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1	1	DW Bergmann	<i>DW Bergmann</i>	8-31-94	L6-24	JL Rathbun	<i>JL Rathbun</i>	9/9/94	R3-09	1	1
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7. Abstract <i>10/13/94 D. Volel</i> The purpose of this Safety Evaluation for Packaging (SEP) is to authorize the ChemNuclear CNS 1-13G packaging to ship samples of irradiated fuel elements from the 100 K East and 100 K West basins to the Postirradiation Testing Laboratory (PTL) in support of the spent nuclear fuel characterization effort. It also authorizes the return of the fuel element samples to the 100 K East facility using the same packaging. The CNS 1-13G cask has been chosen to transport the fuel because it has a Certificate of Compliance (CoC) issued by the U.S. Nuclear Regulatory Commission (NRC) for transporting irradiated oxide and metal fuel in commerce. It is capable of being loaded and offloaded underwater and may be shipped with water in the payload compartment.		
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LIST OF TERMS

ASTM	American Society for Testing and Materials
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
DOT	U.S. Department of Transportation
EDE	effective dose equivalent
FHA	Federal Highway Administration
HRCQ	highway route controlled quantity
ID	inside dimension
NRC	U.S. Nuclear Regulatory Commission
PTL	Postirradiation Testing Laboratory
SARP	Safety Analysis Report for Packaging
SEP	Safety Evaluation for Packaging
SFEC	Single Fuel Element Container
SPR	single pass reactor
WHC	Westinghouse Hanford Company

PART A - GENERAL INFORMATION AND OPERATIONAL CONTROLS

1.0 INTRODUCTION

1.1 BACKGROUND INFORMATION

Spent fuel from the N Reactor and the single pass reactors (SPRs) is presently being stored at the K East and K West Basins. Characterization of the various fuels is essential to evaluate the various interim storage systems and determine the suitability of each. Consequently, it is necessary to transport a cross section of fuel elements from the K Basins to the hot cells at the Postirradiation Testing Laboratory (PTL) in the 300 Area for analysis.

To accomplish this, ChemNuclear's CNS 1-13G cask has been chosen to transport the fuel. This cask was chosen because it has a Certificate of Compliance (CoC) issued by the U.S. Nuclear Regulatory Commission (NRC) for transporting irradiated oxide and metal fuel in commerce. It is capable of being loaded and offloaded underwater and may be shipped with water in the payload compartment.

It is expected that a significant number of the fuel elements will have damaged or failed cladding. Although the CoC does not explicitly forbid the transport of irradiated fuel with failed cladding, the Safety Analysis Report for Packaging (SARP) does not directly address the possibility of shipping breached fuel. To help ensure a degree of safety equivalent to that of a shipment made in commerce, the fuel shall be transported in a water filled inner container to compensate for the possibility of cladding failure.

1.2 SYSTEM DESCRIPTION

The CNS 1-13G packaging consists of a cask and an overpack. Additional lead and polyethylene liners are available from the manufacturer, but will not be used during this campaign. An inner container will be used to assist in maintaining the fuel in a non-reactive environment and to limit its motion while in the shipping cask. The gross weight of the cask and overpack is 25,500 lb. Drawings for the packaging are contained in Part A, Section 9.2.1.

1.2.1 Cask

The cask consists of a right circular cylinder 68 in. high by 38½ in. in diameter. The cask cavity is 54 in. high by 26½ in. in diameter. The cask side wall consists of ½ in. thick outer steel shell, a 5 in. lead shell and a ½ in. thick inner steel shell. The outer base of the cask is comprised of a ¼ in. steel shim plate and backing ring and a ½ in. steel plate welded to form a ¾ in. thick base which is integrally welded to the outer steel shell of the

side wall. The cask lid is a lead-filled flanged plug. The cask closure is sealed by a silicone rubber gasket. Positive closure is accomplished by six 1 in. diameter studs. The cask is equipped with a cavity drain line and plug. For lifting, the cask is provided with two ears bolted and seal welded to the cask body 180° apart. The closure lid is provided with a single lifting lug.

1.2.2 Overpack

During transport, an overpack provides additional protection for the cask. The overpack consists of a double-walled steel cylinder which encloses the cask and is bolted to a steel pallet with 2 in. diameter studs. The overpack is equipped with two rectangular lifting loops and two tiedown ears.

1.2.3 Single Fuel Element Container

The Single Fuel Element Container (SFEC) is constructed of stainless steel. It consists of a 27 in. right circular cylinder 4 in. in diameter equipped with a lid, locking mechanism and lifting hoop. With lid in place the total height of the canister is 29.6 in. and the maximum width is 6.87 in. The internal cavity is 27 in. long by 3.88 in. in diameter. To allow for the venting of any hydrogen, the canister is equipped with a 1/16 in. inside dimension (ID) vent. To allow for greater compatibility with existing K Basin and PTL fixturing, two 5/16 in. thick blocks have been welded to the bottom. It is also equipped with a hoop for lifting. The CNS 1-13G will accommodate up to three SFECs in a rack. Part A, Section 9.2.2 contains the drawings for the SFEC and SFEC rack.

1.3 PURPOSE AND SCOPE

The purpose of the Safety Evaluation for Packaging (SEP) is to authorize the CNS 1-13G packaging to ship samples of irradiated fuel elements from the 100 K East and 100 K West basins to the PTL in support of the spent nuclear fuel characterization effort. It also authorizes the return of the fuel element samples to the 100 K East facility using the same packaging.

Each shipment will transport up to two full N Reactor fuel assemblies or three outer N Reactor Fuel elements. The scope and number of shipments is dependant on the requirements of the characterization effort. This SEP shall expire one year after the date of release or when the last fuel samples required for the characterization effort have been transported back to the 100 K East Facility.

2.0 PACKAGING SYSTEM

2.1 CONFIGURATION AND DIMENSIONS

A sketch of the CNS 1-13G cask with overpack is presented in Figure A2-1.

The nominal dimensions of the CNS 1-13G are:

Overall Height, Without Overpack	68.0 in.
Outside Diameter, Without Overpack	38.5 in.
Inner Cavity Height	54.0 in.
Inner Cavity Diameter	26.5 in.
Overall Height, With Overpack	85.4 in.
Outside Diameter, With Overpack	68.0 in.

Figure A2-2 contains a sketch of the SFEC that will contain the irradiated fuel elements. The nominal dimensions of the SFEC are:

Overall Height (without lifting hoop)	29.6 in.
Maximum Outer Diameter	5.9 in.
Inner Cavity Height	27.0 in.
Inner Cavity Diameter	3.88 in.

2.2 MATERIALS OF CONSTRUCTION

The CNS 1-13G is constructed of American Society for Testing and Materials (ASTM) 304 stainless steel with a radiation shield constructed of 5 in. of lead. A high temperature silicone rubber gasket is incorporated to provide a seal for the cask lid. The overpack is constructed of painted mild carbon steel (1020 Hot Roll).

The SFEC is constructed of ASTM 304 stainless steel, with an elastomer O-ring to provide a seal for the container lid.

Figure A2-1. CNS 1-13G Cask and Overpack.

Figure A2-2. Single Fuel Element Canister.

2.3 MECHANICAL PROPERTIES OF MATERIALS

The mechanical properties of the material used to construct the CNS 1-13G cask, overpack, and SFEC are presented in Table A2-1. The lead shielding of the CNS 1-13G is not a load bearing member.

Table A2-1. Mechanical Properties of CNS 1-13G Cask, Overpack and SFEC.

Material	Yield Strength (psi)	Ultimate Strength (psi)	Poisson's Ratio
304 SS	30,000	75,000	0.305
1020 HR	40,000	69,000	0.290

2.4 DESIGN AND FABRICATION METHODS

The CNS 1-13G was designed and fabricated by ChemNuclear in accordance with the requirements of Title 10, *Code of Federal Regulations*, Part 71 (10 CFR 71).

The SFEC was designed by the Westinghouse Hanford Company (WHC) for this shipping campaign. Fabrication is per Drawing H-1-80394.

2.5 WEIGHTS AND CENTER OF GRAVITY

Table A2-1 contains the weights of the components of the CNS 1-13G. The center of gravity of the package is located at the geometric center.

Table A2-1. Weights of CNS 1-13G Components.

Component	Weight (lb)
Cask Body	17,100
Cask Lid	2,000
Overpack Baseplate	1,400
Overpack Shroud	5,000
Total	25,500

The unloaded weight of the SFEC with lid is 18.0 lb. The center of gravity in the unloaded SFEC will be located along the axial centerline of the package, 16.2 in. above the bottom of the container.

2.6 CONTAINMENT BOUNDARY

The containment boundaries for the shipment shall be provided by the cask body, lid, gasket, and lid bolts.

The SFEC will be continuously vented to the inner cavity of the cask. It will provide a confinement barrier through its body, lid, and O-ring gasket.

2.7 CAVITY SIZE

The internal cavity of the cask is 54.0 in. tall and 26.5 in. in diameter.

The internal cavity of the SFEC is 27.0 in. tall and 3.88 in. in diameter.

2.8 HEAT DISSIPATION

All heat generated by payloads within the CNS 1-13G are dissipated to the environment via convection and radiation to the ambient environment. The CNS 1-13G cask is capable of dissipating up to 600 W of decay heat in this manner (ChemNuclear 1992).

2.9 SHIELDING

Shielding for the CNS 1-13G is provided by 5 in. of lead and two $\frac{1}{2}$ in. layers of steel on the walls, floor, and lid of the cask. Additional shielding is provided by the $\frac{1}{2}$ in. thick steel of the overpack.

The SFEC will provide negligible shielding for the payload.

2.10 LIFTING DEVICES

The cask is outfitted with two 10 in. by 19.375 in. lifting ears. These ears are bolted and welded onto the cask body. The overpack is equipped with two lifting ears on the top of the overpack. The cask shall be placed on the transport vehicle prior to placement of the overpack, therefore the lifting ears on the overpack shall not be used to lift the packaging.

Lifting of the SFEC is accomplished via a lifting hoop attached to body of the canister. During transport operations, the hoop shall be moved to minimize canister height as much as is practical.

2.11 TIEDOWN DEVICES

The CNS 1-13G is bolted directly to a specially equipped trailer. It is also equipped with two tiedown points constructed out of 3 in. thick steel plate welded to the overpack body.

3.0 PACKAGE CONTENTS

3.1 GENERAL DESCRIPTION

The CNS 1-13G cask will be utilized to transport irradiated N Reactor and SPR fuel. The vast majority of the irradiated fuel in the K Basins is from the N Reactor. Since SPR fuel is significantly smaller and has decayed for a much longer period of time than the N Reactor fuel, N Reactor fuel is considered to represent the worst case source term. N Reactor fuel consists of zircalloy clad outer and inner elements which have a maximum length of 26.1 in. and have outer diameters of 2.42 in. and 1.28 in., respectively. The inner element has a maximum weight of 16.5 lb (7,484 g) and the outer element has a maximum weight of 35.2 lb (15,967 g). The elements have been stored in the East and West K Basins for periods of at least seven years. Some elements that are scheduled to be shipped suffered cladding failure while in the reactor or experience significant damage during the discharge process. Other elements have undergone significant corrosion/damage during their storage in the basins. Some of these elements may be broken or similarly degraded.

To help maintain an environment similar to that found in the K Basins, and to prevent possible reactions between the fuel and the cask atmosphere, the fuel shall be shipped with basin water in the SFEC.

3.2 CONTENT RESTRICTIONS

3.2.1 Content Matrix

The CNS 1-13G transfer cask will be permitted to carry up to two inner and two outer fuel elements or three outer fuel elements per shipment held in up to three SFECs, provided that the total amount ^{235}U fissile equivalent material is less than 500 g. Additionally, the contents shall be restricted so that the concentration of hydrogen within the CNS 1-13G will not exceed 2.5%.

3.2.2 Radioactive Materials

The fuel portion of the elements that will be shipped in the CNS 1-13G contain significant amounts of transuranic materials such as ^{240}Pu and ^{239}Pu as well as fission productions such as $^{137}\text{Cs}/^{137}\text{Ba}_m$. The cladding will contain activated material such as ^{60}Co . Table A3-1 provides a list of the worst expected activity levels for an inner and outer fuel element and the worst anticipated activity level for a shipment with 500 ^{235}U equivalent grams of fissile material. The total activity for a worst case payload is 3,095 Ci. The worst case payload will be comprised of approximately 13,000 A₂s of radioactive material, which classifies the payload as Type B, highway route control quantities (HRCQ).

Table A3-1. Activity of Worst Case Inner and Outer N Reactor Fuel Element.

Specie	Inner Fuel Content (Ci)	Outer Fuel Content (Ci)	Maximum Payload (Ci)	Specie	Inner Fuel Content (Ci)	Outer Fuel Content (Ci)	Maximum Payload (Ci)
$^{90}\text{Sr}/^{90}\text{Y}$	65.7	104.2	358.9	^{152}Eu	0.01	0.02	0.06
^{99}Tc	0.01	0.03	0.07	$^{110\text{m}}\text{Ag}$	0.01	0.02	0.05
$^{106}\text{Ru}/^{106}\text{Rh}$	19.01	40.6	130.9	^{238}Pu	0.90	1.93	4.94
$^{113\text{m}}\text{Cd}$	0.036	0.08	0.20	^{239}Pu	1.06	2.26	5.80
$^{125\text{m}}\text{Te}$	1.186	2.53	6.50	^{240}Pu	0.75	1.61	4.11
^{134}Cs	10.62	22.7	58.1	^{241}Pu	85.24	181.9	465.7
$^{137}\text{Cs}/^{137\text{m}}\text{Ba}$	87.86	187.5	480.0	^{241}Am	0.82	1.75	4.48
$^{144}\text{Ce}/^{144}\text{Pr}$	21.4	45.7	116.9	^{244}Cm	0.27	0.57	1.45
^{147}Pm	65.6	139.9	358.2	^{55}Fe	0.31	0.67	1.71
^{151}Sm	0.78	1.66	4.26	^{60}Co	0.10	0.22	0.57
^{154}Eu	2.24	4.78	12.3	^{125}Sb	0.17	0.36	0.93
^{155}Eu	0.72	1.54	3.95	$^{125\text{m}}\text{Te}$	0.04	0.09	0.23
^{85}Kr	7.03	15.0	38.4	$^{119\text{m}}\text{Sn}$	0.02	0.04	0.09
^{125}Sb	4.86	10.4	26.6	^{63}Ni	0.02	0.04	0.09
^3H	0.40	0.85	2.17	^{235}U	1.18×10^{-4}	2.52×10^{-4}	6.46×10^{-4}
$^{144\text{m}}\text{Pr}$	0.26	0.55	1.40	Total	567	1,209	3,095

3.2.3 Non-Radioactive Materials

Throughout the twenty or more years that some of this fuel has been stored at the K Basins, it is possible that corrosion of the fuel combined with the hydrolysis of water has created UH_3 . This material has pyrophoric properties similar to that of metallic uranium. The SFEC shall be configured in a manner that will preserve the chemical integrity of the fuel samples to the greatest possible extent. For example, elements that have been stored in water shall be transported in SFECs that have been filled with water to minimize the possibility of a reaction between any UH_3 and the air within the CNS 1-13G cask.

4.0 TRANSPORT SYSTEM

4.1 TRANSPORTER

The CNS 1-13G will be transported on a low boy trailer specially equipped to handle the CNS 1-13G. The trailer will be supplied by ChemNuclear.

4.2 TIEDOWN SYSTEM

To secure the CNS 1-13G to the transport vehicle, the cask shall be placed onto the cask base which is directly bolted to the specially equipped trailer. The overpack shall then be bolted to the cask base. The packaging will also be secured to the transport vehicle through the use of a high strength strap or chain attached to the trailer. Part A, Section 9.3 contains a detailed description of the tiedown procedures.

4.3 SPECIAL TRANSFER REQUIREMENTS

4.3.1 Routing and Access Control

The transfer route shall be controlled to preclude public access during the transfer.

4.3.2 Radiological Limitations

The contact dose rate at any point on the surface of the package shall be less than 2 mSv/h (200 mrem/h). The dose rate from the outer lateral surfaces of the transfer vehicle shall be less than 0.10 mSv/h (10 mrem/h) at 2 m (6.6 ft). The dose rate at any normally occupied space in the transfer vehicle shall be less than 0.02 mSv/h (2 mrem/h).

Removable contamination on the external surfaces of the CNS 1-13G shall not exceed the following (*WHC Radiological Control Manual*, WHC-CM-1-6):

Nuclide (note 1)	Removable ₂ (dpm/100 cm ²) (note 2)	Total (Fixed + Removable) ₂ (dpm/100 cm ²) (note 3)
U-natural, ²³⁵ U, ²³⁸ U, and associated decay products.	220 alpha	5,000 alpha
Transuranics, ²²⁶ Ra, ²²⁸ Ra, ²³⁰ Th, ²²⁸ Th, ²³¹ Pa, ²²⁷ Ac, ¹²⁵ I, ¹²⁹ I	20	500
Th-nat, ²³² Th, ⁹⁰ Sr, ²²³ Ra, ²²⁴ Ra, ²³² U, ¹²⁶ I, ¹³¹ I, ¹³³ I	200	1,000
Beta-gamma emitters (nuclides with decay modes other than alpha emission or spontaneous fission) except ⁹⁰ Sr and others noted above. Includes mixed fission products containing ⁹⁰ Sr.	1,000 beta/gamma	5,000 beta/gamma
Table notes: 1. The values in this table apply to radioactive contamination deposited on, but not incorporated into the interior of the contaminated item. Where contamination by both alpha-and beta/gamma-emitting nuclides exists, the limits established for the alpha-and beta/gamma-emitting nuclides apply independently. 2. The amount of removable radioactive material per 100 cm ² of surface area should be determined by swiping the area with dry filter or soft absorbent paper, while applying moderate pressure, and then assessing the amount of radioactive material on the swipe with an appropriate instrument of known efficiency. For objects with a surface area less than 100 cm ² , the entire surface should be swiped, and the activity per unit area should be based on the actual surface area. Except for transuranic elements, ²²⁸ Ra, ²²⁷ Ac, ²²⁸ Th, ²³⁰ Th, ²³¹ Pa, and alpha emitters, it is not necessary to use swiping techniques to measure removable contamination levels if direct scan surveys indicate that the total residual contamination levels are below the values for removable contamination. 3. The levels may be averaged over 1 m ² provided the maximum activity in any area of 100 cm ² is less than three times the values in the table.		

4.3.3 Speed Limitations

During transport, the maximum speed of the transport vehicle shall be limited to the posted speed limit.

4.3.4 Mileage Limitations

For this shipping campaign, the CNS 1-13G shall not be transported more than a total of 1,100 miles with a payload during any rotating 12 month period. The transport of an empty CNS 1-13G shall not count against this mileage limit. The mileage shall be recorded by the driver and records of the mileage shall be retained in the transfer vehicle.

4.3.5 Environmental Conditions

Transport shall not take place under icy conditions or when visibility is reduced to less than $\frac{1}{2}$ mile by fog, rain, snow, dust or other phenomena.

To preclude the freezing of the water in the SFECs, the cask shall not be shipped when the ambient temperature is less than 32 °F. The cask shall not be shipped in icy conditions or in wet conditions below 35 °F.

4.3.6 Escort and Emergency Response

The shipper shall notify the Fire Department prior to any transfer. Escorts are required to precede and follow the transfer vehicle.

In the event of a fire accident, subsequent to the extinguishing of the fire, the CNS 1-13G cask shall be cooled by water, or whatever means are available until it can be established the temperature within the cask has been sufficiently lowered to preclude the boiling of water within the SFEC. Any runoff shall be monitored for radioactive material.

4.3.7 Transportation Time Limits

Prior to shipment, the gas generation within each of the SFECs shall be determined by measuring the pressure build up in a sealed SFEC. The time to load, transfer and unload the CNS 1-13G shall be limited so that the concentration of hydrogen within the cask will be less than 5% by volume with a factor of safety of two (2.5% H₂).

4.3.8 Handling Accidents

To allow for handling accidents, the SARP for the CNS 1-13G (ChemNuclear 1992) indicates that the cask may be dropped up to 2 ft without any decrease in its ability to safely transport the payload. The SFEC will also survive such a drop without degradation of its capabilities. In the unlikely event the cask is dropped greater than 2 ft or the integrity of the SFEC is suspect, the cask shall be returned to one of the K Basins or the PTL for offloading and payload inspection.

4.3.9 Exclusive Use

The transport vehicle shall not carry any other packagings containing hazardous material during transfer of the CNS 1-13G.

5.0 ACCEPTANCE OF PACKAGING FOR USAGE

5.1 PRE-USE INSPECTION

Pre-use inspections shall be performed before each use of a CNS 1-13G and/or the SFECs.

1. The user shall ensure that the packaging is in unimpaired physical condition before each use. No obvious flaws (i.e., broken welds, punctures, lid distortions) shall be present.
2. The user shall verify that:
 - All closure bolts are in place and in good condition.
 - That there are no fasteners with suspect headmarks.
 - That no damage to the bolts (e.g., missing or damaged threads) exists that could limit the structural integrity of the closures.
3. The gaskets shall be inspected to ensure that they are in good condition.
4. The CNS 1-13G shall meet all of the acceptance tests required by the CoC and SARP (ChemNuclear 1992).

6.0 OPERATING REQUIREMENTS

6.1 GENERAL REQUIREMENTS

1. The approved SEP shall be considered the controlling document for the shipment of irradiated N Reactor and SPR shipments in the CNS 1-13G. It shall be valid for one year after the date of its release.
2. All applicable instructions and procedures for onsite shipment of radioactive material shall comply with *Hazardous Material Packaging and Shipping*, WHC-CM-2-14.
3. Operational controls presented in the SEP shall take priority over similar requirements presented in other WHC manuals, except where the other WHC requirements are more restrictive.
4. The CNS 1-13G cask, overpack, the SFECs, their designs, leasing specification, transportation method, and SEP shall not be altered or revised without documented approval in accordance with WHC-CM-6-1, *Standard Engineering Practices*, and WHC-CM-2-14.
5. Written operating procedures shall be followed when packaging and transferring payloads in the CNS 1-13G. These procedures shall provide instructions to ensure that the container is being used in accordance with this SEP. These procedures shall be reviewed and approved by Packaging Safety Engineering.
7. Application of paint, plastic film, or similar surface coatings to cover surface contamination is not permitted. After loading the cask shall be decontaminated to the levels prescribed in Part A, Section 4.3.2.
8. Operating procedures shall require that the transfer of CNS 1-13G cask be performed in accordance with Part A, Section 4.0.

6.2 LOADING PACKAGE

6.2.1 Single Fuel Element Container Loading

The SFEC shall be loaded underwater at the K Basins. The loading shall be performed in accordance with established K Basin procedures. Each SFEC shall be limited to a maximum of one inner and one outer fuel element.

