensions, lead gap, 0.94 g/cm^3 neutron shield solution density, and no boron in the neutron shield solution. In the hypothetical accident conditions SAS1 model, the applicant modeled the neutron shield as a void.

The applicant determined bounding dose rates by combining the SAS1 dose rate results for each IFM FHU. Dose rates for a combined shipment of RERTR and HGTR IFM were determined to be less than 0.48 mrem/hr on the surface of the cask and 0.019 mrem/hr at 2 meters. The transport index, which is based on the dose rate at 1 meter from the package surface during normal conditions of transport, is less than one. For hypothetical accident conditions, the dose rate was determined to be 0.10 mrem/hr at 1 meter. All dose rates were below regulatory limits as defined in 10 CFR 71.47 and 10 CFR 71.51.

Based on review of the statements and representations in the application, the staff concludes that the shielding design has been adequately described and evaluated and that the package meets the external radiation requirements of 10 CFR Part 71.

6.0 CRITICALITY

The applicant's criticality model consisted of an infinite array of infinite length fueled regions of the NAC-LWT. The applicant modeled the fuel basket containing either two TRIGA FHUs, two HTGR FHUs, or one of each type. For the undamaged TRIGA fuel analysis, the applicant considered a four by five rectangular array, and a four by four square array with an additional rod in the middle of each side of the array. The array with the square arrangement resulted in the largest pitch, and also the most reactive undamaged fuel system. For damaged TRIGA fuel, the applicant modeled all the rod material homogeneously dispersed within the inner cavity of the TRIGA FHU, neglecting any confinement provided by the aluminum tubes used during shipping. This criticality model was found to be more reactive than the undamaged TRIGA fuel model. The HTGR fuel material was modeled homogeneously dispersed within the inner cavity of the

HTGR FHU. The HTGR FHU was shown to be much less reactive than the TRIGA FHU, due to the small amount of fissile uranium contained in the HTGR material.

The applicant used the KENO V.a 3-D multi-group Monte Carlo criticality program with the 27 group ENDF/B cross-section set for all IFM analyses. Since the TRIGA fuel material proved to be the most reactive of the two contents considered in the analysis, the applicant applied the code bias and uncertainty previously calculated based on TRIGA fuel element critical benchmarks. The resulting maximum calculated k_{eff} was 0.74015, including code bias and uncertainty and two times the Monte Carlo uncertainty, for the NAC-LWT containing two fully moderated damaged TRIGA FHUs, with no water in the LWT cavity, and no external moderation.

The staff performed confirmatory criticality calculations using the CSAS25 criticality analysis sequence in the SCALE code system, along with the 44-group neutron cross section set. The staff modeled the NAC-LWT package, including its TRIGA and HTGR FHU contents, using assumptions similar to those used by the applicant. The staff used the uranium masses given in Table 6.2.9-4, and the enrichment limits specified in Section 6.2.9 of the SAR. The results of the staff's criticality analyses showed that the maximum k_{eff} was less than 0.75 under both normal and hypothetical accident conditions.

The applicant has shown and the staff agrees that the Model No. NAC-LWT continues to meet the criticality safety of 10 CFR Parts 71.55 and 71.59 for both TRIGA and HTGR IFM in a maximum of two FHUs, with fuel material meeting the uranium mass specifications of Table 6.2.9-4 of the SAR, and the enrichment limits specified in Section 6.2.9 of the SAR.

7.0 OPERATING PROCEDURES

The applicant revised the operating procedures for the wet loading of LWR fuel for when site-specific conditions require the trailer to move in conjunction with, or instead of, the overhead crane to upend the cask. In this case the operating procedures may not require the trailer brakes to be set and the wheels to be blocked against movement in either direction.

The applicant also revised the operating procedures to include revised loading procedures for the TRIGA fuel in a dry configuration. Among the revisions to the loading procedures, a specific call out for how to load the GA IFM is included. The applicant also revised the procedures to provide an option to backfill the cask cavity with either helium or nitrogen to one atmosphere. The staff reviewed this change, and determined that it was acceptable because all calculation involving TRIGA fuel conservatively assume air inside the cask cavity.

Based on the statements and representations in the application, the staff concludes that the operating procedures meet the requirements of 10 CFR Part 71 and that these procedures are adequate to assure the package will be operated in a manner consistent with its evaluation for approval.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The applicant revised the maintenance program schedule for the allowable variation of the nominal valve opening pressure on the water jacket relief valve to be ±10 psig. Staff reviewed this change, and determined that it was bounded by the previous value.

CONCLUSION

The Certificate of Compliance (CoC) has been revised to include GA IFM as contents to the NAC-LWT package, including engineering drawings for the top module and spacer, and operating procedures. The transport index for criticality control for the package with GA IFM is 0.0. The certificate was also amended to include a revised address for NAC International, Inc. Based on the statements and representations in the application, the staff agrees that the changes do not affect the ability of the package to meet the requirements of 10 CFR Part 71.

Issued with Certificate of Compliance No. 9225, Revision No. 34,
on June 30, 2003 .

APPENDIX D
EXECUTED COPY OF GA IFM SHIPPING PROCEDURE

TITLE: GA HOT CELL D&D PROJECT: IRRADIATED FUEL MATERIALS SHIPMENT		
DOC. NO. DDP-1.12	ISSUE: A	PAGE 1 of 48

ISSUE SUMMARY

Issue	Date	Prepared By:	Approved By:	Purpose of Issue/Sections Changed
A	8-29-03	V. Barbat 8/27/03	J. Greenwood 8/28/03	Initial Issue 07340 800 20000
			V. Nicolayeff 8/29/03	
			R. DeVelasco	

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HEALTH PHYSICS

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HEALTH PHYSICS

1 PURPOSE

NAC International (NAC) has been contracted to ship two (2) containers (High-Temperature, Gas-Cooled Reactor [HTGR] and Reduced Enrichment Research Test Reactor [RERTR]) of Irradiated Fuel Materials (IFM) from General Atomics (GA) located in San Diego, CA to the INTEC receipt facility located at the Idaho National Engineering and Environmental Laboratory (INEEL), Scoville, ID.

The purpose of this document is to establish a procedure for the safe transfer of the HTGR and RERTR IFM Secondary Enclosures from their current storage location to the NAC-LWT cask for transport. The critical transfer operation of the IFM Secondary Enclosures from the storage casks into the NAC Transfer Cask will be performed in the GA TRIGA Reactor Mark III dry pit in Bldg. 21. The general description of the transfer facility and equipment is given in Ref. 9.13.

2 SCOPE

This procedure outlines the minimum steps necessary to perform the transfer and packaging operations. Sections of this procedure may be performed concurrently (e.g. those steps dealing with cask and equipment setup, testing, etc.)

NOTE

Field changes to this procedure may be implemented in order to more efficiently complete particular steps. All field changes shall be approved by the Project Engineer, QA and HP representatives assigned to the task, and for items affecting NAC operations, the NAC Field Engineer. All changes shall be made to the working copy of this procedure and shall be processed In Accordance With (IAW) Ref. 9.1.

3 DEFINITIONS

3.1 Acronyms

ALARA	As Low As Reasonably Achievable
CWAS	Criticality Warning Alarm System
DPM	Disintegrations per Minute
DTS	Dry Transfer System
GA	General Atomics
H&S	Health and Safety
HEPA	High Efficiency Particulate Air
HP	GA Health Physics
HRCQ	Highway Route Controlled Quantity, Radioactive Material
HTGR	High-Temperature Gas-Cooled Reactor
HWA	Hazardous Work Authorization
IAW	In Accordance With
IH	Industrial Hygienist
IIPP	Industrial Injury Prevention Program
IFM	Irradiated Fuel Materials
INEEL	Idaho National Engineering and Environmental Laboratory
ITS	Intermediate Transfer System

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HEALTH PHYSICS

ISO	Sea-Land Crate
MCO	Mobile Crane Operator
NAC	NAC International
NAC–LWT	NAC Legal Weight Cask Truck, or Shipping Cask
PPE	Personnel Protective Equipment
OSHA	Occupational Safety and Health Administration
QA	Quality Assurance
RERTR	Reduced Enrichment Research and Test Reactor
RWP	Radiation Work Permit
TRIGA	Training, Research, Isotope, reactor General Atomics
TRF	TRIGA Reactor Facility
USDOE	United States Department of Energy
USDOT	United States Department of Transportation
USNRC	United States Nuclear Regulatory Commission

3.2 Terms

Dry Transfer System — Consists of a transfer cask with a GA IFM Basket grapple, a transfer cask carriage and a Shipping Cask adapter.

Intermediate Transfer System — Consists of a lighter transfer system that is used when the crane cannot handle the weight or size of the Dry Transfer System (as is the case in the Mark III Dry Pit. The mobile crane has limited access through the opening in the roof. The installed 5 ton crane is needed for this task). It is used in two (2) configurations. The first is with only the inner shield. The second uses both the inner and outer shield. The configuration to be used for this procedure is the one with both the inner and outer shield.

Personnel — Persons performing work under this procedure.

Operations Personnel — Persons performing work under this procedure and assigned to the Operations Group of GA.

4 SAFETY REQUIREMENTS

4.1 General

All work associated with this procedure shall be performed under the guidance of appropriate section of the GA Industrial Injury Protection Program (IIPP) as defined in this procedure (Ref. 9.2). This document also provides guidance on the applicable Occupational Safety and Health Administration (OSHA) regulations, and provides forms for those operations that require approval prior to starting work. The need for Hazardous Work Authorizations (HWAs) shall be determined by the Health & Safety (H&S) Engineer with the concurrence of the Industrial Hygienist (IH) based on the criteria of the GA-IIPP after a detailed evaluation of the individual proposed tasks prior to commencing work.

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HEALTH PHYSICS

All operations related to this procedure shall be conducted IAW all applicable rules, regulations, and established radiological and industrial safety practices as set forth in Refs. 9.3 - 9.5. Any person assigned to the IFM Shipment Project may stop the work in progress if they believe that the job parameters are being exceeded, or that safety will be jeopardized.

4.2 Requirements

4.2.1 Personnel Training/Certification

All Personnel shall be trained to the requirements of DDP-1-13 (Ref. 9.6). If there is a discrepancy in training requirements between this procedure and DDP-1-13, the training requirements of DDP-1-13 has precedence. All Personnel authorized to perform work with hazardous equipment shall be properly trained and certified in the safe handling and/or safe operating practices. Examples include safe practices regarding the training, certification, and licensing of personnel for safe operation of forklifts, cranes, and hoists.

All Operations Personnel shall also be trained in the GA Radiological Contingency Plan (Ref. 9.7).

4.2.2 Personnel Emergency Training

All Personnel shall be made aware of the TRF Emergency Procedures (Ref. 9.9) which establish the responsibilities and capabilities of coping with various emergencies at the TRF. Emphasis shall be placed on those procedures regarding the proper reporting of emergency conditions and those regarding facility evacuation routes.

4.2.3 Individual Safety

Due to the hazardous nature associated with the installed 5 ton crane and the mobile crane and working in a radiological controlled area, no Personnel shall engage in any operation while working alone in the TRF. Personnel working within the building shall routinely make others in the facility aware of their location.

4.2.4 First Aid

All Operations Personnel shall be made aware of the location of the TRF's first aid station.

4.2.5 Personal Protective Equipment (PPE)

PPE requirements for Personnel shall be controlled by the HP Manager or the cognizant HP Technician. Where H&S concerns are also present, PPE requirements shall be established jointly in the RWP/HWA. The level of protection shall be based on the type of radioactive material present. All Personnel shall be instructed in the use of PPE as appropriate for job function.

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4.3 Equipment Operation

NOTE
QA, through its Training Matrix, maintains records of all of GA's employees and subcontractors regarding operating licenses for cranes, hoists, and forklifts.

4.3.1 Cranes and Hoists

4.3.1.1 Operation is limited to authorized and trained personnel per GA-IIPP (Ref. 9.2)

4.3.1.2 Operator is responsible for checking rigging and crane.

4.3.1.3 All rigging must comply with ASME B30.9-1996, SLINGS.

4.3.1.4 The lifting of the IFM Storage Casks shall be controlled by NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and ANSI N14.6-1993, "American National Standard for Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 lbs or More".

4.3.1.5 While in operation, all personnel not directly required in the area shall keep a safe distance from the equipment.

4.3.1.6 It shall be the responsibility of the rigger to assure that the rigging being used is appropriate for the lift being executed; i.e. is the sling strong enough for the lift, etc.

4.3.2 Forklift Operation

Forklifts shall only be operated under the following conditions:

4.3.2.1 Forklift operation shall be limited to authorized personnel specifically trained in its operation per GA-IIPP (Ref. 9.2).

4.3.2.2 The operator is responsible for checking the safety of the equipment and shall use the safety devices provided with the equipment, including seat belts.

4.3.2.3 While in operation, all personnel not directly required in the area shall keep a safe distance from the equipment.

4.3.2.4 Personnel directly involved in an activity shall avoid moving into the path of operating equipment or any portion thereof. Areas that are blind spots from the operator's field of vision shall be avoided at all times.

4.3.2.5 Additional riders shall not be allowed on equipment unless it is specifically designed for that purpose.

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4.3.2.6 High voltage line clearance shall be maintained IAW Article 37, of OSHA Construction Safety Orders.

4.3.3 Working Elevated

GA-IIPP (Ref. 9.2) covers working elevated tasks. Highlights covered are: Great care must be taken when working in an elevated (greater that 30 in. above floor) position. Safety restraints shall be worn whenever there is a risk of falling. Ladders and scaffolds shall be used IAW the following general rules and CAL OSHA 8 CCR 1629, "Stairways and Ladders". Safety belts or harnesses, and lanyards shall be worn when working above 6 ft. for extended periods, especially if located in one location.

4.3.3.1 Ladders

The following rules shall be enforced for the safe use of ladders:

4.3.3.1.1 Damaged or broken ladders shall be tagged out of service or destroyed.

4.3.3.1.2 Ladders shall not be spliced together to make a longer ladder.

4.3.3.1.3 Straight ladders used for access shall extend at least 3 ft. above the landing and be tied off at the top to prevent movement.

4.3.3.1.4 The feet of a ladder must be adequately attached and non-slip.

4.3.3.1.5 Always face the ladder when using it.

4.3.3.1.6 Always have the free use of both hands for support, going up or down. Use a tool pouch to carry small items or a rope and bucket for large items.

4.3.3.1.7 The base of straight ladders shall be set back a safe distance from the vertical, approximately one fourth the working height of the ladder.

4.3.3.1.8 Manufactured ladders shall be ANSI Type I or II, heavy duty.

4.3.3.2 Personnel Lifts

The following rules shall be enforced for the safe use of personnel lifts:

4.3.3.2.1 Only persons allowed to operate personnel lifts are those trained in the safe operation of personnel lifts.

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4.3.3.2.2 Personnel lifts shall not be used where the floor is uneven or unstable (i.e. loose or uncompacted soil).

4.3.3.2.3 Safety belts or harnesses shall be used when the personnel lift is elevated above 6 ft.

4.3.3.2.4 The manufacturer's recommendations shall be followed regarding operations, maintenance, and maximum weights.

4.3.3.2.5 Standing on the guard rails is prohibited.

4.3.3.3 Bridges

4.3.3.3.1 Any bridges used to span the open Dry Pit must meet CAL-OSHA requirements.

4.4 Hazardous Area Entry

The Mark III Dry Pit is designated as a Hazardous Area Entry and therefore the following measures and the provisions of CAL OSHA 8 CCR Article 108, shall apply.

4.4.1 Employees required to enter confined or enclosed spaces shall be instructed prior to entry as to the nature of the hazards involved, the necessary precautions to be taken and the use of required emergency and protective equipment, as prescribed by the H&S manager or designated person.

4.4.2 A Hazardous Area Entry Permit shall be issued outside of the entry point prior to access into the Mark III Dry Pit (Hereinafter referred to as Dry Pit).

4.4.3 All necessary equipment as specified in the Hazardous Area Entry Permit and individuals trained to respond in case rescue is required shall be available at all times during entry into the Dry Pit.

4.4.4 Suitable PPE shall be worn as necessary.

4.4.5 No sources of ignition shall be brought near the area during tests to detect explosive or toxic gases.

4.4.6 The IH or designated person shall define where (if any) continuous oxygen monitoring with alarms shall be provided. Monitoring shall only be made with instruments designed for the purpose and in proper calibration.

4.4.7 Exhaust ventilation shall be maintained in the Dry Pit to minimize concentrations of toxic and hazardous gases and dusts and to ensure that there is a control over potential airborne radioactive particles. The exhaust tube shall extend to the bottom of the pit to assure maximum protection.

4.4.8 At least one person trained in CPR and first aid shall be immediately available.

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4.4.9 Communication shall be maintained with each person working in the Dry Pit by people outside the Dry Pit.

4.5 Personnel Decontamination

Site Personnel may be subjected to skin contamination from radioactive substances on the job site. However, this is very unlikely since most of the transfer equipment is handled remotely. Personnel decontamination shall be at the discretion of the cognizant HP technician, depending on the level of contamination.