6.2.2 CNS 1-13G Loading

The loading procedures for the cask shall be in accordance with applicable sections of *Procedures, License and Safety Analysis Report for Chem-Nuclear Systems, Inc., CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()F* (ChemNuclear 1992). The loading procedures are reproduced in Part A, Section 9.3 with the permission of the author.

Prior to immersion in the K Basins for loading, operations shall verify that the SFEC rack is in place in the cask inner cavity. The CNS 1-13G shall be limited to a maximum of three SFECs with one inner and outer fuel element apiece. To maintain an adequate margin of safety with respect to criticality, the maximum number of fuel elements that are permitted for a single shipments is two inner and two outer elements or three outer elements.

The time and date of the closure of the CNS 1-13G cask shall be recorded. The CNS 1-13G must be transported within the time permitted to prevent the concentration of hydrogen within the cask from equaling or exceeding 2.5%.

6.3 OFFLOADING PACKAGE

6.3.1 CNS 1-13G Offloading

The offloading procedures for the cask shall be in accordance with applicable section of *Procedures, License and Safety Analysis Report for Chem-Nuclear Systems, Inc., CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()F* (ChemNuclear 1992). The loading procedures are duplicated and appear in Part A, Section 9.4 with the permission of the author.

6.3.2 Single Fuel Element Container Offloading

The SFECs shall be removed from the CNS 1-13G and shoring assembly in the PTL basin. They shall be handled in accordance with established PTL criteria.

6.4 EMPTY PACKAGE

To be transported as empty radioactive containers, the CNS 1-13G and SFECs must

- a. Have a surface dose level equal to or less than 0.005 mSv (0.5 mrem/h).
- b. Have smearable internal contamination not greater than 10^{-3} uCi/cm² for beta/gamma emitting radionuclides and 10^{-4} uCi/cm² for alpha emitting radionuclides.

- c. Be in unimpaired condition and securely closed so that there will be no leakage of radioactive material under normal conditions of transportation.

7.0 QUALITY ASSURANCE PROGRAM

7.1 INTRODUCTION

To establish a quality assurance plan for the CNS 1-13G a graded approach was used to define the safety class of both the system and individual components of the CNS 1-13G. This graded approach determines the impact of each component on the safety of the packaging and assigns appropriate controls on the component and packaging. The application of the Safety Class system is fully documented in the *Management Requirements and Procedures* (MRP) 5.46, WHC-CM-1-3. Quality Assurance instructions or plans shall be developed for the procurement, fabrication and inspection of the package components based on the assigned safety class (1) of the package. The *QA Program Plan for the Hazardous Materials Transportation and Packaging Program*, WHC-IP-0705 (WHC 1990), defines the WHC Quality Assurance and safety class implementation, respectively, for RAM shipping packages.

7.2 ORGANIZATION

The safety class system of categorizing a package and its individual components is designed to identify the consequence to personnel and the environment for a failure of the system or of an individual component. MRP 5.46, WHC-CM-1-3, and EP 1.4, WHC-CM-6-1, provide detailed descriptions of the safety class system and its implementation. For the purposes of applying safety class to a package containing radioactive material, the system will be considered to be the package.

7.2.1 Safety Class Definition

7.2.1.1 Safety Class 1. Components of a system or a system whose failure could result a significant impact on the health or safety of offsite persons. In the case of radioactive materials, the failure of a safety class 1 system/component could result in the an offsite receptor receiving an effective equivalent dose via inhalation equal to or greater than 5 mSv (500 mrem).

7.2.1.2 Safety Class 2. Components of a system or a system other than safety class 1 whose failure could result in a release of radioactive or hazardous material that could have a significant impact on onsite personnel. In the case of radioactive materials, the failure of a safety class 2 system/component could result in an onsite receptor receiving an effective equivalent dose via inhalation equal to or greater than 50 mSv (5 rem).

7.2.1.3 Safety Class 3. Components of a system or a system other than safety class 1 or 2 whose failure could result in a release of radioactive material or hazardous material that could have an impact on onsite workers. In the case of radioactive materials, the failure of a safety class 3

system/component could result in a release of radioactive materials that would cause an onsite receptor to receive an effective equivalent dose via inhalation less than 50 mSv (5 rem) cumulative effective dose equivalent (CEDE).

7.2.1.4 Nonsafety Class 4. Components of a system or a system whose failure will have no significant effect on safety, health or, environmental protection.

7.2.2 System Safety Class

The safety class analysis performed in support of the *Packaging Design Criteria for the N Reactor Fuel Characterization Shipments (WHC 1994b)* indicates that the maximum inhalation dose to an onsite receptor is 750 mSv (75 rem) CEDE and the maximum inhalation dose to an offsite receptor is 18 mSv (1.8 rem) CEDE, the CNS 1-13G cask is considered a safety class 1 system.

7.2.3 Component Class List

The components of the CNS 1-13G and their corresponding safety class are listed in Table A4-1.

Table A4-1. CNS 1-13G Cask Component Safety Class List.

Safety Class 1	CNS 1-13G cask body (including sides and bottom) Cask lid Cask lid gasket Cask lid bolts
Safety Class 2	None
Safety Class 3	SFEC body (including sides and bottom) SFEC lid SFEC gasket
Nonsafety Class 4	All parts not designated as being safety class 1 through 3

7.3 QUALITY ASSURANCE INSPECTION

The CNS 1-13G cask must pass the required inspections of Part A, Section 5.0 and the inspections required by the SARP and the CoC (ChemNuclear 1992).

8.0 MAINTENANCE

The CNS 1-13G will be leased from an offsite vendor, therefore maintenance is the responsibility of the vendor. The maintenance of the CNS 1-13G shall be current in terms of the CoC maintenance and the transport vehicle shall meet all applicable DOT/Federal Highway Administration (FHA) requirements.

9.0 APPENDIX

9.1 REFERENCES

ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()F*, ChemNuclear, Inc, Columbia, South Carolina.

WHC-CM-1-3, *Management Requirements and Procedures*, Westinghouse Hanford Company, Richland, Washington.

WHC-CM-2-14, *Hazardous Material Packaging and Shipping*, Westinghouse Hanford Company, Richland, Washington.

WHC, 1990, *QA Program Plan for the Hazardous Materials Transportation and Packaging Program*, WHC-IP-0705, Westinghouse Hanford Company, Richland, Washington.

WHC, 1994a, *Single Fuel Element Canister*, H-1-80394, Rev. 0, Westinghouse Hanford Company, Richland, Washington.

WHC, 1994b, *Packaging Design Criteria for the N Reactor/Single Pass Reactor Fuel Characterization Shipments*, WHC-SD-TP-PDC-022, Westinghouse Hanford Company, Richland, Washington.

9.2 DRAWINGS

9.2.1 CNS 1-13G Cask/Overpack Drawings

① TIE DOWN SHACKLE

HEX HD. BOLT
2.0-4.5 UNC-2A*4.5 LG.
ASTM A449 (OR EQUAL)
(10 REQD.)

SEAL (SEE DWG*)
110-B-06402-003 G3 OR G5 (UPPER)
110-B-06402-003 G4 (LOWER)
(LOCATION OPTIONAL)

OVERPACK
DWG.- C-110-B-06402-002

HEX HD. BOLT
REQD C-110-B-06402-003

LIFTING EAR

SHIPPING CASE BODY
DWG.- C-110-B-06402-003

LINER (ALL LINERS OPTIONAL)

C-110-B-06402-004 G1 (SHOWN)

C-110-B-06402-004 G2

BASE
DWG.- C-110-B-06402-002

ISOMETRIC VIEW
OF MODEL 1-13G PACKAGE

NOTE :

1. (DELETED - PER REV A)
2. BASE AND OVERPACK INSIDE & OUTSIDE ARE PAINTED WITH RUSTOLEUM GRAY OR EQUIV.
3. DIMENSIONS AND TOLERANCES INDICATE MAXIMUM VARIATIONS BETWEEN PACKAGES AND ARE NOT INDICATIVE OF "FIT" OR INTERCHANGEABILITY.
4. NOMINAL DIMENSIONS FOR STRUCTURAL SHAPES, PLATES, TUBES, ETC. WILL FALL WITHIN STANDARD MILL TOLERANCES.
5. ALL OTHER DIMENSIONS ARE ± 0.50 UNLESS SHOWN OTHERWISE.
6. PACKAGING WT = 25,500 lbs (+ or - 1000 lbs) WITHOUT LINER
30,650 lbs (+ or - 1000 lbs) WITH LINER
7. THE PACKAGE LIFTING EARS ARE BOLTED CLOSED DURING TRANSPORT WITH A STL. FLAP WHICH IS MARKED TO DENOTE THAT THEY ARE NOT TO BE USED FOR TIE-DOWN OR TO LIFT THE ENTIRE PACKAGE. (C-110-B-06402-002) P5

A PER ECD# RA
C-110-B-06402-001-001 4/20/88

TOLERANCE UNLESS OTHERWISE NOTED		ORDERED BY <u>4-23-87</u>	DATE <u>4-23-87</u>	BY <u>MAA/Baj</u>
PLAC	DEC	APPROVED BY <u>MAA/Baj</u>	SCALE	
CHEM-NUCLEAR SYSTEMS, INC.		TITLE 1-13G SHIPPING CONTAINER		
C-110-B-06402-001		C-110-B-06402-001		A

A PER ECD IF
C-110-B-06402-002-001

4-23-57	4-23-57	4-23-57	4-23-57
DESIGNED BY	CHECKED BY	APPROVED BY	DATE
1-136	1-136	1-136	1-136

CHEM-NUCLEAR
SYSTEMS, INC.

1-136

OVERPACK

C-110-B-06402-002-001

WELDING

DE PLATE
304 SST.

ITEM PLATE
304 SST.

ATE

DING

ELL
1.5%

VITY SHELL
SST.

SHELL BOTTOM
x 1/2 THK.

BOTTOM PLATE
104 SST.

WELD
1/8" V. (6)
1/8" THK
304 SST
DIMS SHOWN

ITEM (20)
TO
ITEM (22) SEAL (TYP) 1/8" PT
(6)
(7)
(14) LOCATING PINS (2)

(19) ATTACHMENT BOLTS
1.0-B WHT 2A x 2.75 LG
ASTM A-193-B6

(20) LIFTING EARS

ITEM (19) SEAL (TYP) 1/8" PT
TO
ITEM (20)
(6)
(7)

TOP VIEW OF CASK

(G5) SEAL-UPPER (OPTIONAL)
1/4" THK. SILICONE RUBBER
30-60 DUROMETER

LIFTING EARS
304 SST. STL. 1" THICK

VACUUM CONNECTION

C	PER ECD #	6-110-B-06402-003-03
B	PER ECD #	6-110-B-06402-003-02
A	PER ECD #	6-110-B-06402-003-01

DESIGNED BY	DATE	APPROVED BY	DATE
WHC-SD-TP-SEP-028	11/27/07	WHC-SD-TP-SEP-028	11/27/07
CHEM-NUCLEAR SYSTEMS, INC.		2-136 SHIPPING CASK	
C-110-B-06402-003		C	

9.2.2 Single Element Fuel Canister Drawings

(UNLESS OTHERWISE SPECIFIED)

(UNLESS OTHERWISE SPECIFIED)

7. EACH CANISTER SHALL HAVE A UNIQUE IDENTIFICATION NUMBER. THE IDENTIFICATION NUMBER SHALL BE STAMPED ON COLLAR (PT 91 AS INDICATED ON SH-2. STAMPED CHARACTER SIZE SHALL BE 3/8" MAXIMUM. THE IDENTIFICATION NUMBER SHALL CONSIST OF THE MONTH AND YEAR OF MANUFACTURE (AS A PREFIX), FOLLOWED BY A SEQUENTIAL SERIAL NUMBER.

IN ADDITION, THE SEQUENTIAL SERIAL NUMBER SHALL BE APPLIED, USING HIGH CONTRAST PAINT, TO THE TOP SURFACE OF ONE OF THE LUGS (PT 8). AS SHOWN ON SHEET 3. PAINTED NUMBER SIZE SHALL BE AS LARGE AS POSSIBLE.

8. ALL WELDS SHALL BE PERFORMED IN ACCORDANCE WITH ASME B3.1PVC AND WELDERS SHALL BE QUALIFIED PER ASME SECTION VIII.
9. ESTIMATED WEIGHT OF CANISTER ASSEMBLY IS 18 POUNDS EMPTY, 45 POUNDS WET LOADED WITH AN INNER FUEL ELEMENT, 60 POUNDS WET LOADED WITH AN OUTER FUEL ELEMENT, AND 77 POUNDS WET LOADED WITH AN INNER AND AN OUTER FUEL ELEMENT.
10. BEFORE ASSEMBLY, LUBRICATE INTERNAL THREADS ON LOCKING BAR (ITEM 12), AND EXTERNAL THREADS AND NOSE ON LOCKING BAR SCREW (ITEM 13) WITH "NEVER SEEZ" NUCLEAR GRADE NICKEL BASED LUBRICANT.
11. LEAK TEST: THE CANISTER ASSEMBLY SHALL BE HYDRO TESTED TO 50-52 PSI WITH NO DETECTABLE LEAKS.
12. MATERIAL CERTIFICATION REQUIREMENTS: ITEMS 7-17 AND 22 SHALL BE PROVIDED WITH CERTIFIED MATERIAL TEST REPORTS.
13. WELD TO BE MADE AFTER PARTS 12 & 13 ARE IN PLACE.
14. APPLY A THIN COAT OF "LOCKTITE RC 830" TO THREADED FASTNER PRIOR TO ASSEMBLY.

15 LIFTING BAIL
SCALE: 1/1

SCALE: VI

FOR GENERAL NOTES & PARTS LIST SEE SH-1

[illegible]

SINGLE FUEL ELEMENT CANISTER
ASSEMBLY

7	H-1-80394	26	0
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A9-8

VIEW 0-0

VIEW C-C

8 LUG
SCALE:

SCALE: VI

NOTE: PART MAY BE MADE FROM
6" OD X 1/2 WALL TUBING
ASTM A511 304 SST

FOR GENERAL NOTES & PARTS LIST SEE SH-1

[illegible]

13 LOCKING BAR SCREW
SCALE: 1 1/2 X

2000

[illegible]

F

E

D

C

B

A

1 SUPPORT RACK ARRANGEMENT
SCALE: 1/2

1	2	3	4	5	6	7	8	9	10
11	12	13	14	15	16	17	18	19	20

REV	DATE	DESCRIPTION	BY	CHKD	QTY	UNIT	PRICE	TOTAL	REMARKS
1	10/10/2018	REPAIR WORK			1	NO	1000	1000	
2	10/10/2018	REPAIR WORK			1	NO	1000	1000	
3	10/10/2018	REPAIR WORK			1	NO	1000	1000	
4	10/10/2018	REPAIR WORK			1	NO	1000	1000	
5	10/10/2018	REPAIR WORK			1	NO	1000	1000	
6	10/10/2018	REPAIR WORK			1	NO	1000	1000	
7	10/10/2018	REPAIR WORK			1	NO	1000	1000	
8	10/10/2018	REPAIR WORK			1	NO	1000	1000	
9	10/10/2018	REPAIR WORK			1	NO	1000	1000	
10	10/10/2018	REPAIR WORK			1	NO	1000	1000	
11	10/10/2018	REPAIR WORK			1	NO	1000	1000	
12	10/10/2018	REPAIR WORK			1	NO	1000	1000	
13	10/10/2018	REPAIR WORK			1	NO	1000	1000	
14	10/10/2018	REPAIR WORK			1	NO	1000	1000	
15	10/10/2018	REPAIR WORK			1	NO	1000	1000	

GENERAL NOTES

UNRECORDED COPY

- 1 DIMENSIONS ARE IN INCHES
TOLERANCES FRACTIONAL - .012
DECIMALS - .XX - .XX + .01
ANGLES - .0° - .0° 30'
- 2 ALL MACHINED SURFACES $\sqrt{125}$ IN ACCORDANCE WITH AMS 9451
LATEST REVISION.

3. BREAK ALL SHARP EDGES REMOVE ALL BURRS
4. ALL CORNER AND FILLET RADIUS 0.03 MAX
5. MATERIAL THICKNESS SHALL BE NOMINAL INDUSTRY STANDARDS
6. VENDOR MAY APPLY IDENTIFICATION NUMBERS TO PARTS FOR PROCESSING AND INSPECTION USE PROVIDED APPLICATION METHOD DOES NOT IMPAIR FUNCTION OF PART. VENDOR SHALL SIGHT METHOD TO BE USED FOR IDENTIFICATION MARKING TO BUYER FOR APPROVAL
7. ALL WELDS SHALL BE PERFORMED IN ACCORDANCE WITH ASME SECTION VIII SHALL BE QUALIFIED PER ASME SECTION VIII
8. ESTIMATED WEIGHT OF ASSEMBLY WITHOUT FUEL ELEMENT CANNISTERS IS APPROXIMATELY 77 POUNDS

5. SPACER THICKNESS TO BE DETERMINED BY FIELD.

[illegible]

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2 SUPPORT RACK ASSEMBLY
SCALE: V4

PLATE, TOP

SCALE- V4

7 PLATE, MIDDLE / LOWER
SCALE: 1/4

SCALE: V4

② TUBE, UPPER

SCALE 1/4

② TUBE, LOWER

SCALE 1/4

FOR GENERAL NOTES & PARTS LIST SEE SH-1

[illegible]

9.3 CNS 1-13G LOADING PROCEDURE

4.2.2 Lifting and unloading hardware

4.2.3 Tools

4.2.3.1 0-150 ft.-lbs. and 150-600 ft.-lbs. torque wrenches

4.2.3.2 Proper size sockets

4.2.4 Lifting sling compatible with cask lid, overpack, and loaded cask

4.2.5 Acceptable bolt lubricant (Moly-Z, Neolube, or AntiSeize)

4.2.6 Health physics (HP) instrumentation and support materials.

4.3 Prerequisites

Not applicable.

4.4 Acceptance Criteria

Not applicable.

5.0 HANDLING PRECAUTIONS

5.1 Treat the surfaces of the cask, the internal surface of the overpack, the bottom of the cask lid, and any material removed as contaminated.

5.2 Lift the cask by using the two (2) lifting ears on the side of the cask. DO NOT lift the cask by the lid lifting lug. DO NOT use the rectangular slots on the lifting ears for lifting.

5.3 Secure the cask and overpack to the trailer by the tie-down lugs attached to the side of the overpack. Tie-down using the rectangular slots on the top of the overpack is prohibited.

5.4 Survey the cask cavity for radiation and contamination levels after removing any contents. Decontaminate as required by the health physics technicians after the contents have been removed.

5.5 Remove any liquid from the cask cavity. Treat this liquid as radioactive waste.

5.6 Visually inspect the cask for damage to the lid, gasket, gasket seating surfaces, lifting lugs, lifting slings, or tie-downs.

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- 5.7 To prevent personal injury and cask damage, ensure that freestanding ladders are secured and attended by personnel. Do this before climbing onto the cask.
- 5.8 The cask may be thermally hot. Avoid placing it in contact with any materials that may melt at $\sim 200^{\circ}\text{F}$.
- 5.9 When lowering a liner or a drum into the cask, be alert to the possibility of airborne contamination from the air escaping the cask.
- 5.10 Before the cask leaves the facility, the following shall be confirmed:
 - 5.10.1 That any external lifting lugs are properly covered for transport.
 - 5.10.2 That trailer placarding and cask labelling meet DOT Specifications (CFR Title 49, Part 172).
 - 5.10.3 That exterior radiation levels do not exceed 200 mR/hour on contact, 10 mR/hour at 2 meters, and 2 mR/hour in the tractor cab, in accordance with 49 CFR 173.441.
 - 5.10.4 That the outer package is sealed with anti-tamper seals.
 - 5.10.5 That the drain plug is securely installed.
- 5.11 Packagings containing quantities of radioactive material in excess of Type A quantities specified in 10 CFR 20.205(b) shall be received, monitored, and handled by the licensee receiving the package in accordance with the requirements of 10 CFR 20.205, as applicable.

6.0 LOADING PROCEDURE

CAUTION: TREAT THE INSIDE SURFACES OF THE OVERPACK AND THE LINER OR DRUM IN THE CASK AS CONTAMINATED.

- 6.1 Prepare to remove the overpack.
 - 6.1.1 Remove the seal wires from the ends of the two (2) 1-1/2-inch red-painted bolts on the side of the overpack.
 - 6.1.2 Remove the two (2) 1-1/2-inch bolts from the sides of the overpack.
 - 6.1.3 Remove the ten (10) 2-inch bolts from the overpack baseplate.
 - 6.1.4 Attach the crane hook to the lifting sling.

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6.1.5 Attach the lifting sling to the lifting lugs attached to the sides of the overpack.

6.1.6 Remove the overpack and place it on absorbent material or plastic sheeting. Place it carefully on a level surface to avoid damaging the seating surface of the overpack.

6.1.7 Remove the crane hook.

6.2 Prepare to load the cask.

NOTE: THIS CASK MAY BE LOADED UNDERWATER. IF THIS IS THE CASE, FOLLOW THE PROCEDURE FOR WET LOADING, SECTION 6.2.2.

CAUTION: TREAT THE UNDERSIDE OF THE LID, THE SURFACES OF THE CASK, AND ANY BOLTS OR SEALS REMOVED AS CONTAMINATED.

NOTE: THIS CASK MAY BE LOADED WITH EITHER A 55-GALLON DRUM OR A LINER. AN APPROPRIATE LIFTING DEVICE MUST BE ATTACHED TO THE DRUM OR THE LINER.

6.2.1 Procedure for dry loading.

CAUTION: DO NOT LIFT THE CASK BY THE LIFTING LUG ON THE CASK LID.

KEEP CRANE CABLES VERTICAL AT ALL TIMES TO AVOID LATERAL MOVEMENT OF THE CASK.

6.2.1.1 Prepare to remove the cask from the trailer, if necessary.

NOTE: DO NOT USE THE RECTANGULAR SLOTS ON THE LIFTING EARS FOR LIFTING.

THE CASK SHALL BE REPLACED ON THE TRAILER IN THE SAME CONFIGURATION AS RECEIVED.

6.2.1.1.1 Attach lifting slings with shackles to the round lifting holes on the cask lifting ears (see Figure 1).

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6.2.1.1.2 Attach the crane hook to the lifting sling.

6.2.1.1.3 Move the cask to a level loading area.

6.2.1.1.4 Remove the crane hook.

6.2.1.2 Prepare to remove the cask lid.

6.2.1.2.1 Remove the six (6) 1-inch bolts from the cask lid.

6.2.1.2.2 Attach the crane hook to the lid lifting lug or a sling attached to the lid lifting lug.

6.2.1.2.3 Lift the lid straight up off of the cask and place it on absorbent material or plastic sheeting.

CAUTION: BE ALERT THAT THE CASK LID GASKET MAY LIFT OFF WITH THE CASK LID.