5 REQUIREMENTS or PREREQUISITES

5.1 General

5.1.1 All movement of SNM shall be conducted with prior knowledge and consent of NMA, and under the direct supervision of the MBA Custodian for MBA S-1, or the designated Alternate Custodian.

5.1.2 The Dry Pit has been radiologically prepared for the evolution. i.e. cleared, cleaned, and prepared with appropriate contamination barrier material on the deck and lower walls.

5.1.3 All Personnel shall be briefed prior to commencement of any work. The briefing shall be documented on a GA Record of Training Form (GA 2162) per Ref. 9.6.

5.1.4 For safety and ALARA considerations, only Personnel shall be allowed in the restricted area.

5.1.5 Upon approval of this procedure, an amendment to the TRIGA Work Authorization shall be completed to include this procedure along with the others for TRIGA Operations.

5.2 Equipment and Material Requirements

The following equipment shall be provided by either GA, NAC, the Mobile Crane Vendor, or other support organizations in preparation for implementing this procedure. Equivalent substitutions may be made if appropriate and necessary.

✓ 5.2.1 GAHC-1 storage cask, containing the HTGR IFM primary/secondary enclosure with attached lifting bail. PC-000384 / FHU #032231

5.2.2 BofE 945 storage cask, containing the RERTR IFM primary/secondary enclosure assembly with attached lifting bail.

5.2.3 NAC International NAC-LWT Shipping Cask system, (USNRC Package Identification No. USA/9225/B(U)F85, with current amendment allowing for transport of the GA IFM) with associated Dry Transfer System (DTS), Intermediate Transfer System (ITS), cask base plate, internal spacer, GA IFM basket, and associated auxiliary equipment, handling fixtures, adapters, etc. All equipment shall have current annual maintenance and inspection documentation during the shipping campaign.

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5.2.4 Lifting slings necessary for redundant rigging and handling of GA IFM storage casks, with testing and certification documentation. Two (2) each slings shall have a minimum capacity of 15,000 lbs. and one (1) each shall have a minimum capacity of 30,000 lbs.

5.2.5 Subcontractor mobile crane, 100 ton minimum lift capacity with current inspection certification.

5.2.6 Two way communications (such as walkie-talkies) with two different frequencies to allow communications between the mobile crane director and the mobile crane operator and between other authorized individuals.

5.2.7 Forklift with 10,000 lb minimum lift capacity.

5.2.8 Double person bridge assembly 24 ft. long, with toe boards and guard rails, to span the Dry Pit in room 21/111.

5.2.9 CCTV cameras, with pan-tilt and zoom capabilities, and associated mounting hardware, monitor, cabling, controller, etc. One of these cameras shall monitor the room activity in 21/111 with a remote monitor outside 21/111 to allow observation of the transfer by Personnel not directly associated with the task. The second shall monitor activity inside the Dry Pit.

5.2.10 Two GA IFM grapple assemblies (Refs. 9.10 and 9.26)

5.2.11 Miscellaneous lifting hardware, e.g. 500 lb. (minimum) capability nylon slings, shackles, open hooks, tag line, etc.

5.2.12 Installed 5 ton capacity crane in room 21/111 with current inspection certification.

5.2.13 Portable HEPA-filtered exhaust fan, monitoring equipment, and associated flexible ducts.

5.2.14 Two person man-basket and associated lifting hardware.

5.2.15 Extension ladder with 30 ft. minimum extended length (or if appropriate, a caged ladder for access to the bottom of the Dry Pit).

5.2.16 Miscellaneous plastic sheeting, for use as contamination barrier in room 21/111, Dry Pit, and Bldg. 21 controlled yard (See Sec. 5.1.2).

5.2.17 Miscellaneous anti-contamination clothing, gloves, etc., as required by HP.

5.2.18 Portable diesel-powered air compressor, (90 psig, 80 cfm) as required by NAC for pneumatic operation of certain NAC-LWT Shipping Cask components.

5.2.19 1A cylinder of compressed helium gas (99% purity), for use as tracer gas for NAC-LWT pre-shipment leak check(s).

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5.2.20 Miscellaneous HP Personnel dosimetry and radiation measurement portable instrumentation in current calibration as listed below (Equivalent instrumentation may be substituted as necessary or appropriate).

5.2.20.1 Five (5) Xetex dosimeters
5.2.20.2 Two (2) Teletectors, calibrated to 100 R/hr.
5.2.20.3 RO-2 Ludlum 3 GM
5.2.20.4 RO-2 Ludlum 3
5.2.20.5 RO-2
5.2.20.6 Two (2) area monitors, (Ludlum 300)
5.2.20.7 CWAS System
5.2.20.8 2 RO-2As (50R/hr)
5.2.20.9 Ludlum 2241 w/ 133-8 probe.
5.2.20.10 CAM - Mark F
5.2.20.11 Portable Air Monitor for HEPA exhaust.
5.2.20.12 TLDs
5.2.20.13 Finger rings
5.2.20.14 BC-4

5.2.21 ✓ Two portable personnel lifts, hydraulic/electric, capable of extending at least 15 ft. for personnel access to the uprighted NAC-LWT Shipping Cask top end.

5.2.22 Miscellaneous tools as required, including torque wrenches (calibrated) to tighten the NAC-LWT lid and wrenches to remove the bolts holding the GA storage cask lids.

All other equipment for the transfer shall be provided by NAC. Measuring equipment provided and used by NAC must have current calibration documentation.

6 PERSONNEL QUALIFICATIONS

6.1 General

6.1.1 Personnel shall be knowledgeable and have been trained in the applicable requirements relating to Information Safeguards IAW 10CFR73 as applicable for individual job function.

6.1.2 Personnel involved with preparing radioactive material for transport or involved with loading, unloading, or storage of such packages incident to transportation shall have successfully completed training that meets the requirements of 49CFR 172.700.

6.1.3 All Personnel shall have as a minimum the GA HP Contractor Radiological Safety Training Course, with annual refresher, as applicable. This includes NAC International personnel, Ries Construction personnel, or other subcontractors; an exception to this requirement may be granted to the mobile crane operator, at the discretion of the cognizant HP.

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6.1.4 All personnel operating equipment under this procedure shall meet the requirements set forth in Section 4 of this procedure.

6.1.5 QA shall ensure that Personnel have been trained to (Ref. 9.6) as appropriate.

7 PROCEDURE

NOTE

Activities preceded by "**HOLD POINT**" may not be performed unless all the prerequisites have been completed and the "**HOLD POINT**" signed by a QA, HP, or NAC representative, as applicable.

NOTE

HP surveys shall be conducted in three (3) phases and shall be performed as soon as possible after arrival at the site to ensure compliance with USNRC, USDOT, USDOE or other regulations, as applicable, to establish the contamination baseline for the arriving cask. The three (3) phases are: incoming (before unloading), the cask exterior (Sections 7.1.1 and 7.2.5) and the cask and ISO interior (Section 7.2.24)

7.1 Cask and Equipment Receipt

7.1.1 Perform radiological receipt survey of the transfer equipment, Shipping Cask, and containers per Ref. 9.11.

HOLD POINT

Radiological receipt survey performed and results documented.

Results documented on: Survey MAP G. Sayre / 9-16-03

HP Signature Date

7.1.2 Perform receipt inspection to ensure that the cask, ISO, and miscellaneous equipment is free of material defects and in calibration, as appropriate.

HOLD POINT

Receipt inspection performed and documented

Results documented on: PC-54739

[signature] / 9/16/03

QA Signature Date

7.1.3 Verify all necessary equipment and material listed in Section 5.2 has been delivered and that all items requiring certification has current certification documentation.

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HEALTH PHYSICS

HOLD POINT

All necessary equipment from Section 5.2 has been delivered and items requiring certification have current certification documentation.

QA Signature / Date 7/16/03

NOTE

Unless otherwise noted, all of the following tasks in this procedure shall be performed by a NAC representative or under their direct supervision.

7.2 NAC-LWT Shipping Cask Setup

NOTE

Lifting for the following steps shall be done utilizing either the mobile crane or a forklift as appropriate.

NOTE

The NAC-LWT Shipping Cask shall be referred to as the Shipping Cask.

7.2.1 Set Base Plate (Ref. 9.12) in the designated location (Ref. 9.13) keeping Base Plate as level as possible. The orientation of Base Plate shall be as provided by an NAC Representative (herein after referred to as NAC) to assure Shipping Cask accessibility.

7.2.2 Level Base Plate using an accurate level with shims and/or leveling screws in Base Plate.

7.2.3 Verify levelness of Base Plate (NAC).

NOTE

Provide two (2) portable, personnel lifts for access to the top of the cask (approximately 14 feet above the elevation of the top of the Base Plate) and ultimately the transfer cask and Shipping Cask Adapter on three sides.

NOTE

The Shipping Cask Adapter shall be oriented so that the match mark on the Shipping Cask Adapter gate is in line with the Shipping Cask drain line.

NOTE

Ensure that all ISO lid restraints are fully retracted prior to ISO lid removal.

7.2.4 Remove lid from Shipping Cask ISO container.

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7.2.5 Perform HP survey of Shipping Cask and adjacent surfaces of ISO container per Ref. 9.11 and then perform a general wipe down of accessible Shipping Cask surfaces per Ref. 9.14.

HOLD POINT

HP Survey completed and results documented.

Results documented on: Survey MAP _______ / 9-17-03
HP Signature / Date

7.2.6 Remove top and bottom impact limiters from Shipping Cask and carefully set them aside on top of plywood or similar surface in the vertical orientation. Immobilize them with wheel chocks or wood to prevent rolling.

7.2.7 Remove Shipping Cask Tie-down Strap (Ref. 9.15).

7.2.8 Remove Cask Weather Seal (Ref. 9.16) from Shipping Cask.

7.2.9 Remove vent and drain valve port covers from Shipping Cask.

7.2.10 Carefully inspect o-ring seals in side of valve port cover.

HOLD POINT

O-Rings inspected by NAC.

Found OK X Found Damaged _____ _______ / 9/17/03
QA Signature / Date

NOTE

If the o-rings are damaged, NAC shall replace them.

7.2.11 Visually inspect valved quick disconnect nipple and replace, if necessary.

HOLD POINT

Valved quick disconnect nipple inspected ~~by NAC~~.

Found OK X Found Damaged _____ _______ / 9/17/03
QA Signature / Date

NOTE

If the nipple is damaged, NAC shall replace it.

7.2.12 Attach Lifting Yoke (Ref. 9.17) to hook of mobile crane.

7.2.13 Attach Lifting Yoke to Shipping Cask trunnions.

7.2.14 Carefully raise Shipping Cask to a vertical position on the rear Shipping Cask support and lift Shipping Cask from ISO container.

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7.2.15 Place Shipping Cask onto Base Plate.

7.2.16 Tie down Shipping Cask to Base Plate using chains and turnbuckles and tighten evenly, ensuring Shipping Cask is both level and vertical (Ref. 9.18).

NOTE

Verify the levelness of the Shipping Cask by measuring the verticality of the Shipping Cask body and the levelness of the top. Adjust the turnbuckles as necessary. If required, use stainless steel shims beneath the Shipping Cask to achieve levelness.

7.2.17 Disengage lifting yoke from Shipping Cask and place in a safe location.

CAUTION

The next step must be conducted under the supervision of HP personnel as there may be a release of radiological contamination.

7.2.18 Equalize pressure in Shipping Cask cavity to atmospheric pressure by opening vent valve in the open port.

7.2.19 Replace vent and drain valve port covers.

7.2.20 Remove closure lid bolts and attach lift slings to closure lid.

7.2.21 Remove closure lid and set it on a support that is suitable for radiological control and for maintaining the cleanliness of closure lid.

7.2.22 Visually inspect inner cavity for foreign material or damage.

7.2.23 Verify no free water exists within Shipping Cask cavity.

HOLD POINT

No Free Water in Shipping Cask Cavity:		***No Foreign Material or Damage:***	
[signature]	9/17/03	[signature]	9/17/03
QA Signature	Date	QA Signature	Date
[signature]	9/17/03	[signature]	9/17/03
NAC Signature	Date	NAC Signature	Date

7.2.24 Perform HP receipt survey of interior of Shipping Cask to determine amount of fixed and removable contamination.

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HOLD POINT

HP survey performed and results documented.

Results documented on: Survey Map ___ / 9-17-03

HP Signature — Date

7.2.25 Install drain tube in Shipping Cask

7.2.26 Inspect straightness of drain line assembly (NAC).

7.2.27 Install GA IFM Basket Spacer (Ref. 9.44) into Shipping Cask cavity. Spacer was installed a fwd liner at Blacon RID

7.2.28 Install GA IFM Basket (Ref. 9.19) into Shipping Cask cavity. I saw this operation

HOLD POINT

Verify that the spacer and basket fit properly.

___ / 9/17/03

QA Signature — Date

7.2.29 Remove Cask Adapter Shield Ring from Shipping Cask Adapter.

7.2.30 Attach lift slings to Shipping Cask Adapter (Ref. 9.20) and raise it for a detailed inspection by NAC.

7.2.31 Verify that mating surface of Shipping Cask Adapter is free from debris.

7.2.32 Install Shipping Cask Tie Downs.

7.2.33 Carefully orient and lower Shipping Cask Adapter onto Shipping Cask, while positioning Tie Downs with Shipping Cask Lifting Trunnions.

7.2.34 Attach Adapter Hold Down Lugs (Ref. 9.18) across Shipping Cask Lifting Trunnions.

7.2.35 Verify levelness of Shipping Cask Adapter as installed on Shipping Cask (NAC).

7.2.36 Open Cask Adapter Gate and verify the drain line match mark on the adapter is aligned with Shipping Cask drain line (NAC).

7.2.37 Thoroughly clean Shipping Cask lid seating-surface.

7.2.38 Install Shield Ring (Protective Collar, Ref. 9.21) into Shipping Cask lid cavity, and then remove the eye-bolts from the ring.

7.2.39 Visually verify proper alignment with drain line (NAC).

7.2.40 Remove shield plug and place it in a safe location.

7.2.41 Close Shipping Cask Adapter gate.

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7.2.42 Attach lift sling to Cask Adapter Alignment Ring (Ref. 9.22) and carefully place over the opening in Shipping Cask Adapter insuring the keys in Adapter are aligned with key-ways in the ring.

7.2.43 Attach lift rigging to lift sling.

7.2.44 Connect air supply lines to control panel and transfer cask.

7.2.45 Attach NAC supplied grapple to hoist cable and air hose.

7.2.46 Verify proper operation of hoist and NAC supplied grapple.

HOLD POINT

Verify that all steps in Section 7.2 have been completed.

9/17/03

QA Signature Date

7.3 Dry Run

7.3.1 Verify proper operation of transfer cask, cask adapter and Shipping Cask using an empty or dummy module basket.

NOTE

If proper operation is not achieved, make the necessary adjustments and re-test, then disconnect the air supply lines from the Shipping Cask.

CAUTION

When raising and lowering the basket into the Shipping Cask and transfer cask, verify the hoist cable and air hose are not twisted.

NOTE

All dry run lifts conducted in room 21/111 shall be conducted using the installed 5 ton crane.

✓ 7.3.2 Stage ITS and all necessary support items in room 21/111.

7.3.3 Place mock-up of IFM Secondary Enclosure into a mock-up of IFM Storage Cask.

✓ 7.3.4 Lower mock-up assembly onto the bottom of Dry Pit.

7.3.5 Position heavy GA IFM Grapple Assembly in a convenient location in Dry Pit for later retrieval.

✓ 7.3.6 Set-up and operationally check out CCTV camera, pan/tilt, zoom lens, and monitor system.

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7.3.7 Ensure camera system is capable of monitoring the two (2) locations of the mock-up IFM storage cask and the Inner Shield of the ITS.

✓7.3.8 Install empty GA IFM Basket into Inner Shield of ITS.

✓7.3.9 Install ITS Structural Lift Lid (Ref. 9.23) onto Inner Shield of ITS and tighten the three (3) lid bolts.

✓7.3.10 Clear a level area in center of Dry Pit floor for the transfer operation.

✓7.3.11 Place Inner Shield of ITS down in this area.

CAUTION

Maintain crane engagement of the ITS Structural Lift Lid.

✓7.3.12 Detach ITS Structural Lift Lid by remotely loosening the three (3) lid bolts using pole tools.

7.3.13 Transfer ITS Structural Lift Lid to an interim storage area outside of Dry Pit.

✓7.3.14 Install personnel bridge over Dry Pit.

7.3.15 Follow procedure in Section 7.8 as applicable to the mock-up and remove mock-up IFM Secondary Enclosure from mock-up IFM Storage Cask into GA IFM Basket using GA IFM Grapple.

7.3.16 Transfer ITS Structural Lift Lid from interim storage area to the top of Inner Shield of ITS using installed crane.

7.3.17 Observing match marks, gently lower the Structural Lift Lid into place.

7.3.18 Install ITS Structural Lift Lid and tighten the three (3) lid bolts using pole tools.

7.3.19 Confirm that Outer Shield is secured to Transfer Plate by checking the tightness of the turnbuckles and tie down chains.