6.2.1.2.4 Remove the crane hook.

NOTE: IF CNS 2-INCH LEAD SHIELD IS INSTALLED IN CASK, REFER TO SECTION 10.0 AND 10.1 FOR INSTRUCTIONS.

6.2.1.3 Prepare to remove the liner or the drum from the cask, if necessary.

6.2.1.3.1 Attach the crane hook to the attachment on the liner or the drum.

6.2.1.3.2 Lift out the liner or drum and place it on absorbent material or plastic sheeting.

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6.2.1.3.3 Remove the crane hook.

6.2.1.3.4 Visually inspect the cask cavity to verify integrity. Remove any liquids or foreign materials. Visually inspect the welds around cask bottom to verify integrity. If any defects are noted, contact Chem-Nuclear's Transportation Division.

6.2.1.4 Fill the liner or the drum if necessary.

6.2.1.5 Prepare to load the liner or drum into the cask, if necessary.

CAUTION: AS THE LINER OR DRUM IS LOWERED INTO THE CASK, AIR IN THE CASK WILL CUSHION AND SLOW THE LINER OR DRUM. BE ALERT TO THE POSSIBILITY OF AIRBORNE CONTAMINATION FROM THE AIR ESCAPING THE CASK DURING THIS PROCEDURE.

NOTE: CLEAN THE LINER OR DRUM BEFORE PLACING IT IN THE CASK. TREAT DEBRIS REMOVED AS CONTAMINATED.

6.2.1.5.1 Attach the crane hook to the attachment on the liner or the drum.

6.2.1.5.2 Lower the liner or drum straight into the cask.

6.2.1.5.3 Remove the crane hook.

6.2.1.5.4 Ensure that the liner lift attachment swings clear to allow proper installation of the cask lid.

NOTE: SHORE THE LINER OR DRUM IF NECESSARY TO PREVENT MOVEMENT DURING

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ING NORMAL
TRANSPORT
CONDITIONS.

6.2.1.6 Prepare to replace the cask lid.

NOTE: IF CNS 2-INCH LEAD SHIELD IS
INSTALLED IN CASK, REFER TO
SECTION 10.0 AND 10.2 FOR
INSTRUCTIONS.

6.2.1.6.1 Visually inspect the lid gasket for
cracks or tears. Replace the
gasket if it is damaged. Inspect
and clean the gasket seating
surfaces.

6.2.1.6.2 Attach the crane hook to the lid
lifting lug or sling attached to the
lid lifting lug.

NOTE: CLEAN THE
BOTTOM SURFACE
OF THE LID,
TREATING ALL
DEBRIS REMOVED
AS CON-
TAMINATED.

6.2.1.6.3 Position the lid over the cask using
the alignment pin.

6.2.1.6.4 Lower the lid onto the cask.

NOTE: VISUALLY
INSPECT THE
CASK LID
CLOSURE BOLTS
AND DRAIN PLUG
(IF REMOVED) FOR
SIGNS OF
CRACKING OR
OTHER VISUAL
SIGNS OF
DEFECTS.
REPLACE ANY
DEFECTIVE
MATERIALS.

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- 6.2.1.6.5 Replace the six (6) 1-inch bolts on the cask lid. Torque to 220 ± 22 ft.-lbs. (160 ± 16 ft.-lbs. if lubricated), using a star pattern.

NOTE: TIGHTEN ALL THE
B O L T S
H A N D - T I G H T
B E F O R E S T A R T I N G
T H E T O R Q U E
S E Q U E N C E.

- 6.2.1.6.6 Remove the crane hook.

- 6.2.1.6.7 Refer to Sections 6.3 and 6.4 for replacing the cask on the trailer and replacing the overpack.

6.2.2 Procedure for wet loading.

CAUTION: DO NOT LIFT THE CASK BY THE LIFTING LUG ON THE CASK LID.

KEEP CRANE CABLES VERTICAL AT ALL TIMES TO AVOID LATERAL MOVEMENT OF THE CASK.

- 6.2.2.1 Prepare to remove the cask from the trailer.

NOTE: DO NOT USE THE RECTANGULAR SLOTS ON THE LIFTING EARS FOR LIFTING.

THE CASK SHALL BE REPLACED ON THE TRAILER IN THE SAME CONFIGURATION AS RECEIVED.

- 6.2.2.1.1 Attach lifting slings with shackles to the round lifting holes on the cask lifting ears (see Figure 1).

- 6.2.2.1.2 Attach the crane hook to the lifting sling.

- 6.2.2.1.3 Move the cask to a level setdown area.

- 6.2.2.1.4 Remove the crane hook.

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6.2.2.2 Prepare to remove the cask lid.

6.2.2.2.1 Remove the six (6) 1-inch bolts from the cask lid.

6.2.2.2.2 Attach a sling to the lid lifting lug.

6.2.2.2.3 Attach the crane hook to the lifting sling.

6.2.2.2.4 Lift the lid straight up off of the cask and place it on absorbent material or plastic sheeting.

CAUTION: BE ALERT THAT THE CASK LID GASKET MAY LIFT OFF WITH THE CASK LID.

6.2.2.2.5 Remove the crane hook.

NOTE: IF CNS 2-INCH LEAD SHIELD IS INSTALLED IN CASK, REFER TO SECTION 10.0 AND 10.1 FOR INSTRUCTIONS.

6.2.2.2.6 Visually inspect the cask cavity to verify integrity. Remove any liquids or foreign materials. Visually inspect the weld around cask bottom to verify integrity. If any defects are noted, contact Chem-Nuclear's Transportation Division.

6.2.2.3 Prepare the cask for lowering into the pool.

NOTE: IF BACKUP LIFT STRUCTURE (REDUNDANT SYSTEM) IS TO BE USED, REFER TO SECTION 9.0 FOR INSTRUCTIONS.

NOTE: TREAT ANY LIQUID FROM THE DRAIN AS CONTAMINATED.

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IF THE SLING IS TO BE DISCONNECTED FROM THE CRANE HOOK, CONFIRM THAT THE SLING HAS REMOTE CONNECTING/DISCONNECTING CAPABILITY.

- 6.2.2.3.1 Remove the drain plug from the cask, clean the threads, and retain for re-installation.
- 6.2.2.3.2 Attach the lifting sling to the round holes on the cask lifting ears, if necessary.
- 6.2.2.3.3 Attach the crane hook to the lifting sling.
- 6.2.2.3.4 Lift the cask with the crane.
- 6.2.2.3.5 Position the cask over the pool.
- 6.2.2.3.6 To ensure proper operation of the cask drain line, add demineralized water to cask cavity or lower cask into pool to fill with water and raise cask clear of pool surface. Water should flow freely from the cask drain line. If water does not flow freely, inspect drain line for obstructions.
- 6.2.2.3.7 Lower the cask into the pool to a sufficient depth to ensure safe operating conditions for all personnel.
- 6.2.2.3.8 Detach the crane hook from the lifting sling, if necessary. Confirm that the cask is resting securely in the pool before disconnecting.

6.2.2.4 Fill the liner, if necessary.

6.2.2.5 Load a filled liner into the cask, if necessary.

NOTE: IF CNS 2-INCH LEAD SHIELD IS INSTALLED IN CASK, REFER TO

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SECTION 10.0 AND 10.2 FOR
INSTRUCTIONS.

- 6.2.2.6 Prepare to replace the cask lid while the cask is in the pool.
- 6.2.2.6.1 Visually inspect the lid gasket for cracks or tears. Replace the gasket if it is damaged. Inspect and clean the gasket seating surfaces.
- 6.2.2.6.2 Attach the crane hook to the lid lifting sling.
- 6.2.2.6.3 Position the lid over the cask using the alignment pin.
- 6.2.2.6.4 Lower the lid onto the cask.
- 6.2.2.6.5 Remove the crane hook from the lid lifting sling.
- 6.2.2.7 Prepare to remove the loaded cask from the pool.
- 6.2.2.7.1 Attach the crane to the cask lifting sling.
- 6.2.2.7.2 Lift the cask out of the pool. Allow the cask to remain over the pool until all the water has drained from the cask.
- 6.2.2.7.3 Replace the cask drain plug. Ensure that the drain plug does not leak.
- 6.2.2.7.4 Move the cask to a set-down area.

NOTE: VISUALLY
INSPECT THE
CASK LID
CLOSURE BOLTS
AND DRAIN PLUG
(IF REMOVED) FOR
SIGNS OF CRACK-
ING OR OTHER

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VISUAL SIGNS OF DEFECTS. REPLACE ANY DEFECTIVE MATERIALS.

- 6.2.2.7.5 Replace the six (6) 1-inch bolts on the cask lid. Torque to 220 ± 22 ft.-lbs. (160 ± 16 ft.-lbs. if lubricated), using a star pattern.

NOTE: TIGHTEN ALL THE BOLTS HAND-TIGHT BEFORE STARTING THE TORQUE SEQUENCE.

- 6.2.2.7.6 Decontaminate all external surfaces of the cask.

- 6.2.2.7.7 Remove the crane hook, if necessary.

- 6.2.2.7.8 Remove the lifting sling from the cask lid.

- 6.3 Prepare to replace the cask on the trailer, if necessary.

NOTE: REFER TO SECTION 9.0 FOR REMOVAL OF BACKUP LIFT STRUCTURE (REDUNDANT SYSTEM), IF USED.

NOTE: DO NOT USE THE RECTANGULAR SLOTS ON THE CASK LIFTING EARS FOR LIFTING.

THE CASK SHALL BE REPLACED ON THE TRAILER IN THE SAME CONFIGURATION AS RECEIVED.

- 6.3.1 Attach the crane hook to the cask lifting sling, if necessary.
- 6.3.2 Place the cask on the trailer, confirming that the cask is positioned in the same configuration as received.
- 6.3.3 Remove the crane hook from the cask lifting sling.
- 6.3.4 Detach the lifting sling from the cask lifting ears.

- 6.4 Prepare to replace the overpack.

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NOTE: TIE-DOWN USING THE RECTANGULAR SLOTS ON THE TOP OF THE OVERPACK IS PROHIBITED.

6.4.1 Attach the lifting sling to the lugs on the sides of the overpack.

6.4.2 Attach the crane hook to the lifting sling.

6.4.3 Lift the overpack into position over the cask.

NOTE: ENSURE THAT THE ALIGNMENT PIN IS IN POSITION.

6.4.4 Lower the overpack onto the baseplate and position the overpack.

6.4.5 Remove the crane hook and the lifting sling.

NOTE: VISUALLY INSPECT ALL OVERPACK BOLTS FOR SIGNS OF CRACKING OR OTHER VISUAL SIGNS OF DEFECTS. REPLACE ANY DEFECTIVE MATERIALS.

6.4.6 Replace the ten (10) 2-inch bolts in the baseplate. Torque bolts to 400 ± 40 ft.-lbs. (300 ± 30 ft.-lbs., if lubricated).

6.4.7 Replace and tighten the two (2) 1-1/2-inch bolts on the sides of the overpack. Torque to 300 ± 30 ft.-lbs. (220 ± 22 ft.-lbs. if lubricated).

6.4.8 Place lead wire seals through the designated 1-1/2-inch bolts on the sides of the overpack.

6.4.9 Attach the tie-down cables/chains to the tie-down lugs on the sides of the overpack and tighten the binders on the cables/chains.

6.5 Before the cask leaves the facility, the following shall be confirmed:

6.5.1 That any external lifting lugs are properly covered for transport.

6.5.2 That trailer placarding and cask labelling meet DOT Specifications (CFR Title 49, Part 172).

6.5.3 That exterior radiation levels do not exceed 200 mR/hr on contact, 10 mR/hr at 2 meters, and 2 mR/hr in the tractor cab, in accordance with 49 CFR 173.441.

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6.5.4 That the outer package is sealed with anti-tamper seals.

6.5.5 That the drain plug is securely installed.

6.6 Complete the USER CHECK-OFF SHEET before the cask leaves the facility and send a copy along with the shipment.

7.0 UNLOADING PROCEDURE

NOTE: ALL PERSONS HANDLING A FILLED LINER OR DRUM SHALL OBSERVE ALL ESTABLISHED SITE RADIATION PROTECTION PROCEDURES.

7.1 Position the unloading crane at an optimum distance to facilitate offloading and to minimize operator exposure.

7.2 Prepare to remove the overpack.

CAUTION: TREAT THE INSIDE SURFACES OF THE OVERPACK AS CONTAMINATED.

7.2.1 Remove the seal wires from the ends of the two (2) 1-1/2-inch red painted bolts on the sides of the overpack.

7.2.2 Remove the two (2) 1-1/2-inch bolts on the side of the overpack.

7.2.3 Remove the ten (10) 2-inch bolts from the overpack baseplate.

7.2.4 Attach the crane hook to the lifting sling.

7.2.5 Attach the lifting sling to the lifting lugs on the sides of the overpack.

7.2.6 Remove the overpack and place it on absorbent material or plastic sheeting. Place carefully on a level surface to avoid damaging the seating surface of the overpack.

7.2.7 Remove the crane hook.

7.3 Prepare to remove the cask lid.

CAUTION: TREAT THE UNDERSIDE OF THE LID, THE SURFACES OF THE CASK, AND BOLTS OR SEALS REMOVED AS CONTAMINATED.

NOTE: THIS CASK MAY BE LOADED WITH EITHER A 55-GALLON DRUM OR A LINER. AN APPROPRIATE LIFTING DEVICE

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9.4 CNS 1-13G OFFLOADING PROCEDURE

6.5.4 That the outer package is sealed with anti-tamper seals.

6.5.5 That the drain plug is securely installed.

6.6 Complete the USER CHECK-OFF SHEET before the cask leaves the facility and send a copy along with the shipment.

7.0 UNLOADING PROCEDURE

NOTE: ALL PERSONS HANDLING A FILLED LINER OR DRUM SHALL OBSERVE ALL ESTABLISHED SITE RADIATION PROTECTION PROCEDURES.

7.1 Position the unloading crane at an optimum distance to facilitate offloading and to minimize operator exposure.

7.2 Prepare to remove the overpack.

CAUTION: TREAT THE INSIDE SURFACES OF THE OVERPACK AS CONTAMINATED.

7.2.1 Remove the seal wires from the ends of the two (2) 1-1/2-inch red painted bolts on the sides of the overpack.

7.2.2 Remove the two (2) 1-1/2-inch bolts on the side of the overpack.

7.2.3 Remove the ten (10) 2-inch bolts from the overpack baseplate.

7.2.4 Attach the crane hook to the lifting sling.

7.2.5 Attach the lifting sling to the lifting lugs on the sides of the overpack.

7.2.6 Remove the overpack and place it on absorbent material or plastic sheeting. Place carefully on a level surface to avoid damaging the seating surface of the overpack.

7.2.7 Remove the crane hook.

7.3 Prepare to remove the cask lid.

CAUTION: TREAT THE UNDERSIDE OF THE LID, THE SURFACES OF THE CASK, AND BOLTS OR SEALS REMOVED AS CONTAMINATED.

NOTE: THIS CASK MAY BE LOADED WITH EITHER A 55-GALLON DRUM OR A LINER. AN APPROPRIATE LIFTING DEVICE

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MUST BE ATTACHED TO THE DRUM OR THE LINER.

CAUTION: DO NOT LIFT THE CASK BY THE LIFTING LUG ON THE CASK LID. KEEP CRANE CABLES VERTICAL AT ALL TIMES TO AVOID LATERAL MOVEMENT OF THE CASK.

POSITION THE CASK ON A LEVEL SURFACE TO FACILITATE LIFTING THE LINER STRAIGHT UP OUT OF THE CASK.

- 7.3.1 Remove the six (6) 1-inch bolts from the lid.
- 7.3.2 Attach the crane hook to the lid lifting lug or a sling attached to the lid lifting lug.
- 7.3.3 Lift the lid straight up off of the cask and place it on absorbent material or plastic sheeting.

CAUTION: BE ALERT THAT THE CASK LID GASKET MAY LIFT OFF WITH THE CASK LID.

- 7.3.4 Remove the crane hook.

NOTE: IF CNS 2-INCH LEAD SHIELD IS INSTALLED, REFER TO SECTION 10.0 AND 10.1 FOR INSTRUCTIONS.

- 7.4 The health physics technician shall conduct a radiation and contamination survey to determine offloading precautions.
- 7.5 As the health physics technician directs, vacate all persons from the area except for the crane operator and a rigger. The rigger shall stand in clear view of the crane operator.
- 7.6 Prepare to remove the contents of the cask.
 - 7.6.1 Attach crane hook to the attachment on the liner or the drum.
 - 7.6.2 Lift the liner or drum straight up out of the cask and allow any liquid to drip off.
 - 7.6.3 Place the liner or drum in position for disposal or future handling.
 - 7.6.4 Detach the crane hook from the attachment on the liner or the drum.

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- 7.7 The health physics technician shall survey the interior of the cask and overpack for radiation and contamination levels. Decontaminate if acceptable levels are exceeded.

CAUTION: TREAT ANY LIQUID IN THE CASK OR USED IN THE DECONTAMINATION PROCESS AS CONTAMINATED.

- 7.8 Visually inspect the inside of the cask and overpack for damage or for liquid accumulation. Remove any liquid. If the inside of the cask is damaged, remove the cask from service.

- 7.9 Place a new liner or drum in the cask, if required.

NOTE: CLEAN THE LINER OR DRUM BEFORE PLACING IT INTO THE CASK. TREAT DEBRIS REMOVED AS CONTAMINATED.

- 7.9.1 Attach the crane hook to the attachment on the liner or the drum.

- 7.9.2 Carefully lower the liner or drum into the cask and remove the crane hook. DO NOT damage the gasket seating surfaces, the sides of the cask, or inner walls.

- 7.9.3 Ensure that the liner lift attachment swings clear to allow proper installation of the cask lid.

- 7.10 Prepare to replace the cask lid.

NOTE: IF CNS 2-INCH SHIELD IS INSTALLED, REFER TO SECTION 10.0 AND 10.2 FOR INSTRUCTIONS.

- 7.10.1 Visually inspect the lid gasket for cracks or tears. Replace the gasket if it is damaged. Inspect and clean the gasket seating surfaces.

- 7.10.2 Attach the crane hook to the lifting lug on the cask lid or a sling attached to the lid lifting lug.

NOTE: CLEAN THE BOTTOM SURFACE OF THE LID, TREATING ALL DEBRIS REMOVED AS CONTAMINATED.

- 7.10.3 Position the lid over the cask using the alignment pin.

- 7.10.4 Lower the lid onto the cask.

NOTE: VISUALLY INSPECT THE CASK LID CLOSURE

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BOLTS AND DRAIN PLUG (IF REMOVED) FOR SIGNS OF CRACKING OR OTHER VISUAL SIGNS OF DEFECTS. REPLACE ANY DEFECTIVE MATERIALS.

- 7.10.5 Replace the six (6) 1-inch bolts on the cask lid. Torque to 220 ± 22 ft.-lbs. (160 ± 16 ft.-lbs. if lubricated), using a star pattern.

NOTE: TIGHTEN ALL THE BOLTS HAND-TIGHT BEFORE STARTING THE TORQUE SEQUENCE.

- 7.10.6 Remove the crane hook.

- 7.10.7 Visually inspect the weld around cask bottom to verify integrity. If any defects are noted, contact Chem-Nuclear Systems' Transportation Division.

- 7.11 Prepare to replace the overpack.

NOTE: TIE-DOWN USING THE RECTANGULAR SLOTS ON THE TOP OF THE OVERPACK IS PROHIBITED.

- 7.11.1 Attach the lifting sling to the lugs on the sides of the overpack.

- 7.11.2 Attach the crane hook to the lifting sling.

- 7.11.3 Lift the overpack into position over the cask.

NOTE: ENSURE THE ALIGNMENT PIN IS IN POSITION.

- 7.11.4 Lower the overpack onto the baseplate and position the overpack.

- 7.11.5 Remove the crane hook and lifting sling.

NOTE: VISUALLY INSPECT ALL OVERPACK BOLTS FOR SIGNS OF CRACKING OR OTHER VISUAL SIGNS OF DEFECTS. REPLACE ANY DEFECTIVE MATERIALS.

- 7.11.6 Replace the ten (10) 2-inch bolts in the baseplate. Torque bolts to 400 ± 40 ft.-lbs. (300 ± 30 ft.-lbs., if lubricated).

- 7.11.7 Replace the two (2) 1-1/2-inch bolts in the sides of the overpack. Torque them to 300 ± 30 ft.-lbs. (220 ± 22 ft.-lbs. if lubricated).

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- 7.11.8 Attach anti-tamper seals to the designated bolts on the overpack.
- 7.11.9 Attach the tie-down cables/chains to the tie-down lugs on the sides of the overpack and tighten the binders on the cables/chains.
- 7.12 The Health Physics Technician shall survey all exterior surfaces of the cask for contamination and radiation levels. Decontaminate, as required, to meet the limits set forth in 49 CFR, Sections 173.441 and 173.443.
- 7.13 Before the cask leaves the facility, the following shall be confirmed:
 - 7.13.1 That any external lifting lugs are properly covered for transport.
 - 7.13.2 That trailer placarding and cask labelling meet DOT Specifications (CFR Title 49, Part 172).
 - 7.13.3 That exterior radiation and contamination levels conform to requirements as established in site radiation and contamination release procedures and DOT requirements.
 - 7.13.4 That the outer package is sealed with anti-tamper seals.
 - 7.13.5 That the drain plug is securely installed.

8.0 REPORTS AND RECORDS

The following reports shall accompany all loaded shipments:

- 8.1 Radioactive Shipment Manifest (RSM)--prepared by shipper health physics department
- 8.2 Vehicle Radiation Survey--prepared by shipper health physics department
- 8.3 Bill of Lading--prepared and certified by shipper
- 8.4 User Check-Off Sheet--prepared and signed by the shipper.

9.0 BACKUP LIFT STRUCTURE PROCEDURE

- 9.1 Open shipping box for the 1-13 backup lift structure.

NOTE: TREAT EQUIPMENT AS POSSIBLY CONTAMINATED.

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PART B - PACKAGE EVALUATION

1.0 INTRODUCTION

Part B of this Safety Evaluation for Packaging (SEP) will evaluate the adequacy and acceptability of packaging and transportation systems described in Part A of this SEP. Specifically, this section will evaluate the use CNS 1-13G to assure that it is capable of safely transporting irradiated N Reactor and single pass reactor (SPR) fuel elements from the K Basins to the 327 hot cells.

The SEP will also demonstrate that onsite transport of the CNS 1-13G will provide an equivalent degree of safety as would be provided by packages meeting U.S. Department of Transportation (DOT)/U.S. Nuclear Regulatory Commission (NRC) safety requirements.