7.3.20 Slowly raise Inner Shield of ITS off Dry Pit floor approximately 2-3 inches using installed crane.

7.3.21 Allow load to settle and hold (about 5 minutes).

7.3.22 Raise Inner Shield of ITS and transfer to just above the waiting Outer Shield of ITS and Transfer Plate.

7.3.23 Allow load to settle.

7.3.24 Slowly lower Inner Shield of ITS into Outer Shield of ITS.

7.3.25 Stop approximately 3 inches from being fully seated as indicated by comparing the height difference between the two shields.

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7.3.26 Manually align match marks on Inner Shield to those of Outer Shield using a separate operator.

7.3.27 Lower and fully seat Inner Shield into Outer Shield of ITS.

7.3.28 Disconnect the installed crane from Inner Shield of ITS's lift attachment.

7.3.29 Carefully transfer ITS from Dry Pit area to Shipping Cask loading area using a forklift.

NOTE
The following steps will all use the mobile crane for lifting and lowering.

7.3.30 Remove ITS Adapter Shield Ring from ITS Cask Adapter.

7.3.31 Set ITS Adapter Shield Ring aside in a safe area.

7.3.32 Attach lift slings to ITS Cask Adapter (Ref. 9.24).

7.3.33 Raise ITS Cask Adapter for a detailed inspection.

7.3.34 Open Cask Adaptor Gate.

7.3.35 Verify that mating surface of ITS Cask Adapter is free from debris.

7.3.36 Carefully orient and lower Cask Adapter onto ITS, while aligning bolt holes for attachment.

7.3.37 Install Cask Adapter hold down bolts.

7.3.38 Detach ITS Structural Lift Lid by loosening the three (3) lid bolts.

7.3.39 Attach lift sling to Cask Adapter Alignment Ring and carefully place over opening in Cask Adapter insuring the keys in Cask Adapter are aligned with key-ways in the ring.

7.3.40 Lower and release ITS in the designated transfer area.

7.3.41 Remove forklift from the area.

7.3.42 Confirm Shield Assembly Adapter Gate is open

7.3.43 Attach rigging to Structural Lid.

7.3.44 Lift Structural Lid until it passes Shield Assembly Adapter Gate then shut Gate immediately.

7.3.45 Attach lift sling to ITS Adapter Alignment Ring (Ref. 9.25) and carefully place over opening in ITS adapter insuring keys in adapter are aligned with key-ways in the ring.

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7.3.46 Place transfer cask onto the adapter noting the position of transfer cask shield gate housing gussets and the cutouts in the adapter shield ring.

7.3.47 Transfer part of transfer cask load to ITS adapter assembly.

7.3.48 Open shield gate on Dry Transfer System.

7.3.49 Open shield gate on ITS adapter.

7.3.50 Actuate and open DTS TRIGA Grapple.

7.3.51 Slowly lower and engage DTS TRIGA Grapple with GA IFM basket located in ITS.

7.3.52 Close DTS TRIGA Grapple by remote actuation.

NOTE

The next step will be apparent by the significant increase in the amount of force required to displace the cable

7.3.53 Ensure that DTS TRIGA Grapple engagement has occurred by exerting force on DTS hoist cable to verify that engagement has occurred.

7.3.54 Slowly raise DTS TRIGA Grapple with basket into DTS cask until basket clears DTS shield gate.

7.3.55 Close DTS shield gate.

7.3.56 Disconnect air supply lines from DTS TRIGA Grapple and hoist.

7.3.57 Raise transfer cask to top of Shipping Cask and align it with cask adapter.

7.3.58 Place transfer cask containing loaded GA IFM basket onto adapter noting the position of the transfer cask shield gate housing gussets and the cutouts in the adapter shield ring.

7.3.59 Transfer part of transfer cask load to Shipping Cask - Shipping Cask adapter assembly.

7.3.60 Connect air supply lines to Transfer Cask Grapple and hoist.

7.3.61 Open Cask Adapter Gate.

7.3.62 Open Transfer Cask Gate.

NOTE

As the GA IFM basket enters the transfer cask cavity the lower edge of the basket may hang up. Pulling on the hoist cable will allow the chamfered edge of the basket to enter the cavity.

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7.3.63 Carefully lower GA IFM basket from transfer cask into Shipping Cask.

NOTE
The hoist cable is match marked for the point just above the transfer cask shield gate.

7.3.64 Disengage DTS TRIGA Grapple and retract it back into transfer cask to a point just above Transfer Cask Shield Gate.

NOTE
The DTS TRIGA Grapple release from the basket can be verified by noting the reduction in cable tension.

CAUTION
Rotational control of the DTS TRIGA Grapple may not be maintained following disengagement and release of the basket in the Shipping Cask. Consequently, do not raise the Grapple to the full up position in the transfer cask as damage to the hoist cable fitting and the air hose may occur.

7.3.65 Verify DTS TRIGA Grapple is retracted to a point just above Transfer Cask Shield Gate.

7.3.66 Close Cask Adapter Gate.

7.3.67 Close Transfer Cask Gate.

7.3.68 Remove transfer cask from Shipping Cask and stage in the designated area.

7.3.69 Open Cask Adaptor Gate and visually verify that GA IFM basket is properly seated in Shipping Cask cavity.

NOTE
The dry run is now complete. A detailed debriefing shall be conducted and all comments/concerns shall be addressed prior to initiation of the hot loading evolution. Minutes of the debriefing shall be recorded by QA and attendees shall sign a GA Record of Training Form (GA 2162)

7.4 Dry Run Reset

7.4.1 If necessary, open Cask Adaptor Gate.

7.4.2 If necessary open ITS Adaptor Gate.

7.4.3 Remove GA IFM Basket from NAC-LWT Shipping Cask using appropriate rigging.

7.4.4 Remove mock IFM Secondary Enclosure from GA IFM Basket.

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7.4.5 ✓ Install GA IFM Basket in Inner Shield of ITS. ✓

7.4.6 ✓ Remove mock storage cask from Dry Pit and disposition as appropriate.

7.4.7 Remove personnel bridge.

7.4.8 Attach lift sling to ITS Adapter Ring, carefully remove it from ITS Adapter and decontaminate equipment to levels at or below radiation levels indicated on the receipt survey.

NOTE

Use the match marks to insure proper rotational orientation of the shield plug to the Cask Lid Cavity Shield Ring.

7.4.9 Attach lift sling to Cask Cavity Shield Ring Plug and position it just above Cask Adapter Gate.

7.4.10 Open Cask Adapter Gate.

7.4.11 Carefully lower Shield Plug into Shield Ring. NAC shall visually verify that it is properly seated.

7.4.12 Attach Cask Lid Cavity Shield Assembly Removal Plate to Shield Ring and Shield Plug and rig for lifting.

7.4.13 Carefully lift Shield Assembly from cask and immediately close Shipping Cask Adapter Gate.

7.5 IFM Storage Cask Placement

Crane 52000 lbs AF 9/19/03

NOTE

Refer to Ref. 9.13 to visualize the movement of the IFM Storage Casks.

7.5.1 Access room 21/112 and open the double doors on the south wall with assistance of MBA #S-1 Material Custodian,

7.5.2 Convey the two (2) IFM storage casks out of room 21/112 to the concrete driveway outside the east double door entry to room 21/11 using Hillman Rollers (or equivalent), forklift, and/or appropriate hardware.

7.5.3 Position mobile crane in the driveway adjacent to IFM storage casks.

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HEALTH PHYSICS

CAUTION

IFM Storage Cask rigging must be configured in a dual or redundant fashion such that a single component failure of the lifting arrangement will not result in uncontrolled lowering of the load IAW the requirements of NUREG-0612.

HOLD POINT

Verify that all steps up to now have been completed.

QA Signature: [signature] 9/19/03 Date

HOLD POINT

Verify that all radiological steps have been taken to allow for the lowering of the IFM Storage Casks into the Dry Pit.

HP Signature: [signature] 9-19-03 Date

7.5.4 Hoist and insert HTGR (GAHC1) IFM Storage Cask into 21/111 Dry Pit as follows:
GAHC-1 AF 9/24/03 H 58442

7.5.4.1 ✓ Attach each of the primary (15,000 lb. capacity) slings to HTGR IFM Storage Cask lifting trunnions, and engage slings with primary mobile crane hook.

7.5.4.2 ✓ Attach secondary (30,000 lb. capacity) sling around body of HTGR IFM Storage Cask and engage sling with mobile crane secondary (sister) hook.

7.5.4.3 ✓ Open roof hatch above Dry Pit in room 21/111.

7.5.4.4 ✓ Ensure that installed crane and beams are translated out of the way for HTGR IFM Storage Cask lift and insertion.

7.5.4.5 ✓ Ensure that personnel bridge used during the dry run has been removed.

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HEALTH PHYSICS

NOTE

During the HTGR IFM Storage Cask lifting evolution, the mobile crane primary and sister hooks shall be operated in tandem, with the primary hook lifting the load and the sister hook taking up the slack in the secondary sling.

NOTE

Provide two way walkie-talkie type communications between the mobile crane operator and the crane director.

WARNING

Communications with the mobile crane operator shall be restricted to only one person, the crane director, an individual certified to communicate with the crane operator.

7.5.4.6 X Hoist HTGR IFM Storage Cask approximately 6" off of ground.

7.5.4.7 X Wait for HTGR IFM Storage Cask to stabilize.

7.5.4.8 X Hoist HTGR IFM Storage Cask to roof level.

7.5.4.9 ✓ Translate load over the room 21/111 roof hatch.

7.5.4.10 ✓ Lower HTGR IFM Storage Cask down to approximately 12 in. above floor.

7.5.4.11 ✓ Instruct mobile crane operator to translate HTGR IFM Storage Cask to west side of Dry Pit, with HTGR Storage Cask centerline aligned with central east/west axis of Dry Pit.

7.5.4.12 ✓ Continue to lower HTGR IFM Storage Cask into Dry Pit deck and slacken both crane hooks.

NOTE

If personnel are to be inserted into the Dry Pit using the man-basket, translate the mobile crane boom to the far north center point of the roof hatch to allow for installed crane access over the Dry Pit.

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HEALTH PHYSICS

CAUTION

Make sure all steps as directed in Section 4.4 for Hazardous Area Entry have been completed.

7.5.4.13 ✓ Enter Dry Pit with trained rigging personnel.

7.5.4.14 ✓ Ensure HTGR IFM Storage Cask is properly aligned with east-west centerline of Dry Pit. Rehoist and adjust as necessary.

7.5.4.15 ✓ Disengage secondary and primary slings from HTGR IFM Storage Cask.

7.5.4.16 ✓ Hoist slings from Dry Pit.

7.5.4.17 ✓ Vacate Dry Pit of rigging personnel.

7.5.5 Rig, hoist, and insert RERTR (BofE 945) IFM Storage Cask into Dry Pit by repeating steps 7.5.4.1 through 7.5.4.16, but position RERTR IFM Storage Cask on **east** side of Dry Pit, again centered along east-west axis of the Dry Pit.
S/N GA 945 [initials] 7/20/03 5 8441

7.6 Shipping Cask Loading Preparation

NOTE

Ensure that the Dry Pit in Room 21/111 is clear and appropriately prepared for receipt of the GA IFM Basket and the ITS Inner Shield.

7.6.1 Orient GA IFM Basket in position as shown in Section 10.1 of this document.

HOLD POINT

Verify that the GA IFM Basket is as shown in Section 10.1.

[signature] / 9/17/03
QA Signature / Date

7.6.2 Install HTGR Spacer (Ref. 9.44) in Position 1.

HOLD POINT

Verify that the HTGR Spacer is in Position 1.

[signature] / 9/17/03
QA Signature / Date

7.6.3 Fill in appropriate portion of Section 10.1.

7.6.4 Install RERTR Spacer (Ref. 9.44) in Position 2.

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HEALTH PHYSICS

HOLD POINT

Verify that the RERTR Spacer is in Position 2.

_____________ / 9/17/03
QA Signature / Date

7.6.5 Fill in appropriate portion of Section 10.1.

7.6.6 ✓ Install empty GA IFM Basket into Inner Shield of the ITS.

7.6.7 ✓ Install ITS Structural Lift Lid and tighten the three (3) lid bolts.

7.6.8 ✓ Clear a level area in the middle of Dry Pit floor for the transfer operation.

7.6.9 ✓ Vacate Dry Pit of rigging personnel.

NOTE

Maintain crane engagement of the ITS Structural Lift Lid during the next step. 6/1102 -

7.6.10 ✓ Place Inner Shield of ITS down in this area making sure that Position 1 of GA IFM Basket faces **west**.

HOLD POINT

Verify that Position 1 of the GA IFM Basket faces **west**.

_____________ / 9/19/03
QA Signature / Date

7.6.11 ✓ Detach ITS Structural Lift Lid by remotely loosening the three (3) lid bolts using pole tools.

7.6.12 Transfer ITS Structural Lid to an appropriate interim storage area outside of Dry Pit.

7.7 IFM Storage Cask Preparation for Transfer

7.7.1 Position rigging personnel inside Dry Pit.

NOTE

The following steps are to be done utilizing the rigging personnel inside the Dry Pit.

7.7.2 Unbolt, **but do not open** closure lids of both IFM Storage Casks.

7.7.3 Set closure bolts aside in Dry Pit.

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7.7.4 Attach separate lifting slings to each of the two IFM Storage Cask closure lids rigging them to allow for remote engagement with the crane hook (i.e. individual attached tag lines extending to room 21/111 deck level).

7.7.5 Vacate Dry Pit.

7.7.6 Temporarily raise set-points for Criticality Warning Alarm Systems (CWAS) in both 21/112 and 21/108 to levels as designated in Ref. 9.32.

WTR 9/20/03   9/20/03   WTR 9/20/03

CWAS Location	Current Set-Point	Raised Set-Point	Initial that Raised Set Point Has Been Set 5R/hr
Mark F Area 1	20 mR/hr	5 ~~1~~ R/hr WTR	HP
Mark F Area 2	5 R/hr	leave as-is	HP
Mark I Area 1	20 mR/hr	5 ~~1~~ R/hr	HP
Mark I Area 2	5 R/hr	leave as-is	HP
Criticality	20 mR/hr	5 ~~1~~ R/hr	HP
21/112 Area 1	20 mR/hr	5 ~~1~~ R/hr	HP
21/112 Area 2	20 mR/hr	5 ~~1~~ R/hr	HP

HOLD POINT

CWAS set-points raised.

DUE TO HIGHER RAD LEVELS THAN EXPECTED ON LOADED ITS, CWAS LIMITS (ALARM) SET UP TO 5R/HR. J. GREENWOOD 9/20/03

ALSO, NO PERSONNEL MONITOR REQ'D. at CWAS panels 9/20/03

FIELD CHANGE APPROVED JSL 9/20/03 9/20/03 9/20/03

HP Signature: ____ Date: 9-20-03

7.8 Basket Loading

NOTE

The hoisting of items for the basket loading steps shall entail the use of the installed crane in room 21/111.

7.8.1 ✓ Position heavier GA IFM Grapple Assembly (Ref. 9.10) in a convenient location in Dry Pit for later retrieval.

7.8.2 ✓ Install personnel bridge over Dry Pit.

7.8.3 ✓ Set-up and operationally check out CCTV camera, pan/tilt, zoom lens, and monitor system.

7.8.4 ✓ Ensure that camera system is capable of monitoring the three (3) locations of the two (2) IFM storage casks and GA IFM basket.

7.8.5 Station one (1) person each at the CWAS in the Mark F Control Room and in 21/112.

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NOTE

Each of the above persons shall have a walkie talkie set at a different frequency than the ones used to communicate with the crane operator. These persons shall monitor the radiation levels as indicated on the CWAS indicators. If the radiation levels approach the 1 R/hr level, they must notify the Project Engineer immediately so that work will stop. If the level approaches the 1 R/hr level, the estimated dose rates from the IFM will be higher than calculated and a different approach to the basket loading must be assessed.

7.8.6 ✓ Engage crane with the previously installed rigging on HTGR Storage Cask closure lid.

CAUTION

Make sure all appropriate radiation monitoring equipment as directed by HP is utilized prior to lifting the lids.

NOTE

Per Ref. 9.27, the dose rate at the top of the Dry Pit is expected to be <10mR/hr with the lid removed.

7.8.7 Carefully hoist closure lid off of HTGR IFM Storage Cask and place lid on deck in room 21/111.

7.8.8 Record appropriate radiation levels.

HOLD POINT

Radiation levels as recorded are acceptable by the cognizant HP to proceed and/or all appropriate steps for worker protection have been taken. If radiation levels are above the acceptable level, additional shielding or shifting of crews may be required. Radiation levels are recorded.

Radiation levels recorded in Doc. 50 mR/hr @ Deck Level   [signature] HP Signature   9-20-03 Date

7.8.9 Direct CCTV camera to the interior cavity of open HTGR IFM Storage Cask.

7.8.10 Record HTGR Secondary Enclosure S/N. 072231

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HEALTH PHYSICS

HOLD POINT

Verify the stamped ID of the HTGR is 032231.

QA Signature: [signature] Date: 9/20/03

7.8.11 Engage heavier GA IFM Grapple Assembly with appropriate lifting sling to crane.

7.8.12 Hoist heavier GA IFM Grapple Assembly over HTGR IFM Storage Cask.

NOTE

If the lifting bail is noted to be upright in the HTGR IFM Storage Cask, skip the next two (2) steps.

7.8.13 Engage lifting bail by working from personnel bridge and utilizing separate "Dental Pick" Reach Tool.

7.8.14 ✓ Raise lifting bail to the upright position.

7.8.15 ✓ Position heavier GA IFM Grapple Assembly into HTGR IFM Storage Cask.

7.8.16 ✓ Manually adjust heavier GA IFM Grapple Assembly to expose Grapple Hook.

7.8.17 ✓ Engage heavier GA IFM Grapple Hook with HTGR IFM Secondary Closure lifting bail utilizing crane and CCTV image.

7.8.18 ✓ Manually lock heavier GA IFM Grapple Assembly hook onto HTGR IFM Secondary Enclosure lifting bail by sliding assembly outer tube down and locking in-place with ball-lock pin.

7.8.19 ✓ Slowly hoist HTGR IFM Secondary Enclosure from HTGR IFM Storage Cask enclosure, utilizing crane and CCTV image, making sure that bottom of HTGR IFM Secondary Enclosure is clear of IFM Storage Cask

7.8.20 Translate crane to Position 1 (See Section 10.1) over GA IFM Basket Cavity.

7.8.21 Lower HTGR IFM Secondary Enclosure into Position 1 of GA IFM Basket Cavity making sure it is seated on the bottom.

7.8.22 Flatten lifting bail from its full upright position so that it lays flat inside recess of the End Cap of the HTGR IFM Secondary Enclosure.

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HEALTH PHYSICS

HOLD POINT

Verify the top of the HTGR Secondary Enclosure is approximately 1 in. below the top of the GA IFM Basket tube.

_______________ / 9/20/03
QA Signature Date

7.8.23 Complete appropriate portion of Section 10.1.

7.8.24 Manually unlock heavier GA IFM Grapple Assembly by removing ball lock pin and sliding outer tube up, then lowering and rotating heavier GA IFM Grapple Hook to disengage from IFM Secondary Enclosure lifting bail.

7.8.25 Translate crane with heavier GA IFM Grapple Hook aside, place it in a safe location outside of Dry Pit and disengage it from crane hook.

7.8.26 Position lighter GA IFM Grapple Assembly (Ref. 9.26) in a convenient location in Dry Pit for later retrieval.

7.8.27 Engage crane with the previously installed rigging on RERTR IFM Storage Cask lid.

NOTE

The calculated dose rate at the personnel bridge (based on Ref. 9.27) is expected to be 100 mR/hr

NOTE

While repeating the steps below, be sure to record the data required below and follow the Hold Points below.

7.8.28 Repeat steps 7.8.6 through 7.8.25 to place RERTR IFM Secondary Enclosure in Position 2 (See Section 10.1) of transfer basket utilizing lighter GA IFM Grapple Assembly (Ref. 9.26).

HOLD POINT

Radiation levels as recorded are acceptable by the cognizant HP to proceed and/or all appropriate steps for worker protection have been taken. If radiation levels are above the acceptable level, additional shielding or shifting of crews may be required. Radiation levels are recorded.

Radiation levels recorded in Doc. 140 mR/hr

_______________ / 9-20-03
HP Signature Date

7.8.29 Record stamped ID of RERTR Secondary Enclosure. 032230

Approval Stamp

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HEALTH PHYSICS

HOLD POINT

Verify the stamped ID of the RERTR is 032230.

QA Signature / 9/20/03 Date

HOLD POINT

Verify the top of the RERTR Secondary Enclosure is approximately 1 in. below the top of the GA IFM Basket tube.

QA Signature / 9/20/03 Date

7.8.30 Complete Attachment 10.1.

7.9 Transfer of ITS From Dry Pit

NOTE

The hoisting of items for the transfer of the ITS from the Dry Pit steps shall entail the use of the installed crane in room 21/111.

7.9.1 ✓ Transfer ITS Structural Lid from interim storage area to top of Inner Shield of ITS.

7.9.2 ✓ Gently lower structural lid into place observing match marks.

7.9.3 ✓ Install ITS Structural Lift Lid and tighten the three (3) lid bolts using pole tools.

CAUTION

Verify that the outer shield is secured to the transfer plate by checking the tightness of the turnbuckles and tie down chains.

7.9.4 ✓ Slowly raise ITS off Dry Pit floor approximately 2-3 inches.

7.9.5 ✓ Allow load to settle and hold for 5 minutes.

CAUTION

The estimated dose rate for the Inner Shield of the ITS has been estimated by Ref. 9.27 to be 650 mR/hr at contact on the lid. Take proper caution to minimize exposure to workers during this transfer.

7.9.6 ✓ Raise Inner Shield of ITS and transfer to just above waiting Outer Shield of ITS and transfer plate on the deck of Dry Pit.

7.9.7 ✓ Allow load to settle.

7.9.8 ✓ Slowly lower Inner Shield into Outer Shield of ITS.

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HEALTH PHYSICS

7.9.9 Stop approximately 3 inches from being fully seated as indicated by comparing the height difference between the two shields.

7.9.10 Manually align the match marks on Inner Shield to those of Outer Shield using a separate operator.

7.9.11 Lower and fully seat Inner Shield into Outer Shield of ITS.

7.9.12 Disconnect crane from Inner Shield Lift Attachment.

7.9.13 Carefully transfer ITS from Dry Pit area to Shipping Cask loading area using a forklift.

7.10 Transfer of ITS to DTS

7.10.1 Remove ITS Adapter Shield Ring from ITS Cask Adapter.

7.10.2 Attach lift slings to ITS Cask Adapter and raise it as appropriate to allow for a detailed inspection by an NAC Representative.

7.10.3 Open ITS Cask Adaptor Gate.

7.10.4 Verify that mating surface of ITS Cask Adapter is free from debris (NAC).

7.10.5 Carefully orient and lower ITS Cask Adapter onto ITS, while aligning bolt holes for attachment.

7.10.6 Install ITS Cask Adapter hold down bolts.

7.10.7 Detach ITS Structural Lift Lid by loosening the three (3) lid bolts.

7.10.8 Attach lift sling to ITS Cask Adapter Alignment Ring and carefully place over opening in Shipping Cask adapter. (NAC shall insure the keys in the adapter are aligned with the key-ways in the ring.)

7.10.9 Lower and release ITS in the designated transfer area.

7.10.10 Remove forklift from area.

7.10.11 With Shield Assembly Adapter Gate open, attach rigging to the structural lid.

7.10.12 Lift lid until it passes gate then shut gate immediately.

7.10.13 Attach a lift sling to ITS Adapter Alignment Ring and carefully place over opening in ITS Cask Adapter. (An NAC Representative shall insure the keys in the adapter are aligned with the key-ways in the ring.)

7.10.14 Place Dry Transfer Cask onto Adapter Ring aligning the position of transfer cask shield gate housing gussets and cutouts in adapter shield ring.

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HEALTH PHYSICS

7.10.15 Transfer part of transfer cask load to ITS adapter assembly.

7.10.16 Open shield gate on Dry Transfer System.

7.10.17 Open shield gate on ITS adapter.

7.10.18 Actuate and open DTS TRIGA grapple.

7.10.19 Slowly lower and engage TRIGA grapple with GA IFM basket located in ITS.

7.10.20 Close TRIGA grapple by remote actuation.

NOTE

Verification of engagement will be apparent by the significant increase in the amount of force required to displace the cable.

7.10.21 Ensure grapple engagement has occurred by exerting force on DTS hoist cable to verify that engagement has occurred. (This is to be verified by NAC.)

7.10.22 Slowly raise grapple with basket into DTS cask until basket clears DTS shield gate.

7.10.23 Close DTS shield gate.

7.10.24 Disconnect air supply lines from transfer cask grapple and hoist.

7.11 Installation Into Shipping Cask

7.11.1 Raise transfer cask to the top of Shipping Cask and align it with cask adapter.

7.11.2 Place Transfer Cask containing loaded GA IFM basket onto adapter aligning position of Transfer Cask Shield Gate housing gussets and cutouts in Adapter Shield Ring.

7.11.3 Transfer part of Transfer Cask load to Shipping Cask - Shipping Cask Adapter Assembly.

7.11.4 Connect air supply lines to transfer cask grapple and hoist.

7.11.5 Open cask adapter gate.

7.11.6 Open transfer cask gate.

NOTE

As the GA IFM basket enters the Shipping Cask cavity the lower edge of the basket may hang up. Pulling on the hoist cable will allow the chamfered edge of the basket to enter the cavity.

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HEALTH PHYSICS

7.11.7 Carefully lower GA IFM basket from Dry Transfer Cask into Shipping Cask.

NOTE

The hoist cable is match marked for the next operation.

7.11.8 ✓ Disengage grapple and retract it back into Dry Transfer Cask to a point just above Transfer Cask Shield Gate.

NOTE

The grapple release from the basket can be verified by noting the reduction in cable tension.

CAUTION

Rotational control of the grapple may not be maintained following disengagement and release of the basket in the Shipping Cask. Consequently, do not raise the grapple to the full up position in the transfer cask as damage to the hoist cable fitting and the air hose may occur.

7.11.9 ✓ Verify grapple is retracted to a point just above transfer cask shield gate.

7.11.10 ✓ Close cask adapter gate.

7.11.11 ✓ Close transfer cask gate.

7.11.12 ✓ Remove Dry Transfer Cask from cask loading area and begin a thorough decontamination of outer surface to contamination levels at or below those indicated on the receipt inspection. This shall include the wire rope, air hose, grapple and cavity.

7.11.13 Attach lift sling to the Cask Adapter Ring.

7.11.14 ✓ Carefully remove Cask Adapter Ring from Cask Adapter and decontaminate equipment to levels at or below the radiation levels indicated on the receipt survey.

NOTE

Use the match marks to insure proper rotational orientation of the shield plug to the cask lid cavity shield ring.

7.11.15 ✓ Attach lift sling to Cask Cavity Shield Ring Plug and position it just above Cask Adapter Gate.

7.11.16 ✓ Open Cask Adapter Gate.

7.11.17 ✓ Carefully lower shield plug into shield ring and visually verify that it is properly seated.

7.11.18 Attach Cask Lid Cavity Shield Assembly Removal Plate to Shield Ring and Shield Plug and rig for lifting.

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HEALTH PHYSIC

7.11.19 ✓ Carefully lift Shield Assembly from Shipping Cask and immediately close Shipping Cask adapter gate.

NOTE

Use the match marks to insure proper rotational orientation of the cask closure lid to the shipping adapter and Shipping Cask.

7.11.20 Attach lift sling to Shipping Cask Closure Lid and position it just above Cask Adapter Gate.

NOTE

NAC shall verify the seating surface of the lid has been cleaned and inspected and the metallic o-ring has been replaced prior to replacing the lid on the cask.

HOLD POINT

Verify the above information has been completed.

[signature] 9/20/03
QA Signature Date

7.11.21 Open Shipping Cask Adapter Gate. (NAC shall verify the open position by visually checking the indicator.)

7.11.22 Carefully lower closure lid into position and visually verify that it is properly seated.

7.11.23 Attach lift sling to Shipping Cask Adapter and remove hold down lugs.

R/D 9/18/03

7.11.24 Relieve persons assigned to monitor CWAS from their duty.

[initials] 9/18/03 7.12.1A

~~7.11.25~~ Restore CWAS set-points to their original level as indicated below:

To Page 37

[initials] 9/18/03

CWAS Location	Set-Point	Initial that Set-Point Has Been Restored
Mark F Area 1	20 mR/hr	HP [initials]
Mark F Area 2	5 R/hr	HP [initials]
Mark I Area 1	20 mR/hr	HP [initials]
Mark I Area 2	5 R/hr	HP [initials]
Criticality	20 mR/hr	HP [initials]
21/112 Area 1	20 mR/hr	HP [initials]
21/112 Area 2	20 mR/hr	HP [initials]

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HEALTH PHYSICS

HOLD POINT

CWAS set-points restored to original level.

Laura Gonzalez, 9/20/03
HP Signature Date

7.11.26 Carefully remove Shipping Cask Adapter and decontaminate equipment to levels at or below the radiation levels indicated on the receipt survey before placing it into its shipping container.

NOTE

While the Shipping Cask is being prepared for shipment, all the remaining DTS and ITS equipment may be prepared for shipment as well (except the base plate and the lifting yoke).

7.11.27 Thoroughly decontaminate all tools and equipment to levels at or below the radiation levels indicated on the receipt survey.

7.11.28 Replace all equipment in designated boxes in the same configuration as when originally received after decontamination as certified by the cognizant HP.

7.11.29 Disposition all GA-supplied equipment as appropriate.

7.12 Shipping Cask Assembly and Testing

7.12.1 Install and tighten the twelve (12) each 1 in. socket head cap screws to 260±20 ft-lb using three passes in the sequence indicated on the lid.

HOLD POINT

Torque Wrench S/N HJ-240: NAC

Calibration Expiration Date: 11/13/03.

Torque Verified NECKER

Art Gudell 9/20/03
QA Signature Date

LG 9/18/03 RID 9/18/03 AG 9/18/03

7.12.1A H.P. HOLD POINT and restoration of CWAS set-points

7.12.2 Remove vent and drain port covers.

7.12.3 Attach Lifting Yoke to Shipping Cask trunnions.

7.12.4 Remove Base Plate tie-downs (chains and turnbuckles) from Shipping Cask.

7.12.5 Lift and transfer Shipping Cask to ISO container.

7.12.6 Carefully lower Shipping Cask until engagement of Shipping Cask cut-outs with rear rotation trunnions has been achieved.

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7.12.7 Gently lower Shipping Cask to rest on front tie-down saddle moving crane as required to maintain Shipping Cask engagement to the trailer and crane cables vertical.

7.12.8 Disengage lifting yoke from Shipping Cask trunnions and set it aside.

7.12.9 Install Shipping Cask tie-down.

7.12.10 Dry cavity of NAC-LWT Shipping Cask using following procedure:

NOTE

Specific operations of a vacuum pump will vary based on the make and model used by the Shipping Cask user. The general procedure is as follows:

7.12.10.1 Connect vacuum pump to "vent" valve of Shipping Cask.

7.12.10.2 Verify "drain" valve port cover is securely in place and the bolts are torqued accordingly.

7.12.10.3 Connect pump discharge to a proper receptacle, as discharge may be contaminated.

NOTE

At 70°F the vapor pressure of water is 18mm Hg.

7.12.10.4 Evacuate Shipping Cask cavity until cavity pressure reaches one half of the vapor pressure of water (corrected for the nominal Shipping Cask temperature).

7.12.10.5 Continue drawing a vacuum for 15 minutes.

7.12.10.6 Close Vent Valve.

7.12.10.7 Turn off vacuum pump.

7.12.10.8 Monitor pressure in Shipping Cask.

NOTE

If pressure in Shipping Cask increases more than one quarter of the vapor pressure of water in 10 minutes Shipping Cask drying procedure shall be repeated (See step 7.12.10.4).

7.12.11 Disconnect pump from "vent" valve.

7.12.12 Backfill Shipping Cask cavity to one atmosphere (0 psig) using helium.

7.12.13 Remove coupling from "vent" valve and proceed to next operation.

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NOTE
The Shipping Cask leak test may be performed with the Shipping Cask standing vertical or horizontally once the Shipping Cask has been placed in the transport ISO container.

7.12.14 Perform leak test per applicable sections of procedure MSLT-NAC-1 (Ref. 9.29), noting that cavity is full of helium.

HOLD POINT GPP-8-He-KF25 S/N 2721 4/18/04 Due

Verify the following:

Inspected: Accu-Flow

Model 938

Equipment: Varian S/N DBAL5001

Calibration Expiration: 4/14/04

NDE Report Attached: Dated 9/20/03

[signature] / 9/20/03
QA Signature / Date

45,351 Approx lbs

NOTE
Should any test be rejected, the cause of the rejection must be corrected (replace leaking/defective o-ring) and the pressure test repeated until satisfactory results are obtained.

7.12.15 Inspect seals (under supervision of NAC) on valve port covers; replace them as required.

7.12.16 Place port covers over vent and drain valves.

7.12.17 Install and tighten port cover bolts.

NOTE
If the port cover seals are replaced, they must be tested per the applicable sections of procedure MSLT-NAC-1.

7.12.18 Using pressure test fixture, pressurize annulus between the two port cover seals to 15 psig through the pressure test port located on the valve port cover.

7.12.19 Observe air pressure gauge for 10 minutes after closing isolation valve.

7.12.20 If no drop in air pressure is observed, the seal is acceptable.

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7.12.21 If air pressure drops, remove valve port cover and replace seals.