1.1 BACKGROUND INFORMATION

As part of an effort to characterize the spent N Reactor and SPR fuel presently stored at Hanford in support of remediation efforts, it will be necessary to transfer samples of the irradiated fuel from their storage site in the 100 K Area to the 327 Laboratories.

The CNS 1-13G will be used in conjunction with an inner container to provide the required containment and protection for Type B highway route controlled quantity (HRCQ) of fissile material. The CNS 1-13G is authorized by the NRC to transport irradiated metallic and/or oxide fuel in commerce. The SARP documents the compliance of the cask with NRC Type B fissile requirements (ChemNuclear 1992). This compliance, however, is for irradiated fuel with intact cladding. Some of the samples that will be shipped are anticipated to have undergone cladding failure. Equivalent safety shall be obtained by limiting shipment mileage and via the addition of up to three (3) Single Fuel Element Containers (SFEC). These inner containers will be filled with water to reduce the exposure of the fuel to the atmosphere and to provide an environment similar to that found in the K Basins.

1.2 EVALUATION AND SUMMARY

1.2.1 Radiological Risk Evaluation

The radiological risk evaluation demonstrates that the proposed operation meets the onsite transportation criteria if the annual mileage for the shipping campaign is limited to 1,100 miles or less, based on potential fuel reactions.

1.2.2 Containment

The CNS 1-13 provides containment in accordance with 10 CFR 71. However, due to the unique nature of the fuel to be transported, the CNS 1-13G cask may become overpressurized due to chemical reactions within the cask under certain accident conditions; in this unplanned event, the maximum dose to offsite and onsite persons would be 0.36 mSv (36 mrem) and 17 mSv (1,700 mrem) cumulative effective dose equivalent (CEDE), respectively. This dose consequence mandates that the frequency of such a release be less than 10^{-3} /year (credible). During a fire incident where the water in the SFEC may boil and overpressurize the cask, the maximum dose to an offsite receptor would be 54 mSv (5,400 mrem) CEDE, which mandates that the frequency of such a release be less than 10^{-6} /year (incredible).

1.2.3 Shielding

The shielding analysis determined that the maximum surface dose for the CNS 1-13G during this shipping campaign will be .0631 mSv/h (6.31 mrem/h), the dose 2 m from the package surface is .00135 mSv/h (0.135 mrem/h) and the maximum dose to persons operating the transfer vehicle is 0.0003 mSv/h (0.03 mrem/h). All of these values are substantially less than the shielding acceptance criteria.

The cask will not suffer any degradation in shielding under hypothetical accident conditions.

1.2.4 Criticality

The CNS 1-13G SARP (ChemNuclear 1992) stipulates that the maximum amount of fissile material in the cask cannot exceed 500 g ^{235}U fissile equivalent. To conform to this limit, it is stipulated that each SFEC contain a maximum of one inner and one outer fuel element and that the cask be limited to transporting a maximum of two inner and two outer elements or three outer elements.

1.2.5 Structural

The CNS 1-13G SARP determined that the integrity of the CNS 1-13G will be maintained. The SFEC will maintain its integrity for the normal conditions of transport, but may rupture during accident conditions. Administrative controls, including the annual mileage limitations, render the predicted frequency of such an event to less than 10^{-3} /year to meet the onsite transportation safety criteria.

1.2.6 Thermal

The CNS 1-13G SARP determined that during normal conditions of transport, the surface temperature of the cask/overpack is within acceptable limits. During a fire based accident, the temperatures within the inner

cavity of the cask is predicted to exceed 300 °F, which indicates that the water within the inner containers may boil. This would pressurize the cask above its 15 psig limit, causing a potential release of radioactive material. Administrative controls (Part A, Section 6.0) render the predicted frequency of a fire based release to less than 10^{-6} /year, which is acceptable for the maximum dose associated with a fire based release.

1.2.7 Chemical and Gas Generation

Prior to shipment, the gas generation within the SFECs shall be determined, either through measuring the pressure increase in a sealed SFEC or through a calculational method. The maximum transfer time shall be limited to the number of days it takes for the concentration of H_2 within the CNS 1-13G to reach 2.5% by volume. If the maximum transfer time is less than one week, the payload shall be reduced accordingly to ensure a minimum of a one week shipping window.

It was also determined that in an accident scenario where the SFEC ruptures, exposing the fuel to the inner cask atmosphere, the chemical reaction could overpressurize the cask causing a release of radioactive material. Administrative controls (Part A, Section 6.0) render the predicted frequency of a release due to SFEC rupture and fuel/air reaction to less than 10^{-3} /year, which is acceptable for the maximum dose associated with such a release.

1.3 SAFETY EVALUATION METHODOLOGY

The safety of the transfer is based on the fact that the CNS 1-13G is an NRC certified package. It will be used as per the Certificate of Compliance (CoC) with the exception that a unique fuel will be shipped in it. To compensate for the breached fuel, the payload is placed into the cask in SFECs which will provide the necessary containment barriers and protection for transportation workers and offsite persons.

The packaging system has been evaluated for normal conditions of transport and accident conditions. In cases where the integrity of the cask is uncertain due to the contents, such as a fire condition, the maximum dose consequence of such an accident were evaluated and administrative controls were developed to insure that the frequency of such releases were commensurate with their severity as per *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials* (WHC 1994).

1.4 APPENDIX

1.4.1 References

- ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()*F, ChemNuclear, Inc, Columbia, South Carolina.
- WHC, 1994, *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials*, WHC-SD-TP-RPT-001, Westinghouse Hanford Company, Richland, Washington.

2.0 CONTENTS EVALUATION

2.1 CHARACTERIZATION

The payload for the CNS 1-13G cask consists of irradiated fuel elements that were used in the N Reactor and the SPRs on the Hanford Reservation from 1963 to 1987. The vast majority of the fuel elements that will be transported will be N Reactor fuel elements. Each N Reactor fuel element is a tube-in-tube design constructed of uranium metal with a Zircalloy 2 cladding. Since N Reactor fuel elements are considerably larger and have decayed less than the SPR fuel elements, they represent the worst case for shielding, criticality, and containment. Two preirradiation enrichment levels were used in the N Reactor fuel elements. Mark IV elements have a 0.94% ^{235}U enrichment level in both the inner and outer elements and weigh a total of 55 lb. Mark IA outer elements are 1.25% ^{235}U and the inner elements are 0.94% ^{235}U . The Mark IA elements are somewhat shorter than the Mark I elements and weigh about 40 lb (Bergsman 1993).

The fuel is presently stored in the 100 K East and 100 K West water filled fuel storage basins. The 100 K East basin contains about 1150 MTU of fuel in open canisters and 100 K West contains approximately 950 MTU of fuel in closed containers.

It is estimated an appreciable amount of the stored fuel has damaged cladding due to reactor discharge methods plus handling and transportation. As a consequence, the fuel stored in the open canisters in the 100 K East basin have contaminated the basin water via the release of soluble fission products. In addition, the reaction with the water has resulted in corrosion, and possibly the formation of uranium oxide and local degradation of the fuel. For the encapsulated fuel, the possibility exists that quantities of UH_3 have been created through a reaction of the fuel with the water. Given its highly reactive nature, it is unlikely that significant concentrations of UH_3 exist in the encapsulated fuel. Additionally, hydrogen is created by radiolysis and the corrosion of the uranium fuel.

2.1.1 Shielding and Dose Consequence

For shielding and containment purposes, the worst possible payload for the shipping campaign is assumed to be 16% ^{240}Pu N Reactor Mark I fuel five years after discharge from the N Reactor. This is conservative as few fuel elements reached 16% ^{240}Pu and the minimum decay time for any single fuel element is seven years. The worst case payload is assumed to have sufficient fuel to constitute 500 g ^{235}U fissile equivalent, which is the maximum allowed for criticality reasons (Part B, Section 2.1.2).

Table B2-1 contains the radioisotope activities for inner and outer fuel elements with 16% ^{240}Pu as determined by *Certification of ORIGEN2 Runs for N Reactor Fuel* (Schwarz 1991). This fuel represents the worst case payload for dose consequence due to an accidental release and shielding.

Table B2-1. Radioisotope Content for Worst Case Inner and Outer Fuel Elements.

Specie	Inner Fuel Content (Ci)	Outer Fuel Content (Ci)	Maximum Payload (Ci)	Specie	Inner Fuel Content (Ci)	Outer Fuel Content (Ci)	Maximum Payload (Ci)
$^{90}\text{Sr}/^{90}\text{Y}$	65.7	104.2	358.9	^{152}Eu	0.01	0.02	0.06
^{99}Tc	0.01	0.03	0.07	$^{110\text{m}}\text{Ag}$	0.01	0.02	0.05
$^{106}\text{Ru}/^{106}\text{Rh}$	19.01	40.6	130.9	^{238}Pu	0.90	1.93	4.94
$^{113\text{m}}\text{Cd}$	0.036	0.08	0.20	^{239}Pu	1.06	2.26	5.80
$^{125\text{m}}\text{Te}$	1.186	2.53	6.50	^{240}Pu	0.75	1.61	4.11
^{134}Cs	10.62	22.7	58.1	^{241}Pu	85.24	181.9	465.7
$^{137}\text{Cs}/^{137\text{m}}\text{Ba}$	87.86	187.5	480.0	^{241}Am	0.82	1.75	4.48
$^{144}\text{Ce}/^{144}\text{Pr}$	21.4	45.7	116.9	^{244}Cm	0.27	0.57	1.45
^{147}Pm	65.6	139.9	358.2	^{55}Fe	0.31	0.67	1.71
^{151}Sm	0.78	1.66	4.26	^{60}Co	0.10	0.22	0.57
^{154}Eu	2.24	4.78	12.3	^{125}Sb	0.17	0.36	0.93
^{155}Eu	0.72	1.54	3.95	$^{125\text{m}}\text{Te}$	0.04	0.09	0.23
^{85}Kr	7.03	15.0	38.4	$^{119\text{m}}\text{Sn}$	0.02	0.04	0.09
^{125}Sb	4.86	10.4	26.6	^{63}Ni	0.02	0.04	0.09
^3H	0.40	0.85	2.17	^{235}U	1.18×10^{-4}	2.52×10^{-4}	6.46×10^{-4}
$^{144\text{m}}\text{Pr}$	0.26	0.55	1.40				

2.1.2 Criticality

The SARP for the CNS 1-13G cask (ChemNuclear 1992) sets the maximum level of fissile material for the cask at 500 g ^{235}U fissile equivalent. The worst case would be to assume that the fuel being transported is unirradiated (green) fuel. Table B2-2 contains the ^{235}U content for the worst case N Reactor Mark 1A and Mark IV inner and outer fuel elements. SPR fuel elements are significantly smaller than their N Reactor counterparts and have correspondingly less ^{235}U (Bergsman 1993).

Table B2-2. ^{235}U Content for CNS 1-13G Payloads.

Element Type	Weight (grams)	Enrichment	^{235}U Weight (grams)
Mark 1A Inner	5,993	0.95%	57.0
Mark 1A Outer	12,786	1.25%	159.8
Mark IV Inner	7,484	0.95%	71.1
Mark IV Outer	15,967	0.95%	151.7

2.2 RESTRICTIONS

The CoC for the CNS 1-13G states that the fissile content shall not exceed 500 ^{235}U equivalent grams to maintain an adequate margin of safety with respect to criticality. The effective ^{235}U equivalent fissile content is equal to 1.66 times the sum of the plutonium and ^{233}U content. An inner fuel element contains a maximum equivalent ^{235}U weight of 71.1 g. An outer fuel element has a maximum equivalent ^{235}U mass of 159.8 g. Up to three inner containers will be used in the CNS 1-13G cask. Each of these containers may contain either an inner or outer fuel element or both. Empty SFECs may be used during shipment to ensure that the cask is not overloaded.

2.3 SIZE AND WEIGHT

The majority of the fuel used in the N Reactor was 26.1 in. in length, however, 12% of the fuel was shorter, with some individual elements being as short as 14.9 in. For the purposes of determining the source term and handling characteristics, the full 26.1 in. long fuel dimensions and weights were used. Table B2-3 contains the dimensions and weights of the longer inner and outer fuel elements.

Table B2-3. N Reactor/SPR Fuel Maximum Dimensions.

	Inner Element	Outer Element
Outer Diameter	1.28 in.	2.42 in.
Length	26.1 in.	26.1 in.
Weight	16.5 lb	35.2 lb

2.4 CONCLUSIONS

Based on the CoC for the CNS 1-13G cask, the fissile material that can be placed in the cask must be limited to 500 ^{235}U equivalent grams or less. The combination that would represent the largest amount of fissile material while still remaining under 500 ^{235}U equivalent grams is two inner and two outer elements or three outer elements.

2.5 APPENDIX

2.5.1 References

- Bergsman, K. H., 1993, *Hanford Irradiated Fuel Baseline*, WHC-SD-CP-TI-175, Rev. 1, Westinghouse Hanford Company, Richland, Washington.
- ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()F*, ChemNuclear, Inc, Columbia, South Carolina.
- Schwarz, R. A., 1991, *Certification of ORIGEN2 Runs for N-Reactor Fuel*, (internal memo RAS-91-009 to K. H. Bergsman, October 23), Westinghouse Hanford Company, Richland, Washington.

3.0 RADIOLOGICAL RISK EVALUATION

3.1 INTRODUCTION

The CNS 1-13G cask is a Type B cask that when used in the configurations authorized by the CoC is capable of maintaining containment subsequent to the 30 ft drop, puncture bar drop and fire conditions. For the N Reactor/SPR fuel characterization shipping campaign, a configuration will be utilized that is not authorized in the CoC. With this configuration, it cannot be established that the CNS 1-13G cask will maintain an adequate degree of containment for a fire case, a severe collision or rollover accident, due to the nature of the fuel. Therefore, a radiological risk evaluation is required to determine the predicted frequency of such potential releases and the acceptance criteria for the containment analysis (Part B, Section 4.0).

3.2 SUMMARY OF RESULTS

The radiological risk evaluation determined that if the CNS 1-13G is transported less than 1,100 miles during any rotating twelve month period, and is not shipped during icy conditions or conditions of low visibility, fire based releases can be regarded as incredible events ($<10^{-6}$ /year). Release due to accidents other than fire will be regarded as credible events ($<10^{-3}$ /year) with this mileage limitation.

3.3 RISK ACCEPTANCE CRITERIA

The dose consequences for onsite and offsite personnel are independently evaluated. The onsite dose limitations for probable, credible, and incredible accident frequencies are presented in Table B3-1 (WHC 1994).

The more stringent dose limitations for offsite personnel are presented in Table B3-2 (WHC 1994).

Table B3-1. Maximum Dose Acceptance Criteria for Onsite Personnel.

Description	Annual Frequency	Dose Limit* (mSv)
Incredible	$< 10^{-6}$	Finite
Credible	10^{-6} to 10^{-3}	50/150/500
Probable	10^{-3} to 1	2/6/20

* CEDE/lens of eye/all other organs

Table B3-2. Maximum Dose Acceptance Criteria for Offsite Personnel.

Description	Annual Frequency	Dose Limit* (mSv)
Incredible	$< 10^{-6}$	250
Credible	10^{-6} to 10^{-3}	5/15/50
Probable	10^{-3} to 1	0.1/0.3/1

* CEDE/lens of eye/all other organs

The dose consequences refer to the maximum expected individual exposure. Dose consequence criteria are expressed in terms of EDE or organ values, whichever is more limiting.

The dose consequence study for the transfer of N Reactor/SPR fuel samples (Part B, Section 4.0) indicates that the maximum offsite dose consequence for a loss of containment due to fire is 54 mSv (5,400 mrem) CEDE, which falls into the acceptable dose range for an incredible event. The study also determined that the dose consequence to onsite and offsite persons for a nonfire based release is within the acceptance limits for a credible event. Therefore, the administrative controls in Part A, Section 4.0 must be utilized to ensure that the annual frequency of a loss of containment accident due to fire is less than 10^{-6} , and the frequency of other accidental release events is less than 10^{-3} .

3.4 ACCIDENT FREQUENCY ASSESSMENT

3.4.1 Approach

The SARP for the CNS 1-13G cask (ChemNuclear 1992) indicates that the cask can maintain containment during severe collision and/or fire scenarios as described in 10 CFR 71. However, this shipping campaign will utilize a different configuration for the payload than was analyzed in the SARP. The presence of water filled SFECs could result in steam generation, cask overpressurization and a radioactive release during a fire based accident. If the SFECs fail during an collision or rollover accident and the water drains out of it, the gas generation analysis (Part B, Section 9.0) indicates that the fuel could react with the cask atmosphere, overpressurizing the cask and releasing radioactive material to the environment.

Because the dose consequences of accidents involving fire are substantially higher than the consequences of other accidents, the frequencies of the accident types are calculated separately. The data used are taken from *Severities of Transportation Accidents Involving Large Packages* (Dennis et al. 1978) except where stated otherwise.

The data and models set forth in *Severities of Transportation Accidents Involving Large Packages* (Dennis et al. 1978) form an appropriate basis for the accident frequency calculations of this evaluation because the reference is directed specifically to transportation accidents affecting large packages (>2 tons). The CNS 1-13G is a large package weighing in excess of 12 tons. The study was particularly directed towards radioactive material and has been frequently used as a basis for studies which involve the transportation of spent fuel.

Severities of Transportation Accidents Involving Large Packages (Dennis et al. 1978) contains a conservative overall accident frequency based on national statistics involving trucks of 2.5×10^{-6} /mile for interstate motor carriers where an accident is any event that results in a fatality, injury or property damage of \$250 or more. This value and definition are consistent with reportable truck accident data for the Hanford Site as tracked by DOE.

Table B3-3 contains the data used for the assessment of the frequency of the different types of accidents (Dennis et al. 1978).

Table B3-3. Truck Accident Data.

Accident Type	Percent of Total Accidents	Frequency of Fire (%)	Percent of Total Fires
Collision with Auto	49.7	0.3	10.4
Collision with Truck	15.0	0.8	7.8
*Collision with Fixed Object	14.1	0.4	3.7
*Other Collision	5.1	0.9	3.0
*Ran Off Road	7.1	1.1	4.9
*Overturns	2.7	1.2	2.1
*Other Noncollision	6.0	13.0	48.8
*Fire Only	0.3	100.0	19.3
Total	100.0	1.6	100.0

* Applicable Accident Scenarios for the CNS 1-13G

Due to operational controls imposed on the shipment (Part A, Section 4.0), the collision of the CNS 1-13G transport vehicle with another

truck or an automobile are considered implausible. Only the frequencies for Collision with a Fixed Object, Other Collision, Ran Off Road, Overturns, Other Noncollision and Fire Only are considered.

3.4.2 Accidents Not Involving Fire

The accident frequencies for all applicable accident types are considered in determining the non-fire accident rate. When the Collisions with Other Vehicles and the Fire Only categories are omitted, 35% of the accidents must be considered. It is assumed that the probability of the CNS 1-13G transport vehicle being involved in a non-fire accident resulting in a radioactive release is 35% of 2.5×10^{-6} accidents per mile or 8.8×10^{-7} accidents per mile. The frequency limit for this type of release is 1×10^{-3} /year, imposing an annual mileage limit of 1,100 miles.

3.4.3 Accidents Involving Fire

The frequency for accidents involving fire is calculated to verify that the results do not invoke a more restrictive mileage limit than that required for the accidents not involving fire. The applicable truck accident frequencies in Table B3-3 are multiplied by the fire frequencies to obtain a fire accident conditional probability of 0.013 fires/accident. This value is then multiplied by the overall accident frequency of 2.5×10^{-6} accidents/mile to obtain a value of 3.3×10^{-8} fire accidents/mile.

The fire duration model found in *Severities of Transportation Accidents Involving Large Packages* (Dennis et al. 1978) was considered in setting a 1% conditional probability of damage due to a fire of significant duration. This model was designed to correlate with accident data and the assumptions that form the basis for this model are consistent with a negligible value for the fraction of fires lasting longer than 1 hour. The particular assumptions of the model that support the selection of a 1% conditional probability for the release due to a fire accident experienced by the CNS 1-13G are:

- Local fire departments are effective against all fires.
- Hand extinguishers are effective against fires that involve less than 75 gallons of fuel or 1000 lb of combustible cargo.
- Fire duration is assumed to be less than 45 minutes in all cases.

Due to the thermal mass of the cask, it is highly unlikely that a fire would affect the containment system by heating the water inside the SFECs before mitigating action could be taken. Because the shipments will take place entirely within the boundaries of the Hanford Site, the Hanford Fire Department will respond within 15 minutes to any fire that occurs during the course of the operation. The Hanford Fire Department is equipped and trained to fight fires involving hazardous and radioactive materials. Additionally, the thermal analysis performed in support of the NRC cask license indicates that it would take in excess of two hours for any water in the cask to reach its boiling point (ChemNuclear 1992). It is reasonable to assume that the

Hanford Fire Department will be capable of taking mitigating action within a two hour period after the initiation of the fire accident. Therefore, the 1% release value used by *Severities of Transportation Accidents Involving Large Packages* (Dennis et al. 1978) is considered to be valid for this shipping campaign.

This value of 1% of fire accidents resulting in release yields an overall conditional probability of 3.3×10^{-10} releases/mile due to fire, permitting an annual mileage limit of over 3,000 miles without exceeding the acceptance criteria. Therefore the non-fire accident case is more restrictive.

3.5 DOSE CONSEQUENCE ASSESSMENT

Part B, Section 4.0 documents the dose consequence analysis for a fire based release of radioactive materials. The dose to an offsite receptor is 54 mSv (5,400 mrem) CEDE, which meets the acceptance criteria for an incredible release frequency ($<10^{-6}$ /year). The dose to an onsite receptor for a rollover or collision based release is 17 mSv (1,700 mrem) CEDE and the dose to an offsite receptor is 0.36 mSv (36 mrem) CEDE, both of which meet the acceptance criteria for a credible release frequency ($<10^{-3}$ /year).

3.6 RISK EVALUATIONS AND CONCLUSIONS

If the mileage for the shipping campaign is limited to less than 1,100 miles per 12 month period, the frequency for releases from fire accidents will be in the incredible range, while releases for releases caused by accidents other than fire will have a credible frequency, which meets the acceptance criteria for the maximum doses predicted by the containment analysis (Part B, Section 4.0).

3.7 APPENDIX

3.7.1 References

- ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()*F, ChemNuclear, Inc, Columbia, South Carolina.
- Dennis, A. W., J. T. Foley, W. F. Hartman, and D. W. Larson, 1978, *Severities of Transportation Accidents Involving Large Packages*, SAND77-0001, Sandia National Laboratories, Albuquerque, New Mexico.
- WHC, 1994, *Report on Equivalent Safety for Transportation and Packaging of Radioactive Materials*, WHC-SD-TP-RPT-001, Westinghouse Hanford Company, Richland, Washington.