7.12.21.1 Reinstall cover and repeat test.

7.12.21.2 Perform test on each valve port cover.

7.12.22 Install Shipping Cask impact limiters.

7.12.23 Install tamper indicating seal on Shipping Cask.

7.12.24 Install Shipping Cask weather seal.

7.12.25 Install ISO container lid.

7.12.26 Complete health physics survey and record the vehicle's radiological compliance data.

7.12.27 Complete shipping documents using appropriate personnel and apply appropriate placards and labels.

7.13 Shipment Preparation and Initiation of Shipment Paperwork

The cognizant engineer shall:

Verify, or initiate follow-up for, the following items for the pending shipment:

9/22/03 JSL Refs. 9.37 through 9.40 data are complete, approved by INEEL and are available as are all other required data per Ref. 9.41.

9/22/03 JSL GA has a copy of the Certificate of Compliance for the Shipping Cask (Ref. 9.28), including a copy of the user's list indicating GA as a registered user (Ref. 9.35). (10CFR71.12).

9/22/03 JSL A copy of the latest revision of the NAC-LWT Safety Analysis Report is available for reference (Ref. 9.34).

9/22/03 JSL Security plan (physical protection) arrangements for the shipment have been made (Ref. 9.36).

9/22/03 JSL Specific route approval has been granted by the NRC.

9/22/03 JSL The required hazardous material transport permits are in place (DOT, State)

9/23/03 JSL Initiate the Pre-shipment checklist in Section 10.2.

9/23/03 JSL Nuclear Material Transfer Reports have been completed per Ref. 9.43.

9/23/03 JSL All shipping papers prepared per Section 7.12.27.

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HOLD POINT

Verify all items above are completed.

QA Signature ____ Date 9/23/03

7.14 Shipping and Follow-up

NOTE

The following steps shall be completed by the designated person and information related to the Project Engineer.

7.14.1 Reconfirm all documents completed (GA).

7.14.2 Release shipment (GA).

7.14.3 Notify INEEL, USNRC, and USDOE (NAC/GA).

7.14.4 Verify shipment arrived at INEEL (GA).

7.14.5 Verify shipment was unloaded at INEEL and was accepted by INEEL (GA).

7.14.6 Verify cask and equipment returned to NAC (GA).

7.14.7 Closeout all paper work and file with QA Records Control (GA).

8 RECORDS REQUIREMENTS

All loading, shipping, and material transfer documents shall be filed with the QARC organization IAW QAPD-9445, Current Issue (Ref. 9.38).

9 REFERENCES

9.1 HCD-1.1 GA Hot Cell D&D Project - Procedure Evaluation, Preparation & Control.

9.2 General Atomics Industrial Injury Protection Program, Recent Issue.

9.3 TRF Work Authorization 3252, March 6, 2003.

9.4 USNRC Materials License No. SNM-696, Docket No. 70-734, United States Nuclear Regulatory Commission, Washington. DC, (as amended)

9.5 State of California Radioactive Materials License No. 0145-80, California Department of Health Services, Sacramento, CA (as amended).

9.6 DDP-1.13 GA Hot Cell D&D Project - Training Plan.

9.7 GA Radiological Contingency Plan, Issued 2003.

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9.8 TRF-AP-01, Administrative Procedures - TRIGA Reactors Facility, Rev. 15, May 2003.

9.9 TRF-EP-01, Emergency Procedures - TRIGA Reactor Facility, Rev. 16, March 2003.

9.10 Drawing No. 61102-037, Rev. 0, NAC International, Grapple, General Atomics IFM, LWT Cask, 6/5/03.

9.11 HP Procedure # 1, Receipt, Pickup and Opening of Packages Containing Radioactive Material, Issue AG, 10/97.

9.12 Drawing No. 432-016, Base Plate, NAC International, Current Issue.

9.13 Site Assessment Report General Atomics Technologies, NAC International Document, Current Issue.

9.14 HP Procedure # 64, Procedure for Performing a Routine Wipe (Smear) and Meter Survey, Issue AD, 12/99.

9.15 Drawing No. 315-390-18, Shipping Cask Tie-Down Strap, NAC International, Current Issue.

9.16 Drawing No. 315-390-24, Cask Weather Seal, NAC International, Current Issue.

9.17 Drawing No. 315-390-21, Lifting Yoke, NAC International, Current Issue.

9.18 Drawing No. 432-000, Adapter Hold down Lugs, NAC International, Current Issue.

9.19 Drawing No. 315-391-120, Top Module, GA IFM LWT Cask, NAC International, Current Issue.

9.20 Drawing No. 432-010, Shipping Cask Adapter, NAC International, Current Issue

9.21 Drawing No. 432-019, Protective Collar, NAC International, Current Issue.

9.22 Drawing No. 432-014, Cask Adapter Alignment Ring, NAC International, Current Issue.

9.23 Drawing No. 61102-004, Assy 97, ITS Structural Lift Lid, NAC International, Current Issue.

9.24 Drawing No. 61102-006, Cask Adapter, NAC International, Current Issue.

9.25 Drawing No. 61102-010, ITS Adapter Alignment Ring, NAC International, Current Issue.

9.26 Drawing No. 14641-806, IFM Grapple NAC-STC Cask, Nac International, Current Issue.

9.27 Report, GA IFM Intermediate Transfer System (ITS) Dose Rate Estimate, 4/1/03.

9.28 Certificate of Compliance for Radioactive Material Packages, Certificate #9225, Rev. 34, Docket # 71-9225, Package ID # USA/9225/B(U)F-85.

9.29 Document MSLT-NAC-1, Helium Mass Spectrometer Leak Test Procedure NAC-LWT Cask Pre-Shipment and Annual Maintenance Testing, Rev. 8, 6/18/03.

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9.30 HP Procedure #16, Offsite Shipment or Individual Removal of Radioactive Materials, Issue AQ, 11/98.

9.31 NMA Operating Procedure NMA-4, Shipment of Radioactive Material, Issue N, 5/98.

9.32 Memo, From: Jason Yi, To: John Greenwood, Nuclear Safety Evaluation of Irradiated Fuel Materials Shipment, NS:03:JY:001.

9.33 Deleted.

9.34 Safety Analysis Report for the NAC Legal Weight Truck Cask, Rev. 34, November 2002, Docket No. 71-9225, T-88004.

9.35 Letter, Michelle DeBose to Keith Asmussen, Subject: User Designation, Dated 4/22/03, Docket No. 71-9225.

9.36 STS Doc #20030053, Security Plan Irradiated Fuel Materials, General Atomics, San Diego, CA to Idaho National Engineering & Environmental Laboratory, Scoville, ID, Feb. 28, 2003.

9.37 Document, Fuel RSD Form for HTGR Can # 032231 for FHU # 315-391-120-44-001, Current Issue.

9.38 Document, Packaging RSD Form for HTGR Can # 032231 for FHU # 315-391-120-44-001, Current Issue.

9.39 Document, Fuel RSD Form for RERTR Can #032230 for FHU # 315-391-120-44-001, Current Issue.

9.40 Document, Packaging RSD Form for RERTR Can #032230 for FHU # 315-391-120-44-001, Current Issue.

9.41 Document ID:STD-1120, Standard for Receipt of Spent Nuclear Fuel, INEEL, Rev. ID:0, 8/31/01.

9.42 Document, QAPD-9445, D&D Project and NWPF Operations, Current Issue.

9.43 NUREG/BR-0006, Instructions for Completing Nuclear Material Transfer Reports, (DOE/NRC Forms 741, 741A, and 740M), Rev. 4, 2/1/00.

9.44 Drawing No. 315-391-123, Spacer, GA IFM, LWT Cask, NAC International, Current Issue.

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10 ATTACHMENTS

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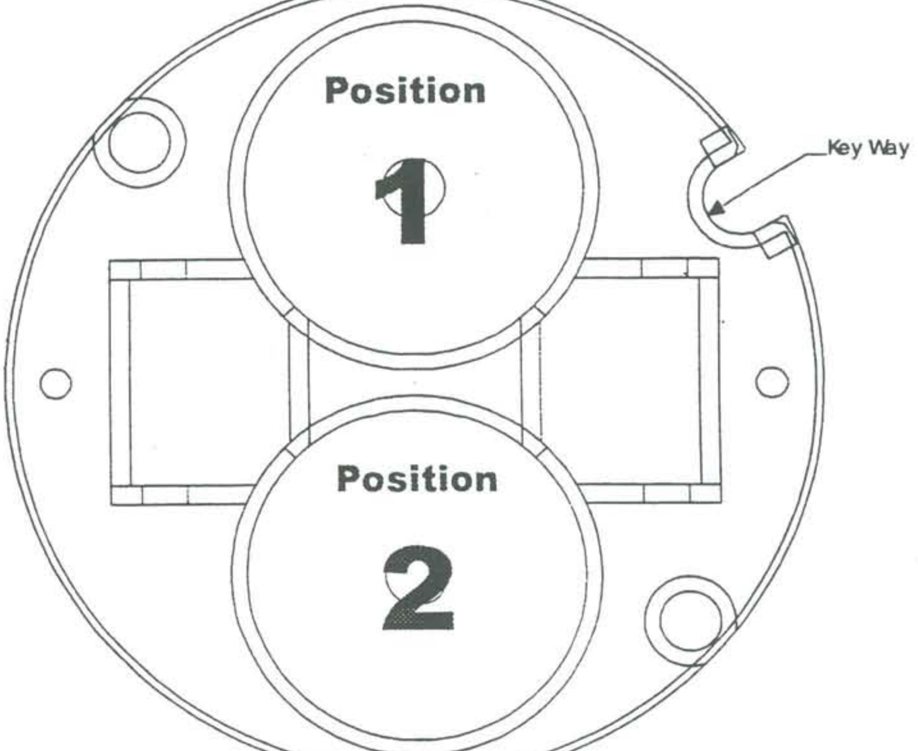
10.1 GA IFM Basket Loading Diagram

General Atomics Loading Diagram



GA IFM Basket - Top View

315-391-120-44-001 mod

Position*	Date	Time	Item	Verified	Witnessed
1	9/17/03	13:00	HTGR Spacer short	[signature]	[signature]
	9/20/03	9:00	HTGR IFM Can #0322231	[signature]	[signature]
2	9/17/03	13:00	RERTR Spacer longer	[signature]	[signature]
	9/20/03	10:00	RERTR IFM Can #032230	[signature]	[signature]

*Note position is verified by basket orientation in relation to the basket keyway

GA IFM Basket Serial Number: 315-391-120-44-001

Cask Serial Number: LWT-7

Date of Cask Loading: 9/20/03

Comments: ____________________

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10.2 Pre-Shipment Checklist

Irradiated Fuel Materials – General Atomics to INEEL

Prior to Shipment			
Initials	Date	Item	Reference Documentation
AA QA	9/23/03	Verify Fuel and Packaging RSDs for the shipment are complete and approved by INEEL. Data to include fission and activation radionuclide content, activity calculated for each radionuclide, calculation date, total activity, irradiated fissile material identification and quantities, irradiation and decay histories, and decay heat calculations. (10CFR71.91)	INEEL Acceptance Letter CCN 44887
AA QA	9/23/03	Verify General Atomics is an NRC registered user of the NAC-LWT cask, and has a copy of the Certificate of Compliance for the cask, including a copy of the user's list indicating General Atomics as a registered user. (10CFR71.12)	NRC Letter DATED 4/22/03
AA QA	9/23/03	Verify a copy of the latest revision of the NAC-LWT Safety Analysis Report is available for reference. (10CFR71.12 and 71.16)	NRC Approval Letter CoC 9225 Rev 34 6/30/03
AA QA	9/23/03	Verify cask pre-shipment testing has been completed IAW the C of C and cask use procedures, and copies of the satisfactory leak test are available.	Report Number dated 9/20/03
QA AA	9/23/03	Verify the Shipment Specific Security plan (physical protection) for the transport of irradiated nuclear fuel has been implemented, covering the requirements of 10CFR73. (10CFR73.37 and 49CFR173.22)	Report Number S/G PACKET INCLUDES SECURITY PLANS. Comm Ctr MONITORS. [signature] 23Sep03
AA QA	9/23/03	Verify the Shipment Specific Route Approval has been obtained from the US NRC. (10CFR73.37, 49CFR397, and NRC SA 01-01, 02) Cannot Copy	NRC Letter 7/25/03

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Prior to Shipment

Initials	Date	Item	Reference Documentation
QA AF	9/22/03	Verify DOT Hazardous Material Transport Permit is in place, and **HAS NOT** expired.	Permit No. C19733- CA Expiration Date: 9/28/03
AF QA	9/20/03	Verify required notifications to the applicable State's Governor's designees have been made. (10CFR73.37 and 10CFR71.97)	Correspondence #. ST20030210-CA 0211-NV, 0212-AZ UT-13, ID-14
MP QA	9/22/03	Verify the required notifications to the NRC have been prepared and submitted. (10CFR73.72 and 10CFR71.97)	Correspondence #. ST 20030208
QA AF	9/22/03	Verify the required notifications to the consignee (INEEL) have been prepared and submitted. (49CFR173.22) need	Correspondence #. ST20030209
QA	AFudel 9/22/03	Verify personnel involved with preparing radioactive material for transport or involved with loading, unloading, or storage of such packages incident to transportation shall have successfully completed training that meets the requirements of 49CFR 172.700.	Training Records and Location. See Attached 39T
QA	AFudel 9/22/03	Verify Personnel are knowledgeable and have been trained in the applicable requirements relating to Information Safeguards IAW 10CFR73. (10CFR73.21)	Training Records and Location. See Attached 39-T
QA	AFudel 9/22/03	Determine the proper shipping name and description, hazard class, special provisions, notations, and transport schedules, as applicable, from the Hazardous Material Table or Dangerous Goods List. (49CFR172.101)	Included in Shipper #. 316331

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GENERAL ATOMICS

TITLE: GA HOT CELL D&D PROJECT: IRRADIATED FUEL MATERIALS SHIPMENT		
DOC. NO. DDP-1.12	ISSUE: A	PAGE 48 of 48

AF 9/22/03
LG 9/22/03
RID 9/22/03

Prior to Shipment

Initials	Date	Item	Reference Documentation
shipping ~~HP~~ BH	9/23/03	Determine the appropriate Criticality Safety Index and Transport Index for the package, for purposes of labeling, storage or segregation, as applicable. (10CFR71.59, 49CFR 172.403)	Included in Shipper #. 316331 Index #. 0.2
HP LG	9/22/03	Obtain a copy of the receiving facility site license, if required.	License #. N/A LQ Gonzales

Highway Carrier Requirements

Initials	Date	Item	Reference Documentation
QA AF	9/22/03	Verify carrier is notified that an HRCQ and irradiated nuclear fuel will be shipped, and that the physical protection requirements of 49CFR173.22 and 10CFR73 will be implemented for the shipment.	Letter #. TELECON BLAKE WILLIAMS 15 SEP 03 w/ JOHN HAUSER TSNT
QA AF	9/23/03	Verify that the carrier, or his designated agent, has established a written route plan, IAW the requirements of 49CFR397, Subpart D, and that the carrier is aware of his responsibilities for compliance with 49CFR397 relating to the carriage of an HRCQ, including route plan contents, driver training requirements, and variations from the route plan. (49CFR397.101)	Route Plan #. 207 Driver Briefed [signature] 22 SEP 03
QA AF	9/23/03	Verify the carrier, or his designated agent, has provided a copy of the written route plan to the shipper and driver prior to the shipment. (49CFR397.101) S/G PACKET INCLUDES NRC ROUTE [signature] 22 SEP 03	(Note: This is Safeguards information. Obtain copy for QA files after the shipment is accepted by INEEL.)
AF QA	9/23/03	Verify the driver has in possession a copy of training certification containing the information required. (49CFR397.101)	Copy Attached: YES

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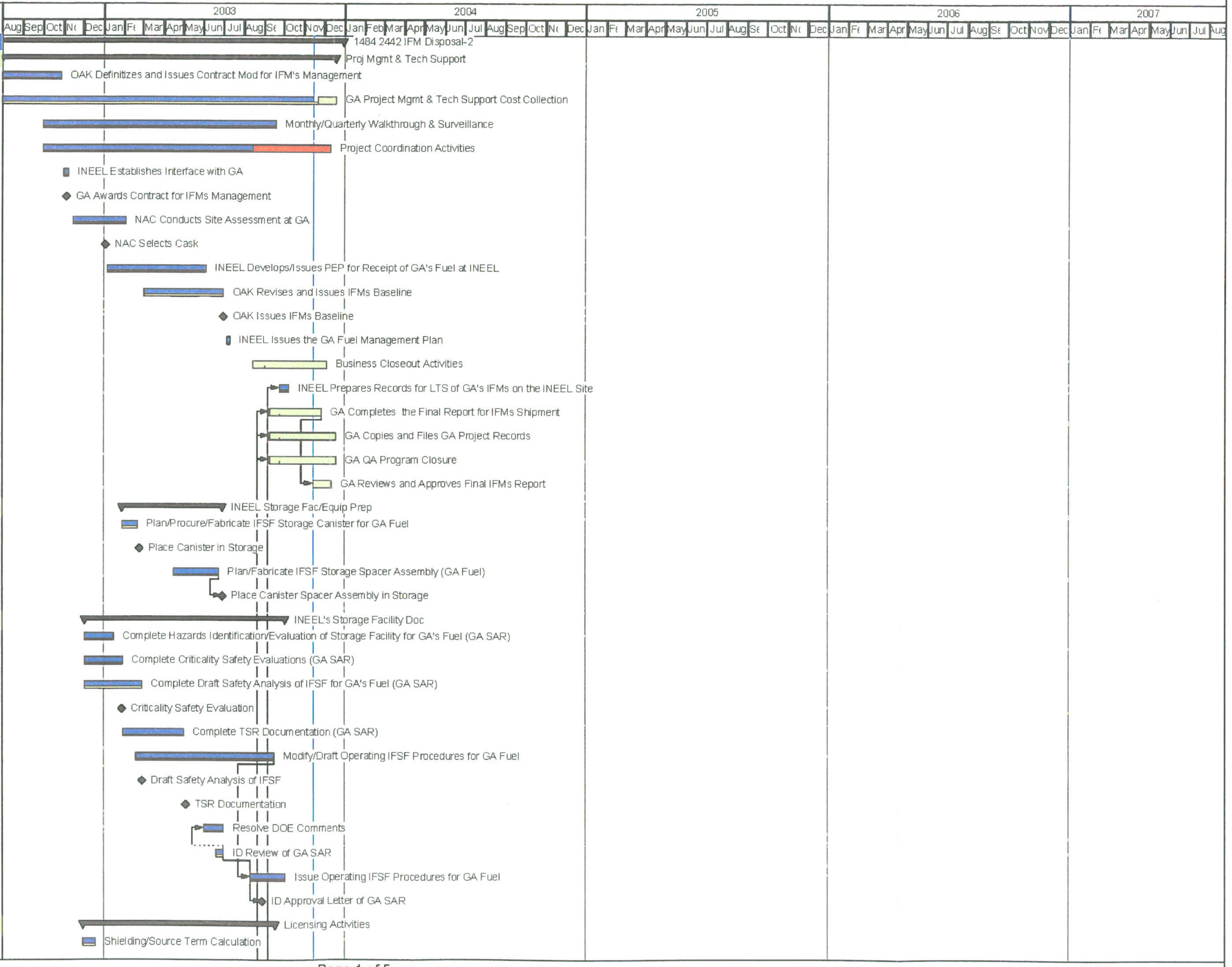
APPENDIX E
PROJECT SCHEDULE

Disposal of Hot Cell IFM

20.Nov.03 14:53

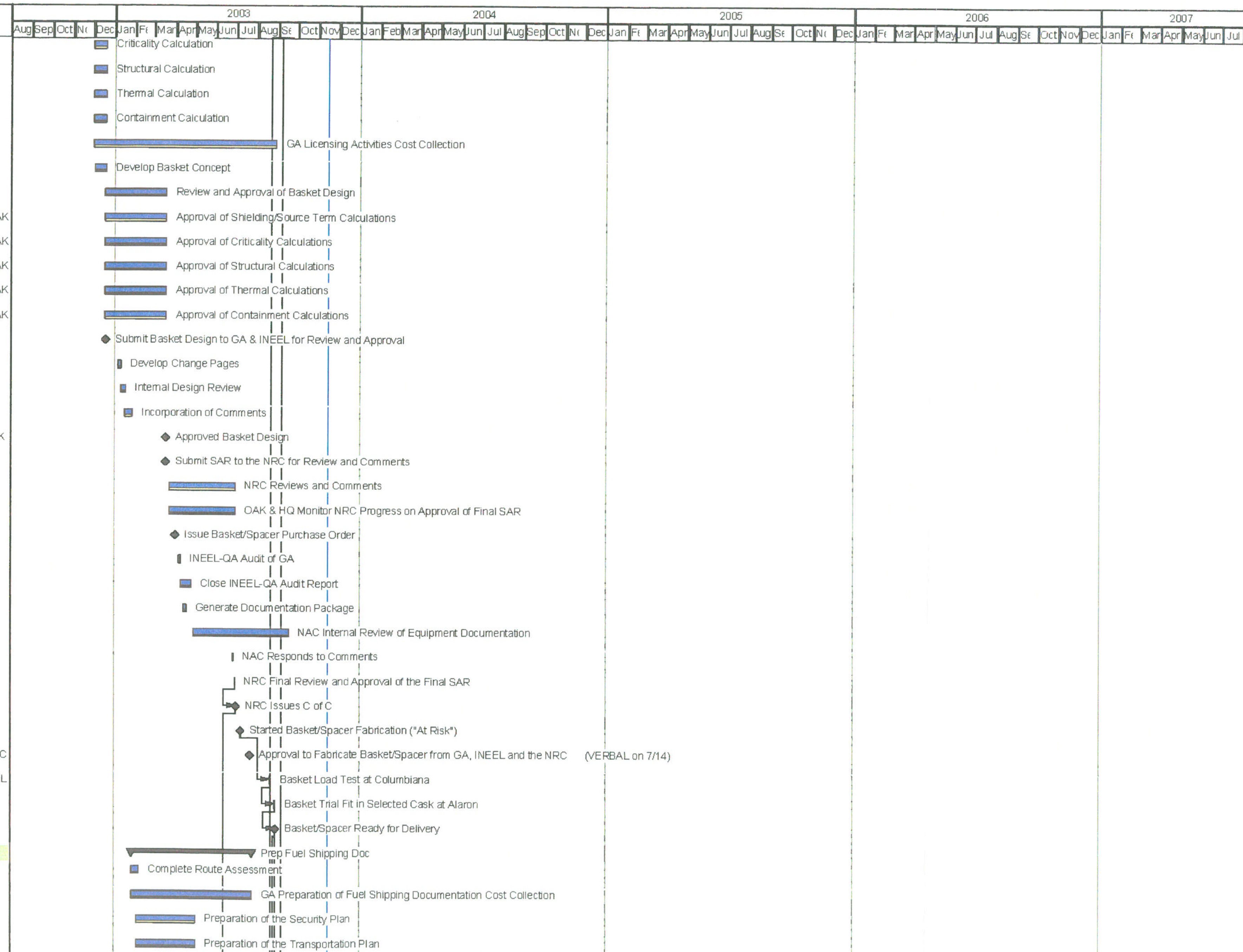
WBS / Activity	ID	DUR	Start	Finish	Status	Cognizant
1484 2442 IFM Disposal-2		473d	15.May.02 A	31.Dec.03		
Proj Mgmt & Tech Support		417d	01.Aug.02 A	19.Dec.03		
1.4.8.4.2.4.4.2.3-2	1.1	63d	01.Aug.02 A	28.Oct.02 A	Completed	OAK, GA
1.4.8.4.2.4.4.2.3-2	3	340d	01.Aug.02 A	19.Dec.03	In Progress	GA
1.4.8.4.2.4.4.2.3-2	1.7	261d	01.Oct.02 A	19.Sep.03 A	Completed	OAK
1.4.8.4.2.4.4.2.3-2	1.5	298d	01.Oct.02 A	10.Dec.03	In Progress	OAK
1.4.8.4.2.4.4.2.3-2	4.1	5d	01.Nov.02 A	07.Nov.02 A	Completed	INEEL, GA
1.4.8.4.2.4.4.2.3-2	3.1.99	0d	05.Nov.02 A		Completed	GA
1.4.8.4.2.4.4.2.3-2	5.1	47d	15.Nov.02 A	03.Feb.03 A	Completed	NAC, GA
1.4.8.4.2.4.4.2.3-2	5.2.99	0d		02.Jan.03 A	Completed	NAC
1.4.8.4.2.4.4.2.3-2	4.2	169d	06.Jan.03 A	04.Jun.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.3-2	1.2	85d	03.Mar.03 A	01.Jul.03 A	Completed	OAK
1.4.8.4.2.4.4.2.3-2	1.2.99	0d	01.Jul.03 A		Completed	OAK
1.4.8.4.2.4.4.2.3-2	4.3	5d	07.Jul.03 A	11.Jul.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.3-2	1.6	77d	02.Sep.03 A	04.Dec.03	In Progress	OAK
1.4.8.4.2.4.4.2.3-2	4.4	6d	25.Sep.03 A	08.Oct.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.3-2	3.2	21d	25.Sep.03 A	26.Nov.03	In Progress	GA
1.4.8.4.2.4.4.2.3-2	3.4	28d	25.Sep.03 A	19.Dec.03	In Progress	GA
1.4.8.4.2.4.4.2.3-2	3.5	55d	25.Sep.03 A	19.Dec.03	In Progress	GA
1.4.8.4.2.4.4.2.3-2	3.3	19d	14.Nov.03	12.Dec.03	Not Started	GA
INEEL Storage Fac/Equip Prep		62d	28.Jan.03 A	30.Jun.03 A		
1.4.8.4.2.4.4.2.2.1-2	4.3	18d	28.Jan.03 A	21.Feb.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.1-2	4.4.99	0d		24.Feb.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.1-2	4.1	48d	17.Apr.03 A	24.Jun.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.1-2	4.2.99	0d		30.Jun.03 A	Completed	INEEL
INEEL's Storage Facility Doc		149d	03.Dec.02 A	02.Oct.03 A		
1.4.8.4.2.4.4.2.2.2-2	4.1	24d	03.Dec.02 A	15.Jan.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	4.3	34d	03.Dec.02 A	29.Jan.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	4.2	55d	03.Dec.02 A	28.Feb.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	4.3.99	0d		29.Jan.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	4.4	66d	30.Jan.03 A	02.May.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	4.10	113d	18.Feb.03 A	16.Sep.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	4.2.99	0d		28.Feb.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	4.4.99	0d		05.May.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	4.5	21d	02.Jun.03 A	01.Jul.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	2.1	27d	20.Jun.03 A	01.Jul.03 A	Completed	ID
1.4.8.4.2.4.4.2.2.2-2	4.11	10d	11.Aug.03 A	02.Oct.03 A	Completed	INEEL
1.4.8.4.2.4.4.2.2.2-2	2.2.99	0d		28.Aug.03 A	Completed	ID
Licensing Activities		270d	01.Dec.02 A	17.Sep.03 A		
1.4.8.4.2.4.4.2.2.3-2	5.2	13d	01.Dec.02 A	18.Dec.02 A	Completed	NAC



Disposal of Hot Cell IFM

20.Nov.03 14:53

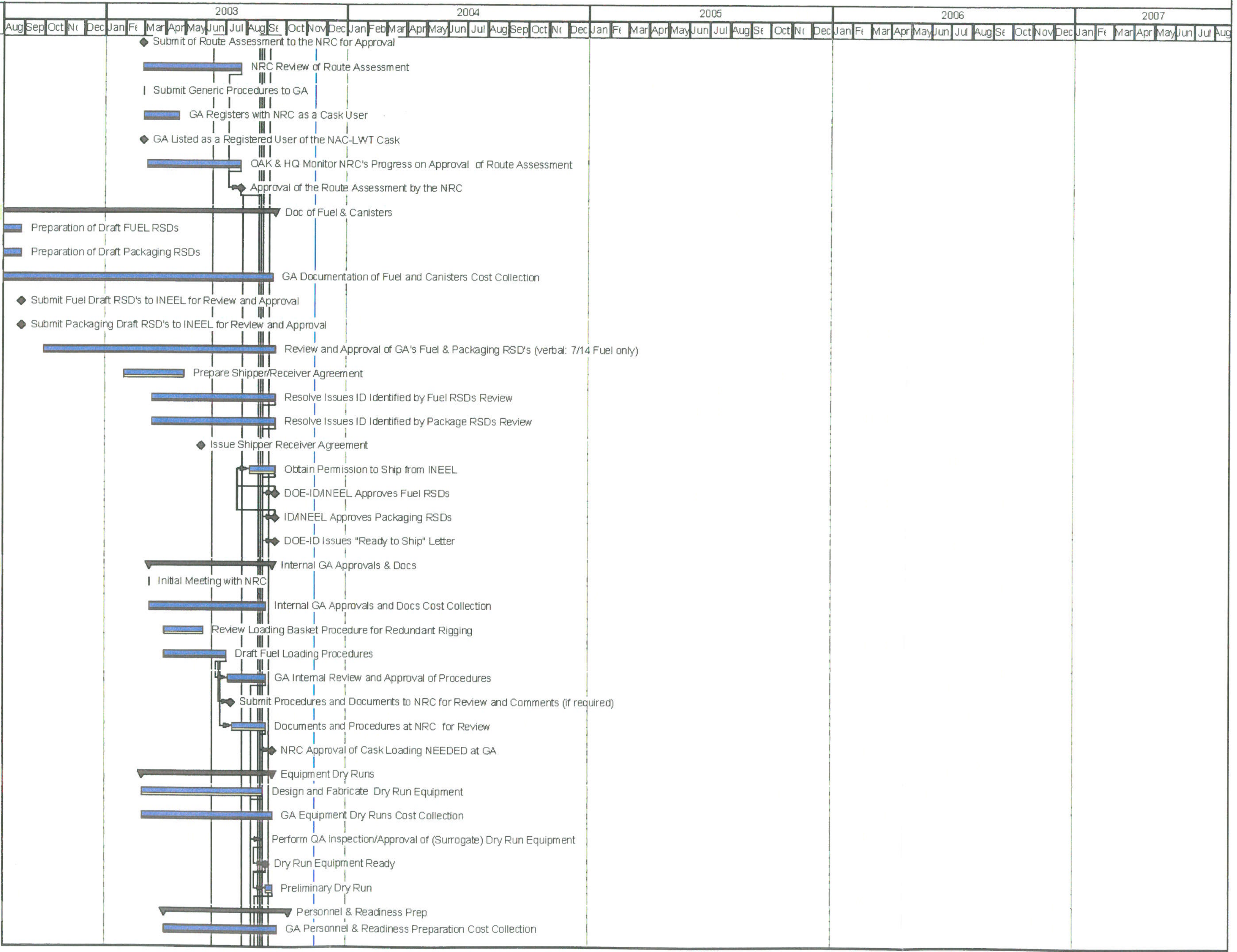
WBS	Activity ID	DUR	Start	Finish	Status	Cognizant
1.4.8.4.2.4.4.2.2.3-2	5.3	13d	01.Dec.02 A	18.Dec.02 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	5.4	13d	01.Dec.02 A	18.Dec.02 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	5.5	13d	01.Dec.02 A	18.Dec.02 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	5.6	13d	01.Dec.02 A	18.Dec.02 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	23	184d	01.Dec.02 A	29.Aug.03 A	Completed	GA
1.4.8.4.2.4.4.2.2.3-2	5.1	13d	02.Dec.02 A	18.Dec.02 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	3.1	56d	18.Dec.02 A	18.Mar.03 A	Completed	GA, INEEL
1.4.8.4.2.4.4.2.2.3-2	3.2	56d	18.Dec.02 A	18.Mar.03 A	Completed	GA, INEEL, OAK
1.4.8.4.2.4.4.2.2.3-2	3.3	56d	18.Dec.02 A	18.Mar.03 A	Completed	GA, INEEL, OAK
1.4.8.4.2.4.4.2.2.3-2	3.4	56d	18.Dec.02 A	18.Mar.03 A	Completed	GA, INEEL, OAK
1.4.8.4.2.4.4.2.2.3-2	3.5	56d	18.Dec.02 A	18.Mar.03 A	Completed	GA, INEEL, OAK
1.4.8.4.2.4.4.2.2.3-2	3.6	56d	18.Dec.02 A	18.Mar.03 A	Completed	GA, INEEL, OAK
1.4.8.4.2.4.4.2.2.3-2	5.1.99	0d	18.Dec.02 A		Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	5.7	5d	06.Jan.03 A	10.Jan.03 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	5.8	4d	11.Jan.03 A	16.Jan.03 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	5.9	8d	16.Jan.03 A	27.Jan.03 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	3.1.99	0d		18.Mar.03 A	Completed	GA, INEEL,OAK
1.4.8.4.2.4.4.2.2.3-2	5.10.99	0d		18.Mar.03 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	6.1	73d	24.Mar.03 A	30.Jun.03 A	Completed	NRC
1.4.8.4.2.4.4.2.2.3-2	1.1	93d	24.Mar.03 A	30.Jun.03 A	Completed	OAK,HQ
1.4.8.4.2.4.4.2.2.3-2	5.98	0d	01.Apr.03 A		Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	4.1	3d	07.Apr.03 A	09.Apr.03 A	Completed	INEEL, GA
1.4.8.4.2.4.4.2.2.3-2	4.2	19d	09.Apr.03 A	24.Apr.03 A	Completed	INEEL, GA
1.4.8.4.2.4.4.2.2.3-2	5.12	5d	14.Apr.03 A	18.Apr.03 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	5.13	80d	29.Apr.03 A	17.Sep.03 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	5.11	0d	27.Jun.03 A	27.Jun.03 A	Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	6.2	20d	30.Jun.03 A	30.Jun.03 A	Completed	NRC
1.4.8.4.2.4.4.2.2.3-2	6.99	0d	01.Jul.03 A		Completed	NRC
1.4.8.4.2.4.4.2.2.3-2	5.14.99	0d	08.Jul.03 A		Completed	NAC
1.4.8.4.2.4.4.2.2.3-2	4.99	0d	22.Jul.03 A		Completed	INEEL, GA, NAC
1.4.8.4.2.4.4.2.2.3-2	3.7	1d	21.Aug.03 A	21.Aug.03 A	Completed	GA, NAC, INEEL
1.4.8.4.2.4.4.2.2.3-2	3.8	2d	28.Aug.03 A	28.Aug.03 A	Completed	GA
1.4.8.4.2.4.4.2.2.3-2	5.99	0d		28.Aug.03 A	Completed	NAC
Prep Fuel Shipping Doc		147d	27.Jan.03 A	25.Jul.03 A		
1.4.8.4.2.4.4.2.2.4-2	5.4	9d	27.Jan.03 A	06.Feb.03 A	Completed	NAC
1.4.8.4.2.4.4.2.2.4-2	24	125d	27.Jan.03 A	25.Jul.03 A	Completed	GA
1.4.8.4.2.4.4.2.2.4-2	5.1	64d	03.Feb.03 A	02.May.03 A	Completed	NAC, TRISM
1.4.8.4.2.4.4.2.2.4-2	5.2	64d	03.Feb.03 A	02.May.03 A	Completed	NAC