4.0 CONTAINMENT EVALUATION

4.1 INTRODUCTION

The CNS 1-13G is an NRC certified Type B cask that is capable of maintaining containment subsequent to the accident conditions, as simulated by test such as the 30 ft drop, puncture bar, etc., as detailed in 10 CFR 71 (Part B, Section 7.0). The cask configuration used to transport irradiated fuel elements from the 100 K Area to the 300 Area differs from the configuration analyzed in the SARP in that it utilizes small, water filled inner containers. It has been determined that a fire case might cause the water to boil, which in turn will cause overpressurization of the cask and a loss of containment. Additionally, a loss of water from the inner containers subsequent to a collision or rollover accident could potentially result in a chemical reaction between the fuel and cask atmosphere that could also overpressurize the cask. This section evaluates the dose consequence of these release scenarios.

4.2 SUMMARY OF RESULTS

The SARP for the CNS 1-13G cask (ChemNuclear 1992) states that for normal conditions of transport and accident conditions the cask will meet or exceed the requirements set forth in 10 CFR 71 and 49 CFR 171-178 for the transport of the package in commerce. However, in the configuration that will be used for the shipment of irradiated fuel from the 100 K to the 300 Area, a loss of containment is possible for a fire based accident and for collision and rollover accidents. The radiological risk analysis (Part B, Section 3.0) determined that a fire based release is an incredible event ($<10^{-6}$ /year) and a collision or rollover event is credible ($<10^{-3}$ /year). The maximum dose consequence for offsite persons for a fire based release is 54 mSv (5,400 mrem) CEDE in the 50 years after a release, which is acceptable for an incredible event. The dose consequence for onsite persons is not required to be evaluated for incredible events. The maximum dose consequence for a collision/rollover based release is 17 mSv (1,700 mrem) CEDE for onsite receptors and 0.36 mSv (36 mrem) CEDE for offsite receptors, which is acceptable for credible events.

4.3 CONTAINMENT SOURCE SPECIFICATION

The worst case fuel for the shipping campaign is assumed to be 16% ^{240}Pu irradiated N Reactor fuel that has been allowed to decay for five years. This is a conservative estimate as few fuel elements were allowed to reach 16% ^{240}Pu , and the minimum amount of decay that has been experienced by an N Reactor/SPR fuel element is seven (7) years.

The fuel portion of the rods that will be shipped in the CNS 1-13G is primarily $^{235}\text{U}/^{238}\text{U}$, but it also contains significant amounts of transuranic materials such as ^{240}Pu and ^{239}Pu as well as fission products such as $^{137}\text{Cs}/^{137\text{m}}\text{Ba}$. The cladding will be contaminated with activated material such as ^{60}Co . Table B4-1 provides the expected activity level for the CNS 1-13G packaging when it contains 500 ^{235}U equivalent grams of fissile material, which is the maximum amount allowed by the SARP (ChemNuclear 1992).

Table B4-1. Activity of Worst Case Payload.

Specie	Maximum Payload (Ci)	Specie	Maximum Payload (Ci)
$^{90}\text{Sr}/^{90}\text{Y}$	358.9	^{152}Eu	0.06
^{99}Tc	0.07	$^{110\text{m}}\text{Ag}$	0.05
$^{106}\text{Ru}/^{106}\text{Rh}$	130.9	^{238}Pu	4.94
$^{113\text{m}}\text{Cd}$	0.20	^{239}Pu	5.80
$^{125\text{m}}\text{Te}$	6.50	^{240}Pu	4.11
^{134}Cs	58.1	^{241}Pu	465.7
$^{137}\text{Cs}/^{137\text{m}}\text{Ba}$	480.0	^{241}Am	4.48
$^{144}\text{Ce}/^{144}\text{Pr}$	116.9	^{244}Cm	1.45
^{147}Pm	358.2	^{55}Fe	1.71
^{151}Sm	4.26	^{60}Co	0.57
^{154}Eu	12.3	^{125}Sb	0.93
^{155}Eu	3.95	$^{125\text{m}}\text{Te}$	0.23
^{85}Kr	38.4	$^{119\text{m}}\text{Sn}$	0.09
^{125}Sb	26.6	^{63}Ni	0.09
^3H	2.17	^{235}U	6.46×10^{-4}
$^{144\text{m}}\text{Pr}$	1.40		

4.4 NORMAL TRANSFER CONDITIONS

The SARP for the CNS 1-13G cask (ChemNuclear 1992) states that for normal conditions of transport:

The cask was analyzed to show that design specifications met or exceed 10 CFR 71 Regulations and DOT 49 CFR 171-178 Regulations. The tests or assessments set forth in accordance with [10 CFR] 71.71, Normal Conditions of Transport, provide the assurance that the products contained in the shielded package CNS 1-13G, during transport and there is no reduction in the effectiveness of the package which meets the requirements set forth in [10 CFR] 71.51.

Therefore, no analysis of releases of radioactive material during normal conditions of transport is required.

4.5 ACCIDENT CONDITIONS

The SARP for the CNS 1-13G (ChemNuclear 1992) states that for accident conditions:

[T]he cask will maintain its containment capability throughout the hypothetical accident conditions. The cask is designed, fabricated, and leak tested to preclude a release of radioactive material in excess of the limits prescribed in 10 CFR 71.51. For verifiable containment criterion, the cask is pneumatically pressurized to 15 psi and, while under pressure, seals are soap bubble checked. The leakage acceptance criteria is no visible bubbles.

The SARP thermal analysis indicates that the maximum temperatures within the inner cavity of the CNS 1-13G will exceed 400 °F, which is sufficient to boil the water within the inner containers. The internal pressure within the cask would then exceed the 15 psig test pressure, indicating that containment might not be maintained.

During a collision or rollover based accident, the inner containers may rupture, exposing the irradiated fuel elements to the atmosphere within the cask. The gas generation analysis (Part B, Section 9.0) indicates that the gas generated by the chemical reaction between the fuel, atmosphere and residual water could potentially overpressurize the cask and cause a release.

Mileage restrictions as described in Part B, Section 3.0 render the frequency of fire based accidents to less than 10^{-6} per year and collision/rollover accidents to less than 10^{-3} per year. This section evaluates the dose consequences of these accidents to offsite persons and, for credible releases ($<10^{-5}$ /year) the maximum dose to onsite persons. For incredible events (10^{-6} /year) evaluation of the dose consequence to onsite persons is not required.

4.5.1 Conditions to be Evaluated

This analysis determines the dose consequence to an offsite receptor for a fire based accident where failure of the gasket and lid causes a release of radioactive material to the environment. A separate analysis determines the dose to onsite and offsite persons for a collision or rollover based release.

4.5.2 Release Acceptance Criteria

The maximum acceptable doses as described in the *Packaging Design Criteria for the N Reactor/Single Pass Reactor Fuel Characterization Shipments* (WHC 1994) for onsite and offsite personnel as a function of release probability are summarized in Table B4-2. The radiological risk analysis, (Part B, Section 3.0) determined that if the annual mileage of the CNS 1-13G is limited to 1,100 miles per year or less, a fire based release will be an incredible event ($<10^{-6}$ /year) and a rollover or collision based release will be a credible event ($<10^{-3}$ /year).

Table B4-2. Dose Limitations for Onsite and Offsite Receptors.

Description	Annual Frequency	Dose Limit Onsite Receptor	Dose Limit Offsite Receptor
Incredible	$< 10^{-6}$	Finite	250
Credible	10^{-6} to 10^{-3}	50/150/500	5/15/50
Probable	10^{-3} to 1	2/6/20	0.1/0.3/1

* EDE/lens of the eye/all other organs (mSv).

4.5.3 Release Model

4.5.3.1 General Approach and Assumptions. The approach for the dose consequence analysis is the inverse of the Q-system methodology as outlined in the IAEA Safety Guide No. 7 (IAEA 1990) as amended in 1990. In the Q-system, five routes of exposure from an accidental release of radioactive material are considered. In special cases such as alpha particle or neutron emitters, other exposure routes are considered. The five major Q-values are external exposure to photon, external exposure to beta particles, inhalation, skin contamination and ingestion, and submersion in a cloud of gaseous isotopes.

Modification to the IAEA scenarios are incorporated to more closely describe the particular conditions of the CNS 1-13G shipment.

In this IAEA system, the Q-values that are calculated are the radionuclide activities corresponding to each exposure route which causes the

individual to receive the effective dose equivalent limit. The minimum Q-values define the A_2 values for the shipped materials. In the case of non-dispersible materials (limited by the A_1 values) only the first two Q-values are used.

Note that for all radiation except neutrons, protons, and heavier charged particles (including alpha particles), $1 \text{ Gy} = 1 \text{ Sv}$, that is $1 \text{ rad} = 1 \text{ rem}$.

For the CNS 1-13G fire dose consequence analysis, all five IAEA exposure pathways are considered. Several pathways are judged to be insignificant with respect to the others and are not evaluated in detail. Detailed calculations are based on a postulated worst case fire accident to the CNS 1-13G cask whenever possible.

The Q system was developed as an all-encompassing generalized methodology using only the isotope as the defining variable. In this evaluation, the package design and the transfer method are considered. Some exposure routes may be considered incredible and although these routes may be addressed in the IAEA model, they are disregarded in the analysis (Part B, Sections 4.5.3.1.4 and 4.5.3.1.5).

Although the presence of the water in the inner containers may preclude ignition, it is conservatively assumed that the uranium will burn nonexplosively during a fire based accident (Katz 1958). A similar reaction is conservatively assumed to take place for the nonfire accident.

The analysis is based on the maximum source term of Table B4-1 which lists the estimated worst case cask inventory.

4.5.3.1.1 External Dose Due to Photon Exposure. The great distance between the cask and the prospective receptor ($>100 \text{ m}$) a significant dose due to photon exposure is not anticipated.

4.5.3.1.2 External Dose Due to Exposure to B-Particle Emitters. Again, due to the distance between the cask and the potential offsite receptor, a significant dose due to B particle exposure is not anticipated.

4.5.3.1.3 Internal Dose Due to Inhalation. The dose due to inhalation was calculated using the GENII computer code (Part B, Section 4.5.3.2). It is conservatively assumed that the source expands from a ground level point source. The offsite receptor is assumed to be at a point just beyond the Hanford boundary at which the highest predicted integrated exposure has historically been exceeded not more than 0.5% of the time. Such weather-dependant data for the various locations on the periphery of the Hanford Site are included in the libraries in GENII.

4.5.3.1.4 Doses Due to Skin Contamination and Ingestion. The IAEA model assumes that 1% of the package contents are spread over an area of 1 m^2

and handling of the debris results in contamination of the hands to 10% of this level. It is further assumed that the person is not wearing gloves but that a recognition of the possibility of contamination results in a washing of the hands within a period of five hours. The IAEA scenario assumes that the person ingests all of the contamination from 10 cm² of skin over a 24 hour period. It is considered very unlikely that skin contamination would be a factor in the accidents identified in the CNS 1-13G fire accident dose consequence analysis, so this route of exposure is not considered in the dose consequence calculation.

4.5.3.1.5 Dose Due to Gaseous Isotopes. Radiation exposure may result from submersion in a cloud of gaseous isotopes which do not become incorporated into the body. Since the CNS 1-13G cask will contain only negligible concentrations of these gasses, no exposure is associated with this path.

4.5.3.1.6 Special Considerations. Bremsstrahlung has been included in the evaluation of photon effects and the effects of short-lived daughter products have been included in all calculations. Where these isotopes are significant they are assumed to be in equilibrium with their longer lived parent isotopes.

4.5.3.1.7 Multiple Exposure Pathways. The Q system analyzes each exposure route separately and limits the activity allowed in the package (A_1/A_2) to the smallest of these values. This analysis for the CNS 1-13G cask also analyzes each exposure route separately and reports the largest of these effective dose equivalent values. However, if two exposure routes are comparable in magnitude they are added together.

4.5.3.1.8 Receptor Location. It is assumed that the release occurs just north of the 300 Area with the closest point of approach for the general population being approximately 1,240 m east, across the Columbia River. To assess the maximum offsite dose, it is assumed that a farmer lives at this point who grows all of his/her own food. To create the most conservative case the release is assumed to take place in the fall, just prior to harvest. The only methods by which dose could be imparted such an individual are inhalation and ingestion. The GENII program was utilized to determine the maximum potential dose. The onsite receptor is assumed to be 100 m east of the release as is consistent with IAEA dose assessment philosophy.

4.5.3.2 Computer Code Description. The GENII computer code is the standard dose assessment code used on the Hanford Site (Napier 1988). GENII is a computer code that uses a straight line bivariate Gaussian plume meteorological model to calculate individual and population doses due to inhalation of, ingestion of, and submersion in radiological releases to the atmosphere, the earth, and/or water. Immersion in gaseous effluents is also calculated. This model is generally accepted for atmospheric dispersion

estimates related to radiological and toxicological releases. It also incorporates the ICRP 26 dose methodology and environmental pathway analysis models.

The GXQ² computer code (Hey 1993) evaluates the Ψ/Q dispersion factor for the accident which is input into the GENII computer code along with the source term information. The Ψ/Q value is based on release and receptor location and Hanford meteorological data. GXQ² calculates a conservative value that will not be exceeded more than 0.5% of the time.

The GENII models are generally considered valid under the following conditions:

1. Distance from the source to the dose point is large compared to the size of the source. The source is assumed to be a point. This model is generally not valid at source to dose point distances of less than 100 m (i.e., for ground releases the code overestimates the dose; for elevated releases it underestimates it). The dose at 100 m is also slightly suspect for this reason. The IAEA Safety Guide No. 7 (IAEA 1990) assumes that conservatism more than offsets non-conservative assumptions in its methodology for airborne releases.
2. The terrain is flat and uniform, and the plume is totally reflected by the ground. This makes the model conservative, especially when the ground is rising since the plume does deposit onto the ground. When the ground is falling, the effective height of the plume is larger.
3. Weather conditions are stationary and turbulence with steady-state pollutant concentration. The atmospheric conditions are set at the time of release. In reality, they seldom stay constant.
4. The wind speed is nonzero. This is correct for elevated releases, but can lead to errors at ground level. This assumption neglects the effects due to diffusion along the direction of travel and the transport is dominated by bulk movement of the air.
5. Long diffusion times and spatially constant flow. This, and the previous condition make GENII overestimate the doses for very short releases such as explosions and tank ruptures.
6. The model does not take into account resuspension or deposition. Instead, the code computes air concentration without losses (or gains) to the plume. However, radioactive decay is included.

4.5.4 Release Calculations

The computer code GENII was used to calculate the dose from an acute release of this source to an onsite receptor 100 m downwind of the release and an offsite receptor located 1,240 m east of the release who grows his/her own food at this site. The GENII input data and results are included in Part B, Section 4.6.2.

The calculations assume that the receptor remains at the dose point throughout the entire passage of the plume of airborne material.

4.5.4.2 Release Fraction.

4.5.4.2.1 Fire Based Release. During a fire based accident, the uranium may burn, but will do so nonexplosively (Katz 1958). According to *Recommended Values and Technical Bases for Airborne Release Fractions, Airborne Release Rates and Respirable Fractions at DOE Non-Reactor Facilities* (Mushima 1993), the fraction of the source that will be suspended in air and will thereby be available for inhalation and/or ingestion by an offsite person is 1.0×10^{-3} . It is assumed that due to excessive overpressurization, the inner container and cask provide a negligible degree of containment.

4.5.4.2.2 Non Fire Based Release. It is assumed that the cask will have a 1 in² breach due to the overpressurization (Sandia 1981) and that the inner container will suffer failure of the tube wall (Part B, Section 7.0). A release analysis performed in support of this shipping campaign (Plys 1994) determined that during a collision or rollover accident, fuel will react nonexplosively. The inner container and cask will confine 98% of the radioisotopes, so only 1.96% of the suspended material will be released to the environment. The total release fraction for the non fire based release will be 7.0×10^{-6} .

4.5.4.3 Inhalation Dose.

4.5.4.2.1 Fire Based Release. For a fire based accident involving a release 10^{-3} of the worst case inventory as detailed in Table B4-1, the maximum effective dose equivalent due to inhalation for an offsite person in the year of the accident is calculated to be 0.83 mSv (83 mrem) and 19 mSv (1,900 mrem) for the 50 years following the release. The dose is primarily to the bone surface, and is caused by ²⁴¹Pu. These dose figures assume that the receptor remains at the dose point throughout the entire passage of the plume of airborne material and are hence conservative.

4.5.4.2.1 Non Fire Based Release. For a release subsequent to a collision or rollover accident, the maximum effective dose equivalent due to inhalation for an offsite person in the year of the accident is 0.006 mSv (0.6 mrem) and 0.133 mSv (13.3 mrem) for the 50 years following the release. This approximately 1% of the fire dose. The dose is primarily due to a large ²⁴¹Pu dose to the bone surface.

The maximum dose to an onsite person 100 m from the release site is 0.34 mSv (34 mrem) the year of the release and 17 mSv (1,700 mrem) in the 50 years subsequent to the release. As with the offsite dose, the largest fraction of the dose is due to ^{241}Pu dose to the bone surface. The dose figures assumed that the receptor remains at the dose point throughout the entire passage of the plume.

4.5.4.4 Ingestion Dose.

4.5.4.4.1 Fire Based Accident. For a fire based accident involving the release of 10^{-3} of the worst case inventory, the maximum dose due to ingestion of contaminated foodstuffs during the first year subsequent to the release is 15 mSv (1,500 mrem) EDE. In the 50 years following the release, the total dose due to ingestion will be 35 mSv (3,500 mrem) CEDE. This dose is primarily due to the dose from ^{137}Cs to the bone surface.

4.5.4.4.2 Non Fire Based Accident. The maximum dose to an onsite person due to contaminated foodstuffs following a non fire based release is 0.10 mSv (10 mrem) in the first year subsequent to the release and 0.25 mSv (25 mrem) for the 50 years following the release. As with the fire case, the dose is primarily due to the high dose to the bone surface from ^{137}Cs .

4.6 CONTAINMENT EVALUATION

4.6.1 Fire Based Accident

The cumulative dose due to inhalation and ingestion for offsite persons is 16 mSv (1,600 mrem) EDE during the first year after the accidental release and 54 mSv (5,400 mrem) CEDE in the 50 years subsequent to the release. This is within the acceptance criteria for offsite persons for an incredible release event.

4.6.2 Non Fire Based Accident

The cumulative dose due to inhalation and ingestion for an offsite person subsequent to a non fire based release of radioactive material from the CNS 1-13G is 0.11 mSv (11 mrem) during the first year and 0.36 mSv (36 mrem) in the 50 years after the release. This is within the acceptance criteria for a credible release event.

Since ingestion is not a credible dose pathway for onsite persons, the maximum dose is the inhalation dose, which is 17 mSv (1,700 mrem) for the 50 years following a release and 0.34 mSv (34 mrem) for the first year after a release, which is within the acceptance criteria for a credible release event.

4.7 APPENDICES

4.7.1 References

- ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()*F, ChemNuclear, Inc., Columbia, South Carolina.
- Hey, B. E., 1993, *GXQ 3.1 User's Guide*, WHC-SD-GN-SWD-30002, Westinghouse Hanford Company, Richland, Washington.
- IAEA, 1990, *IAEA Safety Guide Number 7*, International Atomic Energy Agency, Vienna, Austria.
- Katz, J. J., 1958, *Chemistry of Uranium*, Argonne National Laboratory, Published by U.S. Atomic Energy Commission, Oak Ridge, Tennessee.
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- Napier, B. A., R. A. Peloquin, D. L. Streng, and J. V. Ramsdell, 1988, *GENII - The Hanford Environmental Radiation Dosimetry Software System*, PNL-6584, Battelle Northwest, Richland, Washington.
- Plys, M. G., 1994, *Single Element Canister Drain and Cask Rollover Accidents*, (Letter to P. F. Stevens (WHC), August 23), Fauskey and Associates, Chicago, Illinois.
- SNL, 1981, *Transportation Accident Scenarios for Commercial Spent Fuel*, SAND80-2124, Sandia National Laboratories, Albuquerque, New Mexico.
- WHC, 1994, *Packaging Design Criteria for the N Reactor/Single Pass Reactor Fuel Characterization Shipments*, WHC-SD-TP-PDC-022, Westinghouse Hanford Company, Richland, Washington.

4.7.2 GENII Input and Output Files

4.7.2.1 Input File for Offsite Receptor with Inhalation, Ingestion and 100% Suspension.

Input by: HJ Goldberg

Checked by: PD Rittmann

Program GENII Input File ##### 8 Jul 88

Title: N-Reactor Characterization Shipment Accident in 237 Building-farmer

\GENII\nreact-f.in

Created on 07-22-1994 at 14:10

OPTIONS===== Default =====

F	Near-field scenario?	(Far-field)	NEAR-FIELD:	narrowly-focused
F	Population dose?	(Individual)		release, single site
T	Acute release?	(Chronic)	FAR-FIELD:	wide-scale release,
	Maximum individual data set used			multiple sites

TRANSPORT OPTIONS=====		Section	EXPOSURE PATHWAY OPTIONS=====		Section
T	Air Transport	1	F	Finite plume, external	5
F	Surface Water Transport	2	T	Infinite plume, external	5
F	Biotic Transport (near-field)	3,4	F	Ground, external	5
F	Waste Form Degradation (near)	3,4	F	Recreation, external	5
			T	Inhalation uptake	5,6
			T	Drinking water ingestion	7,8
			T	Aquatic foods ingestion	7,8
			T	Terrestrial foods ingestion	7,9
			T	Animal product ingestion	7,10
			F	Inadvertent soil ingestion	

REPORT OPTIONS=====

T	Report AEDE only
F	Report by radionuclide
F	Report by exposure pathway
F	Debug report on screen

INVENTORY

- 4 Inventory input activity units: (1-pCi 2-uCi 3-mCi 4-Ci 5-Bq)
 0 Surface soil source units (1- m2 2- m3 3- kg)
 Equilibrium question goes here

Use when	---Release Terms---			-----Basic Concentrations-----				
	transport selected			near-field scenario, optionally				
Release Radio- nuclide	Air /yr	Surface Water /yr	Buried Waste /m3	Air /m3	Surface Soil /unit	Deep Soil /m3	Ground Water /L	Surface Water /L
H 3	2.2E+00							
KR85	3.8E+01							
SR90	3.6E+02							
Y 90	3.6E+02							
TC99	7.0E-02							
RU106	1.3E+02							
AG110M	5.0E-02							
CD113M	2.0E-01							
SB125	2.7E+01							
TE125M	6.5E+00							
CS134	5.8E+01							
CS137	4.8E+02							
CE144	1.2E+02							
PR144	1.2E+02							
PR144M	1.4E+00							
PM147	3.6E+02							
EU152	6.0E-02							
EU154	1.2E+01							
EU155	4.0E+00							
U 235	6.5E-04							
U 238	1.4E-02							
PU238	4.9E+00							
PU239	5.8E+00							
PU240	4.1E+00							
PU241	4.7E+02							
AM241	4.5E+00							
CM244	1.5E+00							

Use when	---Derived Concentrations---			
	measured values are known			
Release Radio- nuclide	Terres. Plant /kg	Animal Product /kg	Drink Water /L	Aquatic Food /kg

====BIOTIC TRANSPORT OF BURIED SOURCE=====SECTION 4=====

T Consider during inventory decay/buildup period (T/F)?