Disposal of Hot Cell IFM

20.Nov.03 14:53

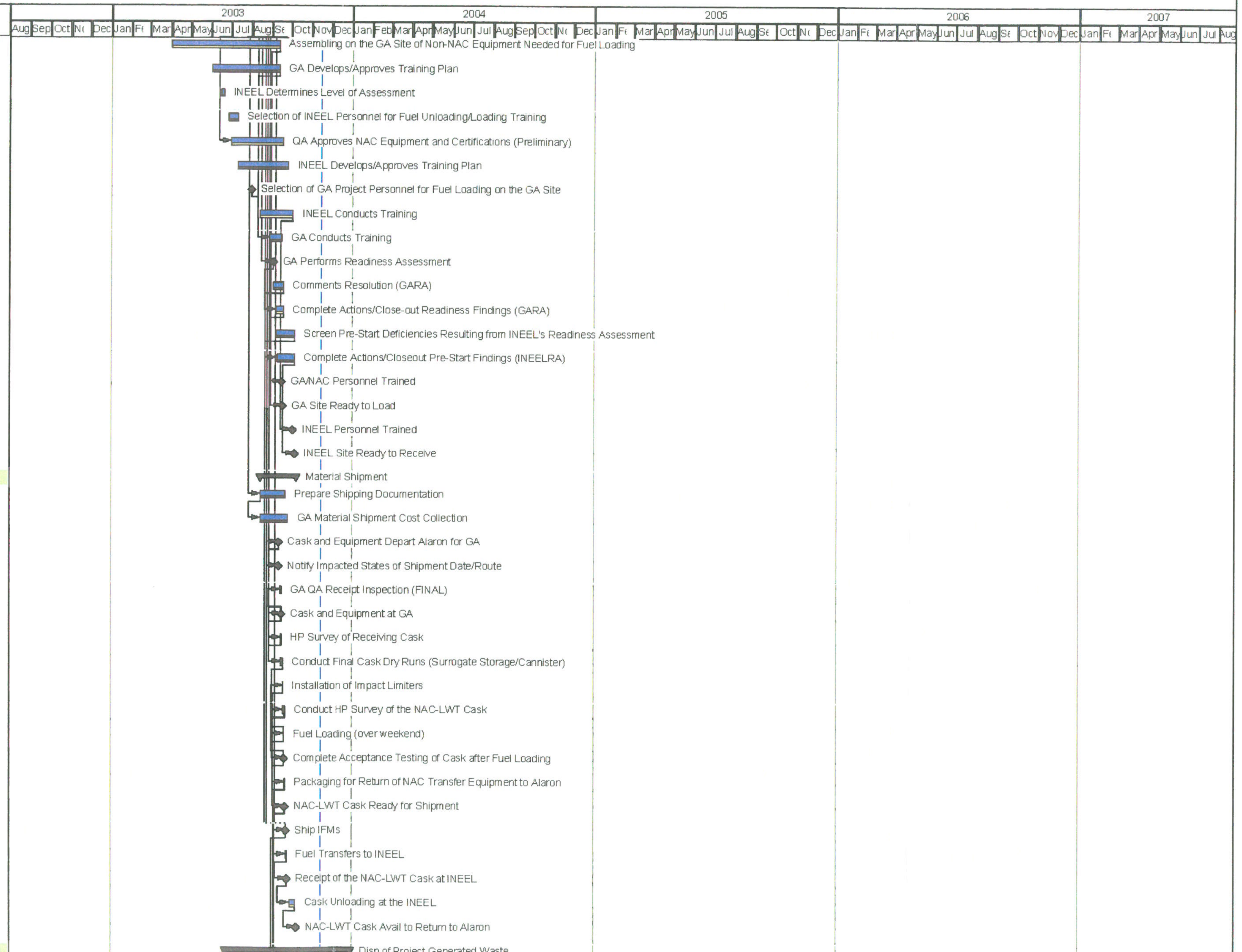
WBS	Activity ID	DUR	Start	Finish	Status	Cognizant
	1.4.8.4.2.4.4.2.2.4-2 5.5.99	0d		27.Feb.03 A	Completed	NAC
	1.4.8.4.2.4.4.2.2.4-2 6.2	87d	27.Feb.03 A	25.Jul.03 A	Completed	NRC
	1.4.8.4.2.4.4.2.2.4-2 5.3	1d	28.Feb.03 A	28.Feb.03 A	Completed	NAC
	1.4.8.4.2.4.4.2.2.4-2 3.1	38d	28.Feb.03 A	22.Apr.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.4-2 6.1.99	0d	28.Feb.03 A		Completed	NRC
	1.4.8.4.2.4.4.2.2.4-2 1.1	121d	05.Mar.03 A	25.Jul.03 A	Completed	OAK, HQ
	1.4.8.4.2.4.4.2.2.4-2 6.2.99	0d		25.Jul.03 A	Completed	NRC
	Doc of Fuel & Canisters	451d	15.May.02 A	16.Sep.03 A		
	1.4.8.4.2.4.4.2.2.5-2 3.2	76d	15.May.02 A	28.Aug.02 A	Completed	GA, OAK
	1.4.8.4.2.4.4.2.2.5-2 3.4	76d	15.May.02 A	28.Aug.02 A	Completed	GA, OAK
	1.4.8.4.2.4.4.2.2.5-2 25	333d	15.May.02 A	11.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.5-2 3.2.99	0d	29.Aug.02 A		Completed	GA
	1.4.8.4.2.4.4.2.2.5-2 3.5.99	0d	29.Aug.02 A		Completed	GA
	1.4.8.4.2.4.4.2.2.5-2 4.1	225d	01.Oct.02 A	16.Sep.03 A	Completed	INEEL
	1.4.8.4.2.4.4.2.2.5-2 2.1	64d	30.Jan.03 A	30.Apr.03 A	Completed	ID, OAK
	1.4.8.4.2.4.4.2.2.5-2 3.3	108d	14.Mar.03 A	16.Sep.03 A	Completed	GA, INEEL
	1.4.8.4.2.4.4.2.2.5-2 3.6	118d	14.Mar.03 A	16.Sep.03 A	Completed	GA, INEEL
	1.4.8.4.2.4.4.2.2.5-2 2.1.99	0d		27.May.03 A	Completed	ID, OAK
	1.4.8.4.2.4.4.2.2.5-2 1.1	10d	08.Aug.03 A	16.Sep.03 A	Completed	OAK, ID
	1.4.8.4.2.4.4.2.2.5-2 2.98	0d		16.Sep.03 A	Completed	ID, INEEL
	1.4.8.4.2.4.4.2.2.5-2 2.99	0d		16.Sep.03 A	Completed	ID, INEEL
	1.4.8.4.2.4.4.2.2.5-2 2.2.99	0d		16.Sep.03 A	Completed	ID
	Internal GA Approvals & Docs	128d	10.Mar.03 A	12.Sep.03 A		
	1.4.8.4.2.4.4.2.2.6-2 3.1	1d	10.Mar.03 A	10.Mar.03 A	Completed	GA, NRC
	1.4.8.4.2.4.4.2.2.6-2 26	119d	10.Mar.03 A	02.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.6-2 3.4	43d	01.Apr.03 A	30.May.03 A	Completed	GA, NAC
	1.4.8.4.2.4.4.2.2.6-2 3.2	54d	01.Apr.03 A	03.Jul.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.6-2 3.3	39d	07.Jul.03 A	02.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.6-2 3.99	0d	11.Jul.03 A		Completed	GA
	1.4.8.4.2.4.4.2.2.6-2 6.1	36d	14.Jul.03 A	02.Sep.03 A	Completed	NRC
	1.4.8.4.2.4.4.2.2.6-2 6.1.99	0d	12.Sep.03 A		Completed	NRC
	Equipment Dry Runs	129d	27.Feb.03 A	12.Sep.03 A		
	1.4.8.4.2.4.4.2.2.7-2 3.1	129d	27.Feb.03 A	28.Aug.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.7-2 27	120d	27.Feb.03 A	12.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.7-2 3.2	1d	29.Aug.03 A	29.Aug.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.7-2 3.2.99	0d	02.Sep.03 A		Completed	GA
	1.4.8.4.2.4.4.2.2.7-2 3.3	1d	02.Sep.03 A	12.Sep.03 A	Completed	GA
	Personnel & Readiness Prep	131d	01.Apr.03 A	06.Oct.03 A		
	1.4.8.4.2.4.4.2.2.8-2 28	110d	01.Apr.03 A	18.Sep.03 A	Completed	GA



Disposal of Hot Cell IFM

20.Nov.03 14:53

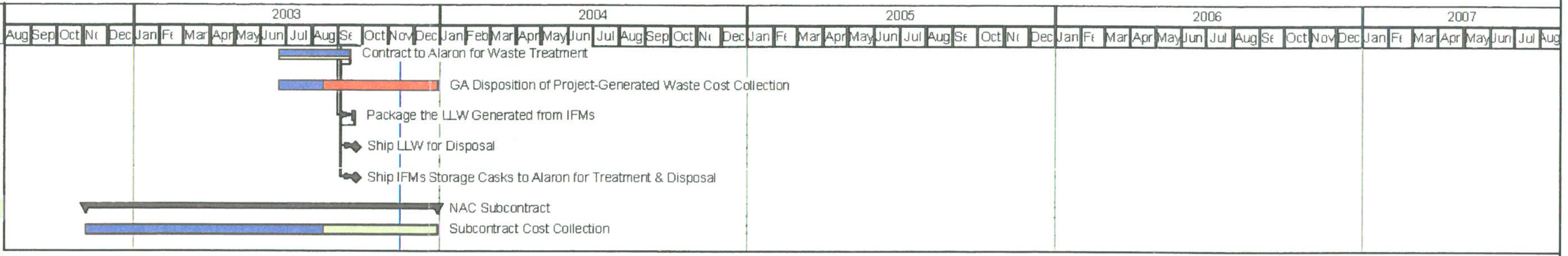
WBS	Activity	ID	DUR	Start	Finish	Status	Cognizant
	1.4.8.4.2.4.4.2.2.8-2	3.3	105d	01.Apr.03 A	12.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	3.1	44d	02.Jun.03 A	12.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	4.1	5d	16.Jun.03 A	20.Jun.03 A	Completed	INEEL
	1.4.8.4.2.4.4.2.2.8-2	2.1	12d	27.Jun.03 A	10.Jul.03 A	Completed	ID
	1.4.8.4.2.4.4.2.2.8-2	3.4	44d	01.Jul.03 A	17.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	2.2	21d	11.Jul.03 A	25.Sep.03 A	Completed	ID
	1.4.8.4.2.4.4.2.2.8-2	3.2	0d	01.Aug.03 A		Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	2.3	0d	14.Aug.03 A	03.Oct.03 A	Completed	ID
	1.4.8.4.2.4.4.2.2.8-2	3.7	9d	29.Aug.03 A	16.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	3.97	0d	03.Sep.03 A		Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	3.5	3d	03.Sep.03 A	18.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	3.6	5d	08.Sep.03 A	18.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	4.2	1d	08.Sep.03 A	06.Oct.03 A	Completed	INEEL
	1.4.8.4.2.4.4.2.2.8-2	4.3	2d	09.Sep.03 A	06.Oct.03 A	Completed	INEEL
	1.4.8.4.2.4.4.2.2.8-2	3.99	0d		16.Sep.03 A	Completed	GA, NAC
	1.4.8.4.2.4.4.2.2.8-2	3.98	0d	18.Sep.03 A		Completed	GA
	1.4.8.4.2.4.4.2.2.8-2	2.3.99	0d		03.Oct.03 A	Completed	ID
	1.4.8.4.2.4.4.2.2.8-2	4.99	0d	06.Oct.03 A		Completed	INEEL
	Material Shipment		38d	15.Aug.03 A	08.Oct.03 A		
	1.4.8.4.2.4.4.2.2.9-2	3.2	20d	15.Aug.03 A	22.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.9-2	29	29d	15.Aug.03 A	25.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.9-2	5.1	0d	12.Sep.03 A		Completed	NAC
	1.4.8.4.2.4.4.2.2.9-2	5.7	0d	12.Sep.03 A		Completed	NAC
	1.4.8.4.2.4.4.2.2.9-2	3.97	0d	15.Sep.03 A	16.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.9-2	5.2	0d	16.Sep.03 A		Completed	NAC
	1.4.8.4.2.4.4.2.2.9-2	3.98	0d	16.Sep.03 A	16.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.9-2	3.1	4d	16.Sep.03 A	18.Sep.03 A	Completed	GA/NAC
	1.4.8.4.2.4.4.2.2.9-2	5.6	2d	19.Sep.03 A	19.Sep.03 A	Completed	NAC, GA
	1.4.8.4.2.4.4.2.2.9-2	3.3	2d	19.Sep.03 A	22.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.9-2	5.3.99	3d	20.Sep.03 A	20.Sep.03 A	Completed	NAC, GA
	1.4.8.4.2.4.4.2.2.9-2	5.4.99	0d		20.Sep.03 A	Completed	NAC, GA
	1.4.8.4.2.4.4.2.2.9-2	5.5	6d	22.Sep.03 A	22.Sep.03 A	Completed	NAC, GA
	1.4.8.4.2.4.4.2.2.9-2	5.6.99	0d		22.Sep.03 A	Completed	NAC, GA
	1.4.8.4.2.4.4.2.2.9-2	3.99	0d	23.Sep.03 A		Completed	GA
	1.4.8.4.2.4.4.2.2.9-2	5.8	3d	23.Sep.03 A	24.Sep.03 A	Completed	TRISM
	1.4.8.4.2.4.4.2.2.9-2	4.1.99	0d		24.Sep.03 A	Completed	INEEL
	1.4.8.4.2.4.4.2.2.9-2	4.2	7d	29.Sep.03 A	07.Oct.03 A	Completed	INEEL
	1.4.8.4.2.4.4.2.2.9-2	5.9	0d		08.Oct.03 A	Completed	NAC
	Disp of Project-Generated Waste		57d	23.Jun.03 A	31.Dec.03		



Disposal of Hot Cell IFM

20.Nov.03 14:53

WBS	Activity	ID	DUR	Start	Finish	Status	Cognizant
	1.4.8.4.2.4.4.2.2.10-2	3.3	42d	23.Jun.03 A	16.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.10-2	210	68d	23.Jun.03 A	31.Dec.03	In Progress	GA
	1.4.8.4.2.4.4.2.2.10-2	3.1	6d	19.Sep.03 A	22.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.10-2	3.2.99	0d		23.Sep.03 A	Completed	GA
	1.4.8.4.2.4.4.2.2.10-2	3.4.99	0d		23.Sep.03 A	Completed	GA
	NAC Subcontract		76d	05.Nov.02 A	31.Dec.03		
	NAC Subcontract-2	A1000	323d	05.Nov.02 A	31.Dec.03	In Progress	



APPENDIX F
IFM SHIPPING DOCUMENTATION PACKAGE

Uniform Domestic Straight Bill of Lading, adopted by Carriers in Official, Southern, Western and Illinois Classification territories, March 15, 1992, as amended June 15, 1941

UNIFORM STRAIGHT BILL OF LADING - ORIGINAL - NOT NEGOTAIBLE PC-000512/0

RECEIVED, subject to the classifications and tariffs in effect on the date of the issue of this Bill of Lading, the property described below, in apparent food order, except as noted (contents and condition of contents of packages unknown), marked consigned, and destined as indicated below, which said carrier (the work carrier being understood throughout the contract as meaning and person or corporation in possession of the property under the contract) agrees to carry to its usual place of delivery at said destination, if on its route, otherwise to deliver to another carrier on the route to said destination. It is mutually agreed as to each carrier of all or any of said property over all or any portion of said route to destination, and as to each party at any time interested in all or any of said property, that every service to be performed hereunder shall be subject to all the terms and conditions of the Uniform domestic Straight Bill of lading set forth (1) in Official, Southern, Western and Illinois Freight Classifications in effect on the date hereof. If this is a rail or rail-water shipment, or (2) in the applicable motor carrier classification on tariff in this is a motor carrier shipment.

Shipper hereby certifies that he is familiar with all the terms and condition of the said bill of lading, including those on the back thereof, set forth in the classification or tariff which governs the transportation of the shipment, and the said terms and conditions are hereby agreed to by the shipper and accepted for himself and his assigns.

FOR ACCOUNT OF SHIPPER

General Atomics
3550 General Atomic Court
San Diego, CA 92121-1122
POC: Keith Asmussen 1-858-455-2823

DATE	REFERENCE NO
23 Sept 2003	JXI-0007

VESSEL / AIRLINE/TRUCK	LOCATION OF GOODS	PORT OF LADING
Truck	San Diego, CA, USA	N/A

B/L NO.	ARRIVAL DATE	FREE TIME EXP.	DELIVERY ORDER ISSUED TO
1 of 1	SAFEGUARDS	-0-	Tri-State Motor Transit Company (TSMT), Joplin, MO, USA

ORIGINATIONG CARRIER
TSMT

CONSIGNED TO
Bechtel BWXT Idaho, LLC (INTEC)

Technical Operations, Safeguards Unit MS-5102

Idaho Falls, ID 83415 POC Ernie Label 1-208-526-2113

ROUTE
TSMT direct via NRC Approved Route

DELIVERING CARRIER
TSMT

TRACTOR	TRAILER
43077	238911

NO. OF PACKAGES	DESCRIPTION OF MERCHANDISE, MARKS AND NUMBERS	WEIGHT	CLASS RATE
1	**RQ, RADIOACTIVE MATERIAL, Type B(U) Package, FISSILE NOS; CLASS 7; UN 2918; Highway Route Controlled Quantity Physical Form: Solid; Chemical Form: Uranium, Plutonium Oxides; Radioactive Yellow III Labels; Exclusive Use Shipment; (Instructions Provided to Drivers and Escorts); DOT North American Emergency Response Guide No. 165 Attached** 20' ISO Container, NACU 101273(2), with NAC LWT-7 Cask Comp Auth: USA/9225/B(U)F-85 Containing 2 canisters of Irradiated Nuclear Fuel Rods and Pieces Radionuclides: U, MFP; U: 3211.81g; U235: 460.0g; Pu: 30.09g; 98.42 TBq; (2660 Ci) TI: 0.2 Seals: Cask: GAHC 0580; Doors: 0601; 0602 Max Dose: @ Surface: <0.02 mSv/h; @ 1 Meter: <0.002 mSv/h **24 HOUR EMERGENCY CONTACT: 800-241-0507 x 333** This is to certify that the above named materials are properly classified, described, packaged, marked and labeled, and are in proper condition for transportation according to the applicable regulations of the Department of Transportation. ______________________ Tractor Equipped with Telecommunications Equipment and Immobilization Device. Continuous Surveillance of Shipment Required. Status Check Required every 2 hours via Qualcomm or Telephone to 800-241-0507 x 333.	62,500 #	

Subject to Section 7 of condition, if this shipment is to be delivered to the consignee without recourse on the consignor, the consignor shall sing the following statement The carrier shall not make delivery of this shipment without payment of freight and all other lawful charges. ....N/A......... (Signature of consignor)	If charges are to be prepaid write or stake here, "To Be Prepaid." ......PPD......	Received $......... To apply in prepayment of the charges on the property described hereon. ....N/A......... Agent of Cashier Per .... (The signature here acknowledges only the amount prepaid)	Charges advanced: $.........N/A.........

If the shipment moves between two ports by a carrier by water, the law requires that the bill of lading shall state whether it is "carrier's or shipper's weight." NOTE-Where the rate id dependent on value, shippers are required to state specifically in writing the agreed or declared value of the property. The agreed or declared value of the property is hereby specifically stated by the shippers to be not exceeding

________LISTED________________________ per __TSMT/NAC Agreed Rates

______________________ Carrier Agent / Date______________

______________________ NAC Intl/GA Agent

APPENDIX G
PHOTO DOCUMENTATION OF IFM SHIPMENT OPERATIONS

Fig. G-1. Hoist and Removal of NAC-LWT Cask
from ISO Shipping Container in GA Bldg. 21 Yard, 9/16/03

Fig. G-2. NAC-LWT Cask Uprighted/Positioned/
Tied-down to Cask Base Fixture, 9/16/03

Fig. G-3. NAC Dry Transfer System (DTS) Staged
in Bldg. 21 Yard, 9/17/03

Fig. G-4. Operational Check of NAC Basket Grapple Fixture, 9/17/03

Fig. G-5. IFM Basket (NAC Top Module) Insertion into Interim Transfer System (ITS), 9/17/03

Fig. G-6. Hoist of NAC ITS Inner Shield over Dry Pit in Bldg. 21, Rm. 21/111, 9/17/03

Fig. G-7. Hoist of GA Storage Cask containing HTGR/IFM
(note redundant rigging), 9/19/03

Fig. G-8. Lowering of HTGR/IFM Storage Cask into Dry Pit in
Rm. 21/111, 9/19/03

Fig. G-9. Hoist of GA Storage Cask containing RERTR/IFM, (note redundant rigging), 9/19/03

Fig. G-10. Lowering of RERTR/IFM Storage Cask thru Roof Hatch over Rm. 21/111, 9/19/03

Fig. G-11. Rm. 21/111 Dry Pit, showing positioning of GA Storage Casks and NAC ITS, 9/19/03

Fig. G-12. CCTV Image: Exposed HTGR/IFM Canister in GA Storage Cask, 9/20/03

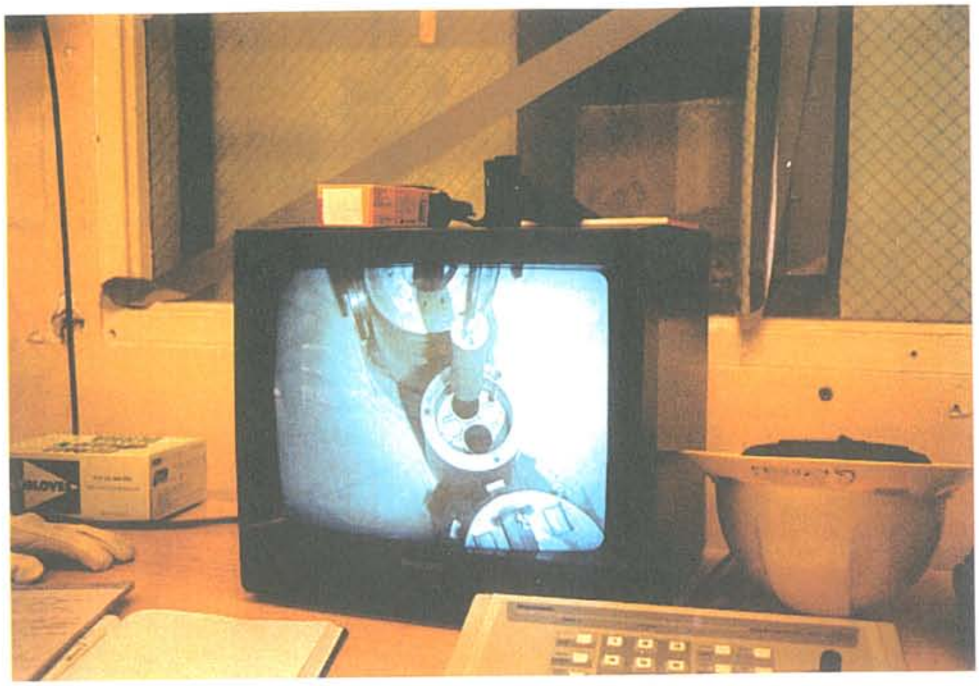


Fig. G-13. CCTV Image: Transfer of Grappled HTGR/IFM Canister into NAC Basket, 9/20/03

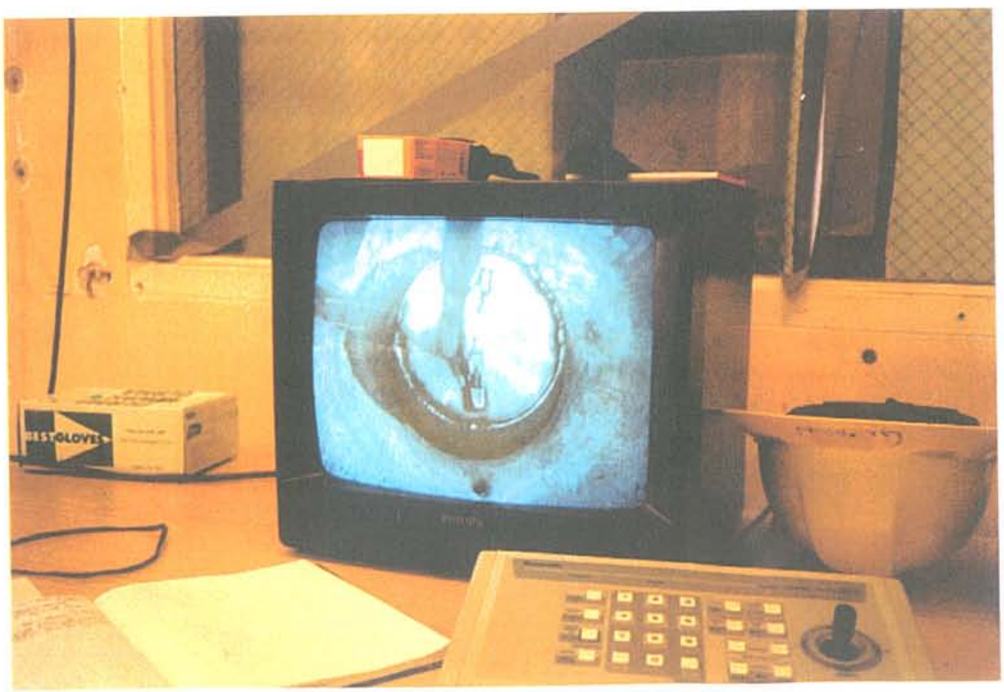


Fig. G-14. CCTV Image: Remote Grapple of RERTR/IFM Canister inside GA Storage Cask, 9/20/03

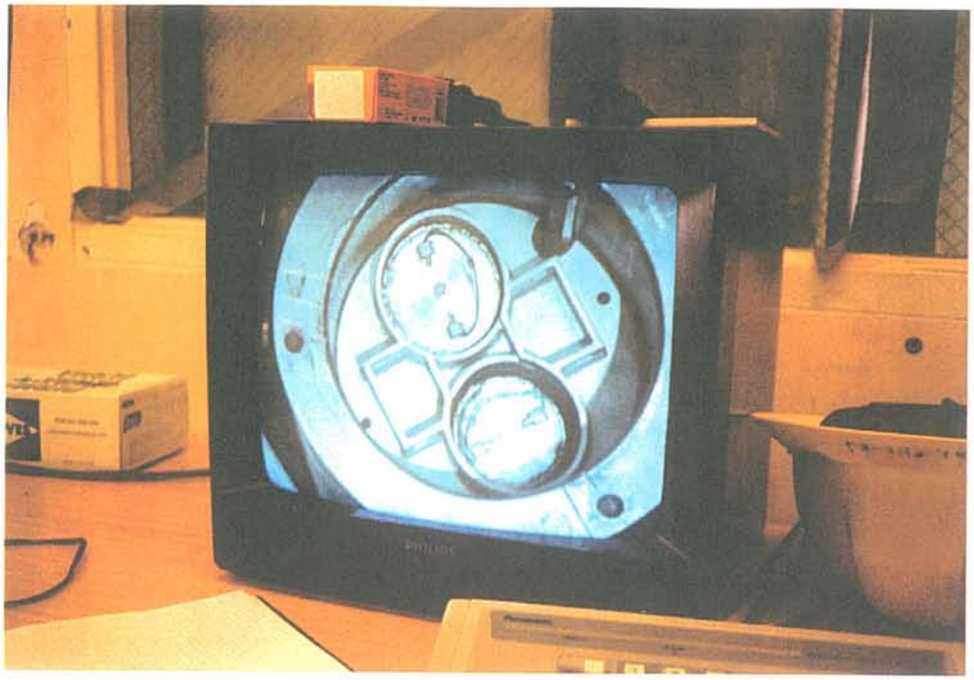


Fig. G-15. CCTV Image: HTGR & RERTR/IFM Canisters Loaded into NAC Basket, 9/20/03

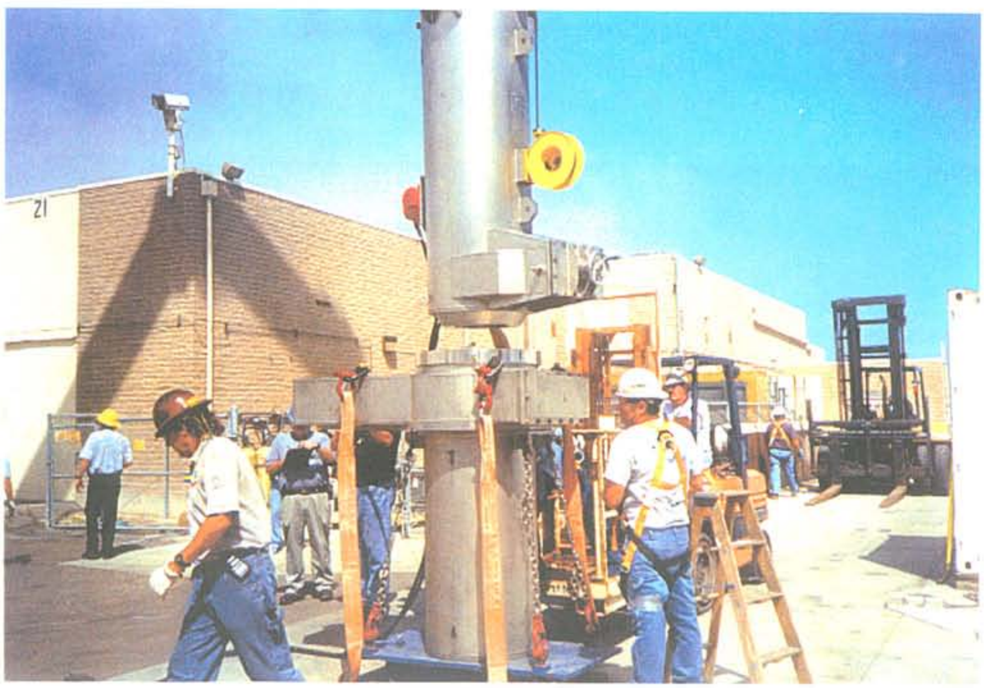


Fig. G-16. Placement of NAC DTS atop NAC ITS and Shield Gate, Bldg. 21 Yard, 9/20/03

Fig. G-17. NAC DTS Placement atop NAC-LWT Cask,
Foreground (L to R), J. Lee (DOE/OAK), R. Develasco (GA),
J. Davis & R. Claverie (DOE/OAK), 9/20/03

Fig. G-18. Closure Lid Placement on NAC-LWT Cask, 9/20/03

Fig. G-19. Hoist and Translation of Loaded NAC-LWT Cask, 9/20/03

Fig. G-20. NAC-LWT Cask in ISO Container,
showing Leak Check Fixtures, 9/20/03

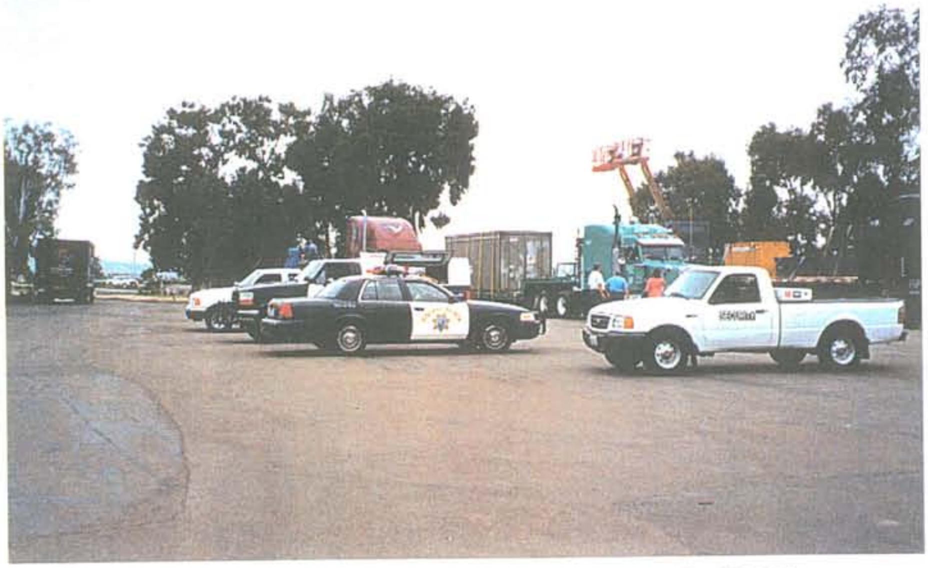


Fig. G-21. NAC-LWT Cask Transport Vehicle, with CHP Escort Vehicles in foreground, 9/23/03

Fig. G-22. Project-related Rad Waste and NAC Equipment Transport Vehicle, 9/23/03

Fig. G-23. NAC-LWT Cask Transport Vehicle Departing GA Site, 9/23/03