T Consider during intake period (T/F)? 1-Arid non agricultural

0 Pre-Intake site condition..... 2-Humid non agricultural

3-Agricultural

EXPOSURE #####

====EXTERNAL EXPOSURE=====SECTION 5=====

Exposure time: Residential irrigation:

0 Plume (hr) T Consider: (T/F)

0 Soil contamination (hr) 0 Source: 1-ground water

0 Swimming (hr) 2-surface water

0 Boating (hr) 0 Application rate (in/yr)

0 Shoreline activities (hr) 0 Duration (mo/yr)

0 Shoreline type: (1-river, 2-lake, 3-ocean, 4-tidal basin)

0 Transit time for release to reach aquatic recreation (hr)

1.0 Average fraction of time submersed in acute cloud (hr/person hr)

====INHALATION=====SECTION 6=====

8766.0 Hours of exposure to contamination per year

0 0-No resus- 1-Use Mass Loading 2-Use Anspaugh model

0 pension Mass loading factor (g/m3) Top soil available (cm)

====INGESTION POPULATION=====SECTION 7=====

1 Atmospheric production definition (select option):

0 0-Use food-weighted chi/Q, (food-sec/m3), enter value on this line

1-Use population-weighted chi/Q

2-Use uniform production

3-Use chi/Q and production grids (PRODUCTION will be overridden)

0 Population ingesting aquatic foods, 0 defaults to total (person)

0.0 Population ingesting drinking water, 0 defaults to total (person)

F Consider dose from food exported out of region (default=F)

Note below: S* or Source: 0-none, 1-ground water, 2-surface water

3-Derived concentration entered above

==== AQUATIC FOODS / DRINKING WATER INGESTION=====SECTION 8=====

F Salt water? (default is fresh)

USE ?	FOOD TYPE	TRAN- SIT hr	PROD- UCTION kg/yr	-CONSUMPTION- HOLDUP da	RATE kg/yr	DRINKING WATER
T	FISH	0.00	0.0E+00	1.00	40.0	0 Source (see above)
T	MOLLUS	0.00	0.0E+00	0.00	6.9	T Treatment? T/F
T	CRUSTA	0.00	0.0E+00	0.00	6.9	1.0 Holdup/transit(da)
T	PLANTS	0.00	0.0E+00	0.00	6.9	730.0 Consumption (L/yr)

TIME #####

1 Intake ends after (yr)
 50 Dose calc. ends after (yr)
 0 Release ends after (yr)
 0 No. of years of air deposition prior to the intake period
 0 No. of years of irrigation water deposition prior to the intake period

FAR-FIELD SCENARIOS (IF POPULATION DOSE) #####

0 Definition option: 1-Use population grid in file POP.IN
 0 2-Use total entered on this line

NEAR-FIELD SCENARIOS #####

Prior to the beginning of the intake period: (yr)
 0 When was the inventory disposed? (Package degradation starts)
 0 When was LOIC? (Biotic transport starts)
 0 Fraction of roots in upper soil (top 15 cm)
 0 Fraction of roots in deep soil
 0 Manual redistribution: deep soil/surface soil dilution factor
 0 Source area for external dose modification factor (m2)

TRANSPORT #####

====AIR TRANSPORT=====SECTION 1=====
 0 0-Calculate PM 0 Release type (0-3)
 1 Option: 1-Use chi/Q or PM value F Stack release (T/F)
 2-Select MI dist & dir 0 Stack height (m)
 3-Specify MI dist & dir 0 Stack flow (m3/sec)
 4.46E-4 Chi/Q or PM value 0 Stack radius (m)
 0 MI sector index (1=S) 0 Effluent temp. (C)
 0 MI distance from release point (m) 0 Building x-section (m2)
 F Use jf data, (T/F) else chi/Q grid 0 Building height (m)

====SURFACE WATER TRANSPORT=====SECTION 2=====
 0 Mixing ratio model: 0-use value, 1-river, 2-lake
 0 Mixing ratio, dimensionless
 0 Average river flow rate for: MIXFLG=0 (m3/s), MIXFLG=1,2 (m/s),
 0 Transit time to irrigation withdrawal location (hr)
 0 If mixing ratio model > 0:
 0 Rate of effluent discharge to receiving water body (m3/s)
 0 Longshore distance from release point to usage location (m)
 0 Offshore distance to the water intake (m)
 0 Average water depth in surface water body (m)
 0 Average river width (m), MIXFLG=1 only
 0 Depth of effluent discharge point to surface water (m), lake only

====WASTE FORM AVAILABILITY=====SECTION 3=====
 0 Waste form/package half life, (yr)
 0 Waste thickness, (m)
 0 Depth of soil overburden, m

====TERRESTRIAL FOOD INGESTION=====SECTION 9=====

USE ? FOOD T/F TYPE	GROW TIME da	--IRRIGATION-- S RATE * in/yr	TIME mo/yr	YIELD kg/m2	PROD- UCTION kg/yr	--CONSUMPTION-- HOLDUP da	RATE kg/yr
T LEAF V	90.00	0	0.0	0.0	1.5	0.0E+00	30.0
T ROOT V	90.00	0	0.0	0.0	4.0	0.0E+00	220.0
T FRUIT	90.00	0	0.0	0.0	2.0	0.0E+00	330.0
T GRAIN	90.00	0	0.0	0.0	0.8	0.0E+00	80.0

====ANIMAL PRODUCTION CONSUMPTION=====SECTION 10=====

USE ? FOOD T/F TYPE	---HUMAN--- CONSUMPTION RATE kg/yr	TOTAL HOLDUP da	PROD- UCTION kg/yr	DRINK WATER CONSUMPTION FRACT.	DIET FRAC- TION	GROW TIME da	---IRRIGATION-- S RATE * in/yr	TIME mo/yr	YIELD kg/m3	STOR- AGE da
T BEEF	80.0	15.0	0.00	0.00	0.00	90.0	0	0.0	0.80	0.0
T POULTR	18.0	1.0	0.00	0.00	0.00	90.0	0	0.0	0.80	0.0
T MILK	270.0	1.0	0.00	0.00	0.00	45.0	0	0.0	2.00	0.0
T EGG	30.0	1.0	0.00	0.00	0.00	90.0	0	0.0	0.80	0.0
BEEF					0.00	45.0	0	0.0	2.00	100.0
MILK					0.00	30.0	0	0.0	1.50	0.0

#####

4.7.2.2 Input File for Onsite Receptor with Inhalation Only and 100% Suspension.

Input by: HJ Goldberg

Checked by: PD Rittmann

Program GENII Input File ##### 8 Jul 88 ####
 Title: N Reactor Characterization Shipments Packaging-Part 2- farmer
 \GENII\nreac-f2.in Created on 08-15-1994 at 13:30

```

OPTIONS===== Default =====
F Near-field scenario?      (Far-field)  NEAR-FIELD: narrowly-focused
F Population dose?          (Individual)  release, single site
T Acute release?            (Chronic)     FAR-FIELD: wide-scale release,
Maximum Individual data set used          multiple sites
Complete
TRANSPORT OPTIONS===== Section EXPOSURE PATHWAY OPTIONS===== Section
T Air Transport              1          F Finite plume, external          5
F Surface Water Transport    2          T Infinite plume, external        5
F Biotic Transport (near-field) 3,4    F Ground, external              5
F Waste Form Degradation (near) 3,4    F Recreation, external          5
T Inhalation uptake          5,6
T Drinking water ingestion   7,8
T Aquatic foods ingestion    7,8
T Terrestrial foods ingestion 7,9
T Animal product ingestion   7,10
T Inadvertent soil ingestion
REPORT OPTIONS=====
T Report AEDE only
F Report by radionuclide
F Report by exposure pathway
F Debug report on screen
  
```


INVENTORY

- 4 Inventory input activity units: (1-pCi 2-uCi 3-mCi 4-Ci 5-Bq)
0 Surface soil source units (1- m2 2- m3 3- kg)
Equilibrium question goes here

Use when	---Release Terms--- transport selected			-----Basic Concentrations----- near-field scenario, optionally				
	Air /yr	Surface Water /yr	Buried Waste /m3	Air /m3	Surface Soil /unit	Deep Soil /m3	Ground Water /L	Surface Water /L
Release Radio- nuclide								
H 3	2.2E+00							
FE55	1.7E+00							
CO60	5.7E-01							
NI63	9.0E-02							
KR85	3.8E+01							
SR90	3.6E+02							
Y 90	3.6E+02							
TC99	7.0E-02							
RU106	1.3E+02							
AG110M	5.0E-02							
CD113M	2.0E-01							
SN119M	9.0E-02							
SB125	2.8E+01							
TE125M	6.7E+00							
CS134	5.8E+01							
CS137	4.8E+02							
CE144	1.2E+02							
PR144	1.2E+02							
PR144M	1.4E+00							
PM147	3.6E+02							
SM151	4.3E+00							
EU152	6.0E-02							
EU154	1.2E+01							
EU155	4.0E+00							
U 235	6.5E-04							
PU238	4.9E+00							
PU239	5.8E+00							
PU240	4.1E+00							
PU241	4.7E+02							
AM241	4.5E+00							
CM244	1.5E+00							

Use when	-----Derived Concentrations----- measured values are known			
	Terres. Plant /kg	Animal Product /kg	Drink Water /L	Aquatic Food /kg
Release Radio- nuclide				

TIME #####

1 Intake ends after (yr)
 50 Dose calc. ends after (yr)
 0 Release ends after (yr)
 0 No. of years of air deposition prior to the intake period
 0 No. of years of irrigation water deposition prior to the intake period

FAR-FIELD SCENARIOS (IF POPULATION DOSE) #####

0 Definition option: 1-Use population grid in file POP.IN
 0 2-Use total entered on this line

NEAR-FIELD SCENARIOS #####

Prior to the beginning of the intake period: (yr)
 0 When was the inventory disposed? (Package degradation starts)
 0 When was LOIC? (Biotic transport starts)
 0 Fraction of roots in upper soil (top 15 cm)
 0 Fraction of roots in deep soil
 0 Manual redistribution: deep soil/surface soil dilution factor
 0 Source area for external dose modification factor (m2)

TRANSPORT #####

====AIR TRANSPORT=====SECTION 1=====

	0-Calculate PM	0	Release type (0-3)
1	Option: 1-Use chi/Q or PM value	F	Stack release (T/F)
	2-Select MI dist & dir	0	Stack height (m)
	3-Specify MI dist & dir	0	Stack flow (m3/sec)
1.0	Chi/Q or PM value	0	Stack radius (m)
0	MI sector index (1=S)	0	Effluent temp. (C)
0	MI distance from release point (m)	0	Building x-section (m2)
F	Use jf data, (T/F) else chi/Q grid	0	Building height (m)

====SURFACE WATER TRANSPORT=====SECTION 2=====

Mixing ratio model: 0-use value, 1-river, 2-lake
 0 Mixing ratio, dimensionless
 0 Average river flow rate for: MIXFLG=0 (m3/s), MIXFLG=1,2 (m/s),
 0 Transit time to irrigation withdrawal location (hr)
 If mixing ratio model > 0:
 0 Rate of effluent discharge to receiving water body (m3/s)
 0 Longshore distance from release point to usage location (m)
 0 Offshore distance to the water intake (m)
 0 Average water depth in surface water body (m)
 0 Average river width (m), MIXFLG=1 only
 0 Depth of effluent discharge point to surface water (m), lake only

====WASTE FORM AVAILABILITY=====SECTION 3=====

0 Waste form/package half life, (yr)
 0 Waste thickness, (m)
 0 Depth of soil overburden, m

```

=====BIOTIC TRANSPORT OF BURIED SOURCE=====SECTION 4=====
T Consider during inventory decay/buildup period (T/F)?
T Consider during intake period (T/F)?
0 Pre-Intake site condition.....
                                1-Arid non agricultural
                                2-Humid non agricultural
                                3-Agricultural

```

EXPOSURE #####

```

=====EXTERNAL EXPOSURE=====SECTION 5=====
Exposure time:
0 Plume (hr)
0 Soil contamination (hr)
0 Swimming (hr)
0 Boating (hr)
0 Shoreline activities (hr)
0 Shoreline type: (1-river, 2-lake, 3-ocean, 4-tidal basin)
0 Transit time for release to reach aquatic recreation (hr)
1.0 Average fraction of time submersed in acute cloud (hr/person hr)

Residential irrigation:
T Consider: (T/F)
0 Source: 1-ground water
2-surface water
0 Application rate (in/yr)
0 Duration (mo/yr)

```

```

=====INHALATION=====SECTION 6=====
8766.0 Hours of exposure to contamination per year
0 0-No resus- 1-Use Mass Loading 2-Use Anspaugh model
0 pension Mass loading factor (g/m3) Top soil available (cm)

```

```

=====INGESTION POPULATION=====SECTION 7=====
1 Atmospheric production definition (select option):
0 0-Use food-weighted chi/Q, (food-sec/m3), enter value on this line
1-Use population-weighted chi/Q
2-Use uniform production
3-Use chi/Q and production grids (PRODUCTION will be overridden)
0 Population ingesting aquatic foods, 0 defaults to total (person)
0.0 Population ingesting drinking water, 0 defaults to total (person)
F Consider dose from food exported out of region (default=F)

```

Note below: S* or Source: 0-none, 1-ground water, 2-surface water
3-Derived concentration entered above

===== AQUATIC FOODS / DRINKING WATER INGESTION=====SECTION 8=====

F Salt water? (default is fresh)

USE T/F	FOOD TYPE	TRAN- SIT hr	PROD- UCTION kg/yr	-CONSUMPTION- HOLDUP da	RATE kg/yr	DRINKING WATER	
T	FISH	0.00	0.0E+00	1.00	40.0	0	Source (see above)
T	MOLLUS	0.00	0.0E+00	0.00	6.9	T	Treatment? T/F
T	CRUSTA	0.00	0.0E+00	0.00	6.9	1.0	Holdup/transit(da)
T	PLANTS	0.00	0.0E+00	0.00	6.9	730.0	Consumption (L/yr)

FOOD INGESTION=====SECTION 9=====

=====TERRESTRIAL

USE ?	FOOD T/F TYPE	GROW TIME da	--IRRIGATION-- S RATE * in/yr	TIME mo/yr	YIELD kg/m2	PROD- UCTION kg/yr	--CONSUMPTION-- HOLDUP da	RATE kg/yr
T	LEAF V	90.00	0	0.0	0.0	1.5	0.0E+00	1.0
T	ROOT V	90.00	0	0.0	0.0	4.0	0.0E+00	5.0
T	FRUIT	90.00	0	0.0	0.0	2.0	0.0E+00	5.0
T	GRAIN	90.00	0	0.0	0.0	0.8	0.0E+00	180.0

=====ANIMAL PRODUCTION CONSUMPTION=====SECTION 10=====

USE ?	FOOD T/F TYPE	---HUMAN--- CONSUMPTION RATE kg/yr	TOTAL HOLDUP da	PROD- UCTION kg/yr	DRINK WATER CONTAM FRACT.	DIET FRAC- TION	GROW TIME da	--IRRIGATION-- S RATE * in/yr	TIME mo/yr	STOR- FEED YIELD kg/m3	AGE da
T	BEEF	80.0	15.0	0.00	0.00	0.00	90.0	0	0.0	0.00	0.80
T	POULTR	18.0	1.0	0.00	0.00	0.00	90.0	0	0.0	0.00	0.80
T	MILK	270.0	1.0	0.00	0.00	0.00	45.0	0	0.0	0.00	2.00
T	EGG	30.0	1.0	0.00	0.00	0.00	90.0	0	0.0	0.00	0.80
	BEEF					0.00	45.0	0	0.0	0.00	2.00
	MILK					0.00	30.0	0	0.0	0.00	1.50

#####

4.7.2.3 Output File for Offsite Receptor with Inhalation, Ingestion and 100% Suspension.

 GENII Dose Calculation Program
 (Version 1.485 3-Dec-90)

Case title: N-Reactor Characterization Shipment Accident in 237 Building
 -farmer

Executed on: 07/22/94 at 14:10:47

Page A. 1

 This is a far-field (wide-scale release, multiple site) scenario.
 Release is acute
 Individual dose

THE FOLLOWING TRANSPORT MODES ARE CONSIDERED
 Air

THE FOLLOWING EXPOSURE PATHS ARE CONSIDERED:
 Infinite plume, external
 Inhalation uptake
 Drinking water ingestion
 Aquatic foods ingestion
 Terrestrial foods ingestion
 Animal product ingestion

THE FOLLOWING TIMES ARE USED:
 Intake ends after (yr): 1.0
 Dose calculations ends after (yr): 50.0

===== FILENAMES AND TITLES OF FILES/LIBRARIES USED =====

Input file name: \GENII\nreact-f.in
 GENII Default Parameter Values (28-Mar-90 RAP)
 Radionuclide Library - Times<100 years (23-July-93 PDR)
 PNL Food Transfer Factor Library (by Z, with Fr&Os 7/19/93 PDR)
 Bioaccumulation Factor Library - (30-Aug-88) RAP
 External Dose Factors for GENII in person Sv/yr per Bq/n (8-May-90 R
 Worst-Case Solubilities, Yearly Dose Increments (23-Jul-93 PDR)

-----Release Terms-----

Release Radio- nuclide	Air Ci/yr	Water Ci/yr	Surface Buried Source Ci/m3
H 3	2.2E+00	0.0E+00	0.0E+00
KR85	3.8E+01	0.0E+00	0.0E+00
SR90	3.6E+02	0.0E+00	0.0E+00
Y 90	3.6E+02	0.0E+00	0.0E+00
TC99	7.0E-02	0.0E+00	0.0E+00
RU106	1.3E+02	0.0E+00	0.0E+00
AG110M	5.0E-02	0.0E+00	0.0E+00
CD113M	2.0E-01	0.0E+00	0.0E+00
SB125	2.7E+01	0.0E+00	0.0E+00
TE125M	6.5E+00	0.0E+00	0.0E+00
CS134	5.8E+01	0.0E+00	0.0E+00
CS137	4.8E+02	0.0E+00	0.0E+00
CE144	1.2E+02	0.0E+00	0.0E+00
PR144	1.2E+02	0.0E+00	0.0E+00
PR144M	1.4E+00	0.0E+00	0.0E+00
PM147	3.6E+02	0.0E+00	0.0E+00
EU152	6.0E-02	0.0E+00	0.0E+00
EU154	1.2E+01	0.0E+00	0.0E+00
EU155	4.0E+00	0.0E+00	0.0E+00

U 235	6.5E-04	0.0E+00	0.0E+00
U 238	1.4E-02	0.0E+00	0.0E+00
PU238	4.9E+00	0.0E+00	0.0E+00
PU239	5.8E+00	0.0E+00	0.0E+00
PU240	4.1E+00	0.0E+00	0.0E+00
PU241	4.7E+02	0.0E+00	0.0E+00
AM241	4.5E+00	0.0E+00	0.0E+00
CM244	1.5E+00	0.0E+00	0.0E+00

===== AIR TRANSPORT =====
 4.5E-04 Input E/Q value (s/m³)

===== EXTERNAL EXPOSURE =====
 1.0E+00 Fraction of time spent in cloud

===== INHALATION =====
 Resuspension not considered

===== INGESTION POPULATION =====
 1 Atmospheric production definition: 1 - Use population-weighted chi/Q

===== DRINKING WATER SOURCE/IRRIGATION =====
 7.3E+02 Drinking water consumption rate (l/yr)
 0 Drinking water source: 1-ground, 2-surface, 3-system
 T Drinking water treatment: T/F
 1.0 Drinking water transit/holdup time (d)

===== AQUATIC FOODS INGESTION =====

FOOD TYPE	TRAN- SIT h	PROD- UCTION kg/yr	----CONSUMPTION----
			HOLDUP d RATE kg/yr
Fish	0.00E+00		1.00E+00 4.00E+01
Mollusc	0.00E+00		0.00E+00 6.90E+00
Crustace	0.00E+00		0.00E+00 6.90E+00
Aqu Plnt	0.00E+00		0.00E+00 6.90E+00

===== TERRESTRIAL FOOD INGESTION =====

FOOD TYPE	GROW TIME d	--IRRIGATION-- S RATE * in/yr	TIME mo/yr	YIELD kg/m ²	PROD- UCTION kg/yr	--CONSUMPTION-- HOLDUP d RATE kg/yr
Leaf Veg	90.0	0	0.0	1.5		1.0 3.0E+01
Oth. Veg	90.0	0	0.0	4.0		5.0 2.2E+02
Fruit	90.0	0	0.0	2.0		5.0 3.3E+02
Cereals	90.0	0	0.0	0.8		180.0 8.0E+01

===== ANIMAL FOOD INGESTION =====

FOOD TYPE	---HUMAN---	TOTAL CONSUMPTION kg/yr	PROD- HOLDUP d	DRINK WATER kg/yr	DIET FRAC- TION	GROW TIME d	--IRRIGATION-- S RATE * in/yr	TIME mo/yr	STOR- YIELD kg/m ³	AGE d
Meat	8.0E+01	15.0		0.00		90.00	0	0.0	0.0	0.80
Poultry	1.8E+01	1.0		0.00		90.00	0	0.0	0.0	0.80
Cow Milk	2.7E+02	1.0		0.00		45.00	0	0.0	0.0	2.00
Eggs	3.0E+01	1.0		0.00		90.00	0	0.0	0.0	0.80

	-----FRESH FORAGE-----				
Meat	45.00	0	0.0	0.0	2.00
Cow Milk	30.00	0	0.0	0.0	1.50

GENII Dose Calculation Program
(Version 1.485 3-Dec-90)

Case title: Autumn: N-Reactor Characterization Shipment Accident in 237
Building-farmer

Executed on: 07/22/94 at 14:11:52

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Acute release
Uptake/exposure period: 1.0
Dose commitment period: 50.0
Dose units: Rem

Organ	Committed Dose Equivalent	Weighting Factors	Weighted Dose Equivalent
Gonads	2.4E+03	2.5E-01	6.0E+02
Breast	1.4E+03	1.5E-01	2.1E+02
R Marrow	9.6E+03	1.2E-01	1.2E+03
Lung	1.8E+03	1.2E-01	2.1E+02
Thyroid	1.4E+03	3.0E-02	4.2E+01
Bone Sur	6.8E+04	3.0E-02	2.0E+03
Liver	1.0E+04	6.0E-02	6.0E+02
LL Int.	3.3E+03	6.0E-02	2.0E+02
UL Int.	2.1E+03	6.0E-02	1.3E+02
S Int.	1.7E+03	6.0E-02	1.0E+02
Stomach	1.6E+03	6.0E-02	9.5E+01

Internal Effective Dose Equivalent 5.4E+03
External Dose 4.0E-02

Annual Effective Dose Equivalent 5.4E+03

Controlling Organ: Bone Sur
Controlling Pathway: Ing
Controlling Radionuclide: CS137

Total Inhalation EDE: 1.9E+03
Total Ingestion EDE: 3.5E+03

GENII Dose Calculation Program
(Version 1.485 3-Dec-90)

Case title: Autumn: N-Reactor Characterization Shipment Accident in 237
Building-farmer

Executed on: 07/22/94 at 14:11:52

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Acute release

Uptake/exposure period:

1.0

Dose commitment period:

50.0

Dose units:

Rem

	Dose Commitment Year				
	1	2	3	...	
Internal Intake Year:					
3			0.0E+00	...	
			+		
2		0.0E+00	0.0E+00	...	Internal Effective Dose Equivalent
		+	+		
1	1.6E+03	+ 2.7E+02	+ 1.6E+02	+ ... = 5.4E+03	
Internal Annual Dose	1.6E+03	+ 2.7E+02	+ 1.6E+02	+ ... = 5.4E+03	Cumulative Internal Dose
	+	+	+	+	
External Annual Dose	4.0E-02	0.0E+00	0.0E+00	... 4.0E-02	
Annual Dose	1.6E+03	+ 2.7E+02	+ 1.6E+02	+ ... = 5.4E+03	Cumulative Dose
				1.6E+03	Maximum Annual Dose Occurred In Year 1

GENII Dose Calculation Program
(Version 1.485 3-Dec-90)

Case title: Autumn: N-Reactor Characterization Shipment Accident in 237
Building-farmer

Executed on: 07/22/94 at 14:11:52

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Acute release
Uptake/exposure period: 1.0
Dose commitment period: 50.0
Dose units: Rem

Radio-nuclide	Inhalation Effective Dose Equivalent	Ingestion Effective Dose Equivalent	External Dose	Internal Effective Dose Equivalent	Annual Effective Dose Equivalent
H 3	2.9E-05	2.7E-03	2.2E-14	2.7E-03	2.7E-03
KR 85	0.0E+00	0.0E+00	9.7E-06	0.0E+00	9.7E-06
SR 90	1.1E+01	7.0E+02	6.6E-06	7.1E+02	7.1E+02
Y 90	4.6E-01	5.4E+01	2.4E-04	5.5E+01	5.5E+01
TC 99	9.0E-05	2.3E-03	2.4E-10	2.4E-03	2.4E-03
RU 106	8.9E+00	3.6E+01	2.8E-03	4.5E+01	4.5E+01
AG 110M	2.8E-03	7.1E-02	1.3E-05	7.4E-02	7.4E-02
CD 113M	4.5E-02	4.2E-01	4.2E-09	4.7E-01	4.7E-01
SB 125	5.1E-02	1.0E+00	1.2E-03	1.1E+00	1.1E+00
TE 125M	6.9E-03	4.1E-01	8.1E-06	4.2E-01	4.2E-01
CS 134	3.6E-01	1.8E+02	8.9E-03	1.8E+02	1.8E+02
CS 137	2.1E+00	1.2E+03	2.5E-02	1.2E+03	1.2E+03
CE 144	6.7E+00	2.3E+01	1.5E-04	3.0E+01	3.0E+01
PR 144M	3.9E-06	7.6E-04	7.6E-07	7.6E-04	7.6E-04
PR 144	8.1E-04	1.3E-01	5.0E-04	1.3E-01	1.3E-01
PM 147	2.1E+00	5.6E+00	6.9E-07	7.6E+00	7.6E+00
EU 152	1.8E-03	6.8E-03	6.5E-06	8.6E-03	8.7E-03
EU 154	5.0E-01	2.0E+00	1.3E-03	2.5E+00	2.5E+00
EU 155	2.4E-02	1.0E-01	1.5E-05	1.3E-01	1.3E-01
CM 244	5.5E+01	5.0E+01	1.8E-08	1.0E+02	1.0E+02
PU 240	2.6E+02	1.9E+02	5.6E-08	4.5E+02	4.5E+02
PU 238	2.8E+02	2.0E+02	6.9E-08	4.8E+02	4.8E+02
U 238	2.5E-01	5.2E-02	0.0E+00	3.0E-01	3.0E-01
TH 234	0.0E+00	2.7E-03	0.0E+00	2.7E-03	2.7E-03
PA 234	0.0E+00	7.2E-07	0.0E+00	7.2E-07	7.2E-07
PU 241	5.6E+02	4.0E+02	1.6E-11	9.6E+02	9.6E+02
U 237	0.0E+00	4.9E-04	0.0E+00	4.9E-04	4.9E-04
AM 241	2.9E+02	2.3E+02	4.4E-06	5.2E+02	5.2E+02
PU 239	3.7E+02	2.6E+02	5.5E-08	6.3E+02	6.3E+02
U 235	1.2E-02	2.5E-03	0.0E+00	1.4E-02	1.4E-02
TH 231	0.0E+00	1.2E-05	0.0E+00	1.2E-05	1.2E-05

4.7.2.4 Output File for Onsite Receptor with Inhalation and 100% Suspension.

 GENII Dose Calculation Program
 (Version 1.485 3-Dec-90)

Case title: N Reactor Characterization Shipments Packaging-Part 2- worke
 r

Executed on: 08/15/94 at 13:34:20

Page A. 1

This is a far-field (wide-scale release, multiple site) scenario.
 Release is acute
 Individual dose

THE FOLLOWING TRANSPORT MODES ARE CONSIDERED
 Air

THE FOLLOWING EXPOSURE PATHS ARE CONSIDERED:
 Infinite plume, external
 Inhalation uptake

THE FOLLOWING TIMES ARE USED:
 Intake ends after (yr): 1.0
 Dose calculations ends after (yr): 50.0

===== FILENAMES AND TITLES OF FILES/LIBRARIES USED =====

Input file name: \GENII\nreac-w2.in
 GENII Default Parameter Values (28-Mar-90 RAP)
 Radionuclide Library - Times<100 years (23-July-93 PDR)
 External Dose Factors for GENII in person Sv/yr per Bq/n (8-May-90 R
 Worst-Case Solubilities, Yearly Dose Increments (23-Jul-93 PDR)

=====

-----Release Terms-----

Release	Surface Buried		
Radio- Air	Water	Source	
nuclide	Ci/yr	Ci/yr	Ci/m3
H 3	2.2E+00	0.0E+00	0.0E+00
FE55	1.7E+00	0.0E+00	0.0E+00
CO60	5.7E-01	0.0E+00	0.0E+00
NI63	9.0E-02	0.0E+00	0.0E+00
KR85	3.8E+01	0.0E+00	0.0E+00
SR90	3.6E+02	0.0E+00	0.0E+00
Y 90	3.6E+02	0.0E+00	0.0E+00
TC99	7.0E-02	0.0E+00	0.0E+00
RU106	1.3E+02	0.0E+00	0.0E+00
AG110M	5.0E-02	0.0E+00	0.0E+00
CD113M	2.0E-01	0.0E+00	0.0E+00
SN119M	9.0E-02	0.0E+00	0.0E+00
SB125	2.8E+01	0.0E+00	0.0E+00
TE125M	6.7E+00	0.0E+00	0.0E+00
CS134	5.8E+01	0.0E+00	0.0E+00
CS137	4.8E+02	0.0E+00	0.0E+00
CE144	1.2E+02	0.0E+00	0.0E+00
PR144	1.2E+02	0.0E+00	0.0E+00
PR144M	1.4E+00	0.0E+00	0.0E+00
PM147	3.6E+02	0.0E+00	0.0E+00
SM151	4.3E+00	0.0E+00	0.0E+00
EU152	6.0E-02	0.0E+00	0.0E+00
EU154	1.2E+01	0.0E+00	0.0E+00
EU155	4.0E+00	0.0E+00	0.0E+00
U 235	6.5E-04	0.0E+00	0.0E+00
PU238	4.9E+00	0.0E+00	0.0E+00
PU239	5.8E+00	0.0E+00	0.0E+00
PU240	4.1E+00	0.0E+00	0.0E+00

PU241	4.7E+02	0.0E+00	0.0E+00
AM241	4.5E+00	0.0E+00	0.0E+00
CM244	1.5E+00	0.0E+00	0.0E+00

```

===== AIR TRANSPORT =====
1.0E+00 Input E/Q value (s/m3)

===== EXTERNAL EXPOSURE =====
1.0E+00 Fraction of time spent in cloud

===== INHALATION =====
Resuspension not considered
=====
    
```

GENII Dose Calculation Program
(Version 1.485 3-Dec-90)

Case title: N Reactor Characterization Shipments Packaging-Part 2- worke

Executed on: 08/15/94 at 13:34:48

Page C. 1

Acute release
Uptake/exposure period: 1.0
Dose commitment period: 50.0
Dose units: Rem

Organ	Committed Dose Equivalent	Weighting Factors	Weighted Dose Equivalent
Gonads	1.1E+06	2.5E-01	2.8E+05
Breast	6.3E+03	1.5E-01	9.5E+02
R Marrow	6.2E+06	1.2E-01	7.4E+05
Lung	7.8E+05	1.2E-01	9.4E+04
Thyroid	6.1E+03	3.0E-02	1.8E+02
Bone Sur	7.7E+07	3.0E-02	2.3E+06
Liver	1.3E+07	6.0E-02	7.9E+05
LL Int.	2.6E+04	6.0E-02	1.6E+03
UL Int.	1.4E+04	6.0E-02	8.2E+02
S Int.	8.3E+03	6.0E-02	5.0E+02
Stomach	7.4E+03	6.0E-02	4.4E+02
Internal Effective Dose Equivalent			4.2E+06
External Dose			8.9E+01
Annual Effective Dose Equivalent			4.2E+06

Controlling Organ: Bone Sur
Controlling Pathway: Inh
Controlling Radionuclide: PU241

Total Inhalation EDE: 4.2E+06
Total Ingestion EDE: 0.0E+00

 GENII Dose Calculation Program
 (Version 1.485 3-Dec-90)

Case title: N Reactor Characterization Shipments Packaging-Part 2- worke

Executed on: 08/15/94 at 13:34:48

Page C. 2

Acute release

Uptake/exposure period:

1.0

Dose commitment period:

50.0

Dose units:

Rem

		Dose Commitment Year				
		1	2	3	...	
Internal Intake Year:	3			0.0E+00	...	
				+		
	2		0.0E+00	0.0E+00	...	Internal Effective Dose Equivalent
			+	+		
	1	1.7E+05	+ 1.1E+05	+ 1.0E+05	+ ... = 4.2E+06	
Internal Annual Dose		1.7E+05	+ 1.1E+05	+ 1.0E+05	+ ... = 4.2E+06	Cumulative Internal Dose
		+	+	+	+	
External Annual Dose		8.9E+01	0.0E+00	0.0E+00	... = 8.9E+01	
Annual Dose		1.7E+05	+ 1.1E+05	+ 1.0E+05	+ ... = 4.2E+06	Cumulative Dose
					1.7E+05	Maximum Annual Dose Occurred In Year 1

 GENII Dose Calculation Program
 (Version 1.485 3-Dec-90)

Case title: N Reactor Characterization Shipments Packaging-Part 2- worke
 r

Executed on: 08/15/94 at 13:34:48

Page C. 3

Acute release
 Uptake/exposure period: 1.0
 Dose commitment period: 50.0
 Dose units: Rem

Radio-nuclide	Inhalation Effective Dose Equivalent	Ingestion Effective Dose Equivalent	External Dose	Internal Effective Dose Equivalent	Annual Effective Dose Equivalent
H 3	6.6E-02	0.0E+00	5.1E-11	6.6E-02	6.6E-02
FE 55	1.5E+00	0.0E+00	1.2E-04	1.5E+00	1.5E+00
CO 60	3.8E+01	0.0E+00	2.9E-01	3.8E+01	3.9E+01
NI 63	9.1E-02	0.0E+00	7.0E-09	9.1E-02	9.1E-02
KR 85	0.0E+00	0.0E+00	2.1E-02	0.0E+00	2.1E-02
SR 90	2.4E+04	0.0E+00	1.4E-02	2.4E+04	2.4E+04
Y 90	1.0E+03	0.0E+00	5.2E-01	1.0E+03	1.0E+03
TC 99	2.1E-01	0.0E+00	5.4E-07	2.1E-01	2.1E-01
RU 106	2.0E+04	0.0E+00	6.3E+00	2.0E+04	2.0E+04
AG 110M	6.4E+00	0.0E+00	2.9E-02	6.4E+00	6.4E+00
CD 113M	1.0E+02	0.0E+00	9.6E-06	1.0E+02	1.0E+02
SN 119M	1.9E-01	0.0E+00	7.7E-05	1.9E-01	1.9E-01
SB 125	1.2E+02	0.0E+00	2.8E+00	1.2E+02	1.2E+02
TE 125M	1.6E+01	0.0E+00	1.9E-02	1.6E+01	1.6E+01
CS 134	8.1E+02	0.0E+00	2.0E+01	8.1E+02	8.3E+02
CS 137	4.8E+03	0.0E+00	5.5E+01	4.8E+03	4.9E+03
CE 144	1.5E+04	0.0E+00	3.4E-01	1.5E+04	1.5E+04
PR 144M	8.6E-03	0.0E+00	1.7E-03	8.6E-03	1.0E-02
PR 144	1.8E+00	0.0E+00	1.1E+00	1.8E+00	2.9E+00
PM 147	4.7E+03	0.0E+00	1.5E-03	4.7E+03	4.7E+03
SM 151	4.2E+01	0.0E+00	4.2E-06	4.2E+01	4.2E+01
EU 152	4.2E+00	0.0E+00	1.5E-02	4.2E+00	4.2E+00
EU 154	1.1E+03	0.0E+00	3.0E+00	1.1E+03	1.1E+03
EU 155	5.3E+01	0.0E+00	3.5E-02	5.3E+01	5.3E+01
CM 244	1.2E+05	0.0E+00	4.2E-05	1.2E+05	1.2E+05
PU 240	6.1E+05	0.0E+00	1.3E-04	6.1E+05	6.1E+05
PU 238	6.3E+05	0.0E+00	1.6E-04	6.3E+05	6.3E+05
PU 241	1.3E+06	0.0E+00	3.6E-08	1.3E+06	1.3E+06
U 237	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
AM 241	6.6E+05	0.0E+00	9.7E-03	6.6E+05	6.6E+05
PU 239	8.3E+05	0.0E+00	1.2E-04	8.3E+05	8.3E+05
U 235	2.6E+01	0.0E+00	1.3E-05	2.6E+01	2.6E+01
TH 231	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00

5.0 SHIELDING EVALUATION

5.1 INTRODUCTION

The SARP for the CNS 1-13G (ChemNuclear 1992) does not specifically address the shielding performance of the package with irradiated fuel elements. Instead, the SARP demonstrated that the cask would remain in compliance with the shielding requirements of 10 CFR 71 with a payload of 39,000 Ci of ^{60}Co and a special cask liner. To authorize the transfer of CNS 1-13G within the boundaries of the Hanford Site, a shielding analysis based on the SARP analysis was performed assuming a payload of irradiated N Reactor/SPR fuel without the additional shielding provided by a liner.

5.2 SUMMARY OF RESULTS

The highest surface dose anticipated during normal conditions of transport and accident conditions for the CNS 1-13G transporting irradiated N Reactor/SPR fuel is 0.063 mSv/h (6.3 mrem/h) which is less than the 2.00 mSv/h (200 mrem/h) limit for normal conditions of transport. The maximum dose anticipated for persons occupying the transfer vehicle is 0.00030 mSv/h (0.03 mrem/h), which is significantly less than the 0.02 mSv/h (2 mrem/h) limit.

5.3 DIRECT RADIATION SOURCE SPECIFICATION

The worst case irradiated fuel for gamma, beta and neutron radiation is represented by 16% ^{240}Pu irradiated fuel that has been in storage for 5 years. This is conservative as 16% ^{240}Pu is rare and the minimum amount of time that fuel has decayed is seven years. Due to criticality limitations (Part B, Section 6.0), the maximum fissile content for a single shipment is limited to 500 g of ^{235}U equivalent. Table B5-1 contains the activity of individual isotopes for a 500 g ^{235}U equivalent worst case payload.

5.4 SUMMARY OF SHIELDING PROPERTIES OF MATERIALS

The dose rate at a point from a given source is a function of the distance from source to that point and the materials in between. In general, the ability of a material to shield gamma rays is a function of density, whereas the ability to stop neutrons is a function of the atomic number (i.e., carbon is a better neutron shield than lead). In addition, as separation distance increases, the dose rate drops as the square of the distance.

Shielding for the CNS 1-13G cask is provided a minimum 5 in. thick lead shield contained between two $\frac{1}{2}$ in. layers of stainless steel. The carbon steel overpack provides additional shielding through an equivalent of 1 in. of steel.

Table B5-1. Activity of Worst Case CNS 1-13G Cask Payload.

Specie	Maximum Payload (Ci)	Specie	Maximum Payload (Ci)
$^{90}\text{Sr}/^{90}\text{Y}$	358.9	^{152}Eu	0.06
^{99}Tc	0.07	$^{110\text{m}}\text{Ag}$	0.05
$^{106}\text{Ru}/^{106}\text{Rh}$	130.9	^{238}Pu	4.94
$^{113\text{m}}\text{Cd}$	0.20	^{239}Pu	5.80
$^{125\text{m}}\text{Te}$	6.50	^{240}Pu	4.11
^{134}Cs	58.1	^{241}Pu	465.7
$^{137}\text{Cs}/^{137\text{m}}\text{Ba}$	480.0	^{241}Am	4.48
$^{144}\text{Ce}/^{144}\text{Pr}$	116.9	^{244}Cm	1.45
^{147}Pm	358.2	^{55}Fe	1.71
^{151}Sm	4.26	^{60}Co	0.57
^{154}Eu	12.3	^{125}Sb	0.93
^{155}Eu	3.95	$^{125\text{m}}\text{Te}$	0.23
^{85}Kr	38.4	$^{119\text{m}}\text{Sn}$	0.09
^{125}Sb	26.6	^{63}Ni	0.09
^3H	2.17	^{235}U	6.46×10^{-4}
$^{144\text{m}}\text{Pr}$	1.40		

5.5 SHIELDING EVALUATION FOR NORMAL TRANSFER CONDITIONS

5.5.1 Acceptance Criteria

The following dose rate limits for normal conditions of transport were established by the *Packaging Design Criteria for the N Reactor Fuel Characterization Shipments* (WHC-SD-TP-PDC-022) for authorized transport of the CNS 1-13G cask.

1. When loaded with the worst case payload, the exterior surface of the transfer cask shall be surveyed to insure that the maximum dose not exceed 2 mSv/h (200 mrem/h).
2. The allowable dose rate at 2 m (6.6 ft) from any surface of the transfer cask shall not exceed 0.10 mSv/h (10 mrem/h).
3. The maximum dose rate in any normally occupied portion of the transfer vehicle, such as the cab, shall not exceed 0.02 mSv/h (2 mrem/h).

5.5.2 Shielding Model

To evaluate the maximum direct radiation doses associated with the transport of the CNS 1-13G cask, the ISOSHL D computer code was used. The ISOSHL D code utilizes extensive material property, attenuation and isotopic decay libraries to allow direct doses to be calculated with only composition and the geometry of the source and shielding being specified. ISOSHL D has been extensively verified and documented under the Westinghouse Hanford Quality Assurance Program (Lourant 1989).

Figure B5-1 shows the model configuration for the ISOSHL D calculations. The following assumptions were made for the ISOSHL D calculations:

1. The source was in the form of a sphere 17.12 cm (6.74 in.) in diameter, which has the same volume that the worst case payload would occupy.
2. A 1 in. void space was assumed between the source and the inner cask surface. This is equal to the gap between the cask and overpack.
3. The gaps between the fuel and the cask and within the overpack were conservatively neglected.
4. Inner surface of the cask modeled as 1.27 cm (0.5 in.) thick steel
5. Lead shield modeled as 12.7 cm (5 in.) of lead
6. Outer cask surface and overpack modeled as a single 3.81 cm (1.5 in.) layer of steel.
7. Occupants of the transfer vehicle are located 4.6 m (15 ft) from the cask surface.
8. Transfer vehicle will provide negligible shielding.

The input files for the ISOSHL D calculations are in Part B, Appendix 5.7.2.

Figure B5-1. ISOSHLD Model Geometry for the CNS 1-13G Cask Transporting Irradiated N Reactor/SPR Fuel.

5.5.3 Shielding Evaluation

Table B5-2 presents the maximum dose rates for the worst case payload under normal conditions of transport. The doses are well below the limits set by the acceptance criteria.

Table B5-2. Maximum Normal Condition Dose Rates for CNS 1-13G Cask.

Maximum Normal Condition Dose Rate	Dose (mSv/h)
Surface of Package	0.063
2 m from Package Surface	0.0014
3.3 m from Surface	0.00055
4.6 m from Package Surface (Vehicle Occupants)	0.00030

5.6 SHIELDING EVALUATION FOR ACCIDENT CONDITIONS

The SARP for the CNS 1-13G cask (ChemNuclear 1992) indicates that after an accident event, the geometry of the CNS 1-13G cask would not be appreciably affected. Therefore, the shielding capability of the cask after an accident event will be identical to its shielding capability during normal conditions of transport. The only shielding requirement established for the CNS 1-13G cask subsequent to an accident (WHC 1994) is that the dose 1 m from the cask be less than 10 mSv/h (1,000 mrem/h). This criteria is met as the cask surface dose will be less than 0.10 mSv/h (10 mrem/h).

5.7 APPENDICES

5.7.1 References

- ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()*F, ChemNuclear, Inc, Columbia, South Carolina.
- Lourant, I. O., 1989, *Validation of ISO SHLD-PC on the IBM PC(PS-2/AT/XT)*, SDSQA-CSWD-303, Rev 0, Westinghouse Hanford Co, Richland, Washington.
- WHC, 1994, *Packaging Design Criteria for the N Reactor/Single Pass Reactor Fuel Characterization Shipments*, WHC-SD-TP-PDC-022, Westinghouse Hanford Company, Richland, Washington.

5.7.2 ISOSHLD Input Files

Input by: RJ Morford
 Checked by: FA Schmittroth

```

0      2  CHEM NUC 1-13G CSK,N-REAC,K-BASIN,16%PU240, SURF DOSE
      CHEM NUC 1-13G CSK,N-REAC,K-BASIN,16%PU240, SURF DOSE
&INPUT
      NEXT= 1,  ISPEC= 3,OPTION=1,
      IGEOM= 4,  X= 32.056,
      NSHLD= 5,  JBUF= 5,
      T(1)=8.56,T(2)=5.715,T(3)=1.27,T(4)=12.7,T(5)=3.81,
      WEIGHT(82)=358.8,WEIGHT(84)=358.8,WEIGHT(170)=103.9,
      WEIGHT(172)=103.9,WEIGHT(319)=58.0,WEIGHT(335)=479.9,
      WEIGHT(336)=479.9,WEIGHT(376)=116.9,WEIGHT(377)=116.9,
      WEIGHT(415)=12.25,WEIGHT(55)=38.4,WEIGHT(269)=27.53,
      WEIGHT(408)=0.06,WEIGHT(191)=0.05,WEIGHT(492)=4.93,
      WEIGHT(493)=5.80,WEIGHT(494)=4.11,WEIGHT(495)=465.6,
      WEIGHT(496)=4.48,WEIGHT(500)=1.45,WEIGHT(472)=0.57,
&
      FE      9      0.0      0.0      7.8      0.0      7.8
      1PB     14     19.      0.0      0.0     11.3      0.0
2 METERS FROM SURFACE OF PACKAGE
&INPUT NEXT=4, X=232.056 &
330 CM FROM SURFACE OF PACKAGE
&INPUT X=362. &
460 CM FROM SURFACE (15 FT)-DRIVER POS
&INPUT X=492.0 &
Last case
&INPUT NEXT=6 &

```

```

0      2  PT SRC,CHEM NUC 1-13G CSK,N-REAC,K-BASIN,16%PU240, SURF DOSE
      PT SRC, CHEM NUC 1-13G CSK,N-REAC,K-BASIN,16%PU240, SURF DOSE
&INPUT
      NEXT= 1,  ISPEC= 3,OPTION=1,
      IGEOM= 4,  X= 32.056,
      NSHLD= 5,  JBUF= 5,
      T(1)=0.01,T(2)=5.715,T(3)=1.27,T(4)=12.7,T(5)=3.81,
      WEIGHT(82)=358.8,WEIGHT(84)=358.8,WEIGHT(170)=103.9,
      WEIGHT(172)=103.9,WEIGHT(319)=58.0,WEIGHT(335)=479.9,
      WEIGHT(336)=479.9,WEIGHT(376)=116.9,WEIGHT(377)=116.9,
      WEIGHT(415)=12.25,WEIGHT(55)=38.4,WEIGHT(269)=27.53,
      WEIGHT(408)=0.06,WEIGHT(191)=0.05,WEIGHT(492)=4.93,
      WEIGHT(493)=5.80,WEIGHT(494)=4.11,WEIGHT(495)=465.6,
      WEIGHT(496)=4.48,WEIGHT(500)=1.45,WEIGHT(472)=0.57,
&
      FE      9      0.0      0.0      7.8      0.0      7.8
      1PB     14     19.      0.0      0.0     11.3      0.0
2 METERS FROM SURFACE OF PACKAGE
&INPUT NEXT=4, X=232.056 &
330 CM FROM SURFACE OF PACKAGE
&INPUT X=362. &
460 CM FROM SURFACE (15 FT)-DRIVER POS
&INPUT X=492.0 &
Last case
&INPUT NEXT=6 &

```

6.0 CRITICALITY EVALUATION

6.1 SUMMARY OF RESULTS

The amount of N Reactor and SPR fuel that is to be shipped in the CNS 1-13G shall be limited so that the total amount of fissile material shall be less than 500 g ^{235}U equivalent. This is in compliance with the CoC and the CNS 1-13G SARP (ChemNuclear 1992). Additionally, it is less than the minimum critical mass for optimally configured ^{235}U as determined by *Critical Dimensions of Systems Containing ^{235}U , ^{239}Pu and ^{233}U* (LANL 1986). Therefore, no criticality analysis is required to ship the cask onsite.

6.2 CRITICALITY EVALUATION OF PACKAGING

The fissile limit in the CoC and for this shipment is limited to 500 g ^{235}U equivalent. The appropriateness of this fissile limit for N Reactor and SPR fuel has been evaluated (Part B, Section 6.3.2). The maximum quantity of fissile material will occur in unirradiated (green) fuel. For unirradiated fuel, nearly all of the fissile material will be ^{235}U . Table B6-1 contains the maximum amount of fissile material for N Reactor Mark 1A and Mark IV fuel. SPR fuel is considerably smaller than N Reactor Fuel and has correspondingly less fissile material.

Table B6-1. ^{235}U Content for CNS 1-13G Payloads.

Element Type	Weight (g)	Enrichment	^{235}U Weight (g)
Mark 1A Inner	5,993	0.95%	57.0
Mark 1A Outer	12,786	1.25%	159.8
Mark IV Inner	7,484	0.95%	71.1
Mark IV Outer	15,967	0.95%	151.7

The CNS 1-13G may transport up to two inner and two outer elements in one shipment or up to three outer elements regardless if they are Mark 1A or Mark IV.

6.3 APPENDICES

6.3.1 References

ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()*F, ChemNuclear, Inc, Columbia, South Carolina.

LANL, 1986, *Critical Dimensions of Systems Containing ^{235}U , ^{239}Pu and ^{233}U* , LA-10860, Los Alamos National Laboratories, Albuquerque, New Mexico.

6.3.2 Criticality Review

Westinghouse
Hanford Company

Internal
Memo

From: Nuclear Analysis and Characterization
Phone: 376-4146 H0-38
Date: September 8, 1994
Subject: CRITICALITY REVIEW OF CNS 1-13G SHIPPING CASK CERTIFICATE OF COMPLIANCE FOR USE WITH N REACTOR FUEL

To: R. J. Smith G2-02

cc: D. W. Bergmann R3-86
J. G. Field G2-02
D. E. Friar R3-01
P. F. Stevens G2-02
H. Toffer H0-38
DGE/File-LB 9407

- References:
1. K. N. Schwinkendorf, 1994, WHC-SD-NR-CSER-010 Rev. 0, *Criticality Safety Evaluation Report for 300 Area N Reactor Fuel Fabrication and Storage Facility*, Westinghouse Hanford Company, Richland, Washington.
 2. W. D. Wittekind, April 1994, WHC-SD-NR-ANAL-014 Rev. 0, *Consolidated Fuel Decay Heat Calculations*, Westinghouse Hanford Company, Richland, Washington.
 3. H. Toffer, April 1976, UNI-283-ADD1, *Nuclear Criticality Safety Analysis and Technical Basis for the Storage of N Reactor Fuels in K Reactor Basins - 1900 Ton Capacity*, United Nuclear Corporation, Richland, Washington.

The Certificate of Compliance for the CNS 1-13G shipping cask was reviewed to assure that shipments of irradiated N Reactor fuel element components or assembled components are within the operating envelope analyzed for the cask. The criticality analysis that goes with the Certificate of Compliance was also reviewed.

An N Reactor fuel assembly is a tube-in-tube design with inner and outer zirconium-clad uranium metal fuel elements. There are two different types of assemblies; the MKIV assembly with 0.95 wt% ^{235}U enriched inner and outer fuel elements, and the MKIA assembly with a 0.95 wt% ^{235}U enriched inner element and a 1.25 wt% ^{235}U enriched outer element.

The Certificate of Compliance has two parts of interest for this review. Section 5b1 part i specifies fuel assemblies of 500 grams ^{235}U equivalent mass and a decay heat not to exceed 600 watts. Section 5b1 part iii discusses irradiated PuO_2 or UO_2 fuel rods with a fissile limit of 1,200 grams ^{235}U equivalent mass and decay heat not to exceed 600 watts.

The criticality analysis for the cask discusses irradiated PuO_2 or UO_2 fuel rods, but the results presented in the analysis are for pure ^{239}Pu metal.

R. J. Smith
Page 2
September 8, 1994

The analysis shows that 1,200 grams of ^{239}Pu is safe with a maximum calculated $k_{\text{eff}} = 0.976$. N Reactor MKIA fuel elements, the most reactive type, with a maximum enrichment of 1.25 wt% ^{235}U , are much less reactive than pure ^{239}Pu . The minimum critical mass of the maximum enriched N Reactor fuel elements is 1,087 kgs uranium metal (Reference 1), with approximately 13.6 kgs of ^{235}U . Experimental data (Reference 3) show that it takes about 120 unirradiated N Reactor MKIA fuel assemblies optimally moderated and reflected to reach a $k_{\text{eff}} = 0.98$, and 60 assemblies have a measured $k_{\text{eff}} \approx 0.90$. The data also show that irradiated fuel is much less reactive than unirradiated fuel. The cask would contain a maximum of two assemblies or three outer elements. This is a small fraction of the quantity necessary for a criticality. Calculations of irradiated N Reactor fuel decay heat show that the assemblies with the highest exposure have a decay heat of less than 3 watts each (Reference 2), this would give a thermal load of less than 9 watts for the loaded cask.

This review shows that N Reactor fuel falls within the authorization of the Certificate of Compliance under both of the identified sections. Therefore, shipments of less than 500 grams fissile equivalent ^{235}U mass, consisting of up to two inner and two outer elements or any combination of three inner or outer elements is allowable.

Prepared by:

David G. Erickson 9-8-94
D. G. Erickson, Senior Scientist
Nuclear Analysis and Characterization

Reviewed by:

Paul D. Friar for D.E.F. per telecon 9-8-94
D. E. Friar, Principal Engineer
Criticality and Radiological Analysis

Approved by:

Hann Toffler 9-8-94
H. Toffler, Manager
Nuclear Analysis and Characterization

kls

7.0 STRUCTURAL EVALUATION

7.1 INTRODUCTION

The SARP for the CNS 1-13G demonstrates that the cask can withstand all structural tests required by federal regulations (ChemNuclear 1992). Since the planned shipping campaign will include the transfer of elements with breached fuel pin cladding, SFECs will be used to accentuate the safety of the packaging system.

The SFECs and shoring weigh considerably less than the optional shielding liners that were assumed for the SARP structural analysis. Therefore the SFECs and shoring will not impose forces on the cask greater than those used in the SARP analysis during normal conditions of transport and under accident conditions.

7.2 SUMMARY OF RESULTS

The CNS 1-13G has been structurally evaluated in the SARP (ChemNuclear 1992) and has been determined to maintain adequate levels of containment and shielding during normal conditions of transfer and under accident conditions.

The SFECs will retain their integrity for normal conditions of transport, which include handling drops of the cask onto a concrete surface from a distance of 2 ft.

Under accident conditions, although the CNS 1-13G will maintain its integrity, the SFECs may rupture, potentially allowing the fuel to react chemically with the atmosphere inside the cask. A severe reaction could potentially overpressurize the cask, causing a release of radioactive material to the environment (Part B, Section 9.0). The radiological risk assessment (Part B, Section 3.0) addresses the frequency of such an event, while the dose consequence analysis (Part B, Section 4.0) determines the resulting doses to on and offsite persons from such a release.

7.3 STRUCTURAL EVALUATION OF THE PACKAGING

7.3.1 CNS 1-13G Cask

From the CNS 1-13G SARP (ChemNuclear 1992):

[I]t is concluded that under normal conditions of transport, the package complies with the following conditions:

1. There will be no release of radioactive material from the containment vessel.
2. The effectiveness of the packaging will not be substantially reduced.
3. There will be no mixture of gases or vapors in the package which could, through any credible increase in pressure or an explosion, significantly reduce the effectiveness of the package.

Additionally:

The accident tests or assessments described above demonstrated that the package is adequate to retain the product contents and that there is not change in spacing. Therefore it is concluded that the CNS Shielded Container - Model 1-13G is adequate as packaging for the contents specified in section 1.2.3 (of the SARP).

7.3.2 Single Fuel Element Containers

7.3.2.1 Structural Design and Features. The SFEC is constructed of a 27 in. high by 4 in. OD 0.065 in. wall thickness tube. The tube has a welded bottom and is outfitted with a flange to allow for positive closure of the lid and for lifting via a hoop attached to the flange. To enhance the container's compatibility with existing K Basin equipment, two 5/16 in. blocks were welded to the bottom of the container. Part A, Section 9.2.2 contains the drawings for the SFEC.

7.3.2.2 Mechanical Properties of Materials.

7.3.2.3 Chemical and Galvanic Reactions. The materials of the SFEC have been chosen to preclude any significant chemical and/or galvanic reactions with the payload or the CNS 1-13G cask.

7.3.2.4 Weights and Center of Gravity. The weight of an empty SFEC is 18 lb. The maximum weight of the payload will be 51 lb for an inner and outer fuel element and 8.9 lb for the water. The gross weight of a loaded SFEC will be 62.1 lb. The center of gravity of the canister is located approximately 1.5 in. above the canisters geometric center.

7.3.2.8 Brittle Fracture. Significant degradation of structural properties of the SFEC is not anticipated as the thickness of all load bearing materials precludes brittle fracture at low temperatures as per NUREG 7.11 (NRC 1991).

7.4 NORMAL CONDITIONS OF TRANSPORT

7.4.1 Conditions to be Evaluated

The CNS 1-13G cask will be transported on a 2 ft high lowboy trailer. During onloading or offloading, the possibility exists that the cask could suffer a hard set down. To determine the effects of this, the SFEC was evaluated for a 2 ft drop of the fully loaded CNS 1-13G onto an unyielding surface.

7.4.2 Acceptance Criteria

Subsequent to a 2 ft drop of the fully loaded CNS 1-13G cask, the SFEC shall not suffer any reduction in its ability to retain the irradiated fuel samples and the water used to preclude chemical reactions with the cask atmosphere.

All portions of the inner canister that will bear loads during a two foot handling drop are constructed of 304 stainless steel. The working strength for 304 stainless steel is 20,000 psi (ASME 1989). The minimum buckling stress for the SFEC is 276,000 psi (Part B, Section 7.6.2).

7.4.3 Structural Model

The SFEC was evaluated using first principles for a 2 ft drop onto an unyielding surface. The can was evaluated for a oblique drop and for a 30° drop, which represents the largest angle possible for a two foot drop. The cask and SFEC were assumed to experience the same deceleration, which is conservative since even minor cask deformation could significantly reduce the forces experienced by the SFEC.

The SFEC was evaluated for buckling of container wall, failure of the container bottom and failure of the bottom weld. It was assumed that the SFEC was filled with water and contained an inner and outer fuel element weighing 51 lb. Part B, Section 7.6.2 contains the detailed structural evaluation.

7.4.4 Initial Conditions

7.4.4.1 Environmental Heat Loading. The limiting temperature for the SFEC is 225 °F, which is the maximum operating temperature for the container gasket. The thermal evaluation in the CNS 1-13G SARP (ChemNuclear 1992), determined that the maximum temperature within the cask will not exceed 190 °F for normal conditions of transport with a payload decay heat of 600 W. The maximum anticipated payload decay heat for this shipping campaign is 8.45 W (Part B, Section 8.3), so the temperature of the SFEC will be considerably less than 190 °F.

7.4.4.2 Maximum Pressure Stresses. Since the SFEC is vented to the inner cavity of the CNS 1-13G cask, there will be no build up of pressure.

7.4.5 Evaluation

The maximum stresses experienced by the canister are evaluated in Part B, Section 7.6.2 and are shown in Table B7-1. All components that will see a load during a 2 ft handling drop have positive margins of safety. Therefore, integrity of the SFECs will be maintained.

Table B7-1. Maximum Stresses for the CNS 1-13G Single Fuel Element Container During a Two Foot Handling Drop.

Component	Allowable Stress	Maximum Stress	Margin of Safety
Container Wall (Bending)	20,000 psi	14,770 psi	+0.35
Container Wall (Buckling)	276,000 psi	6,104 psi	Large
Container Bottom	20,000 psi	14,410 psi	+0.39
Container Bottom Weld	20,000 psi	17,270 psi	+0.16

7.5 ACCIDENT CONDITIONS

During an accident condition, such as a collision or rollover, it cannot be established that the SFEC will maintain its integrity. The analysis of normal conditions of transport (Part B, Section 7.4) indicate that in an accident, the SFEC walls would fail due to hydrostatic pressure. The chemical and gas generation analysis (Part B, Section 9.0) determined that if the SFEC loses its integrity, the reaction between the irradiated fuel and the cask atmosphere could potentially cause overpressurization and leakage of radioactive materials. Part B, Sections 3.0 and 4.0 analyze the frequency and dose consequence respectively of such a release.

7.6 APPENDICES

7.6.1 References

ASME, 1989, *ASME Boiler and Pressure Vessel Code, Section III, Division 1*, American Society for Mechanical Engineering.

ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()F*, ChemNuclear, Inc, Columbia, South Carolina.

Crede, H., and R. Harris, 1961, *Shock and Vibration Handbook*, McGraw-Hill, New York.

NRC, 1991, *Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 inches (0.1 m)*, NUREG 7.11, U.S. Nuclear Regulatory Commission.

Young, W. C., 1982, *Roark's Formulas for Stress and Strain*, Fifth Edition, McGraw-Hill, New York.

7.6.2 Structural Analysis for CNS 1-13G Single Fuel Element Container During Two Foot Drop

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 1 of 4
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject Fuel Element Canister Analysis for 2' Drop.
 (8) Originator W.W. SMYTH W.W. Smyth Date 3 Aug 94
 (9) Checker PF STEVENS Paul F. Stevens Date 8/6/94

(10)

Find the effects of dropping a water-filled canister 2' onto an unyielding surface.
 USE A .002 second force response time & a sinusoidal pulse per [SHOCK & VIBRATION H'BOOK]
 SINCE THE TUBE IS FILLED WITH WATER, THE REACTION FORCE ON THE BOTTOM WILL INCREASE THE EFFECTIVE PRESSURE IN THE TUBE DUE TO THE FLUID HEAD.
 4" OD 0.05" WALL

Find the peak static equivalent acceleration. After a 24" drop, the impact velocity is $\sqrt{2 \cdot 386.4 \cdot 24} = 136.2 \text{ in/sec}$
 rigid body acceleration

$$a = A \sin \frac{\pi t}{2T} \quad \text{where } A = \text{peak accel.} \\ t = \text{time}$$

$T = \text{pulse rise time} = .002 \text{ seconds}$

integrate to find $V(t)$ & find A to cause $V(.002) = 0$

$$V = V_0 - \frac{2AT \cos \frac{\pi t}{2T}}{\pi} \Big|_0^{t=.002} = -V_0 + \frac{2A(.002)}{\pi} = 0$$

$$A = \frac{V_0 \pi}{.004} = \frac{136.2 \pi}{.004} = 107,000 \text{ in/sec}^2$$

$$\text{this is } A/g = \frac{107,000}{386.4} = 277 \times g$$

The max pressure in the can will be $277 \cdot 27'' \cdot \frac{62.4 \text{ PCF}}{1728 \text{ in}^3}$

$$P_{\text{max}} = 270 \text{ psi}$$

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 2 of 4
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator W. W. Smyth W. W. Smyth Date 3 Aug 94
 (9) Checker PF STEVENS Paul F. Stevens Date 8/6/94

(10) $G_1 = \frac{18'' \times 277}{2\pi \times 2 \times .065} = 6104 \text{ psi}$ USE ROARK, 5th ed
 $\frac{R}{t} = \frac{2''}{.065} = 31 > 10$ SO THIN CYLINDER THEORY IS OK.

$G_1 = 6104$ [ROARK, p 448]

$G_2 = \frac{pR}{t} = \frac{270 \times 2}{.065} = 8310 \text{ psi}$

max SI = $8310 + 6100 = 14410$ ASME section 3 limits SI to 20 KSI for type 304 stainless steel. [ASME, 1989]
 The bottom is well supported - but will create bending stresses in the tube by preventing the tube from expanding.

Free expansion of tube: $\Delta R = \frac{qR^2}{Et}$ [ROARK]

$q = 270 \text{ psi}$ $E = 28.3 \text{ EG psi}$ [ASME, 1989]

$\Delta R = \frac{270(2)^2}{28.3 \text{ EG} \times .065} = 587 \text{ E-6 in} < .065$

This expansion is prevented by a base shear V_0 + a moment M_0

From Roark p 458, $\lambda = \left[\frac{3(1-\nu^4)}{R^2 t^2} \right]^{1/4} = \left[\frac{3(1-.3^4)}{2^2 (.065)^2} \right]^{1/4}$

A "LONG TUBE" is $2.6 \lambda = 1.68'' = 3.565/\text{inch}$

SINCE LENGTH $> 27''$, THIS IS A LONG TUBE

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 3 of 4
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator W² W. Smyth W-L Smyth Date 3 AUG 94
 (9) Checker PF STEVENS Paul F. Stevens Date 8/6/94

(10)

USE ROARK p 461 + 462

$$\text{due to } V_0: \psi = \frac{V_0}{2D\lambda^2} \quad \gamma = +\frac{V_0}{2D\lambda^3}$$

$$\text{due to } M_0: \psi = -\frac{M_0}{D\lambda} \quad \gamma = -\frac{M_0}{2D\lambda^2}$$

$$\sum \psi = 0 = \frac{V_0}{2D\lambda^2} - \frac{M_0}{D\lambda} = 0$$

$$M_0 = \frac{V_0}{2\lambda}$$

$$\sum \gamma = \Delta R(\text{free}) = +587E-6 = \frac{V_0}{2D\lambda^3} - \frac{V_0/2\lambda}{2D\lambda^2} = \frac{V_0}{4D\lambda^3}$$

$$D = \frac{Et^3}{12(1-\nu^2)} = \frac{28.3E6(.065)^3}{12(1-.3^2)} = 712$$

$$V_0 = 4(712)(3.565)^3 \cdot 587E-6 = 75.8 \text{ \#}/\text{in}$$

$$M_0 = \frac{75.8}{2 \times 3.565} = 10.4 \text{ \#-in}/\text{in}$$

At the extreme base, $G_2 = 0$

$$f_b = \frac{6M}{t^2} = \frac{6 \times 10.4}{.065^2}$$

bending stress limit is $= 14,770 \text{ psi}$

DESIGN CALCULATION

(1) Drawing _____ (2) Doc. No. _____ (3) Page 4 of 4
 (4) Building _____ (5) Rev. _____ (6) Job No. _____
 (7) Subject _____
 (8) Originator W. W. Smyth W. W. Smyth Date 3 Aug 94
 (9) Checker PF STEVENS Paul F. Stevens Date 8/6/94

(10)

Assuming the can's lid + body weigh 18[#], the compressive loading in the tube is

$$277 \times 18 = 4986 \quad \text{stress } \sigma_c = - \frac{4986}{4\pi \times .065} = 6104 \text{ psi}$$

ROARK P 555 STABILITY

$$\text{approx lower bound } \sigma_{cr} = .3 \frac{E}{r}$$

$$= .3 \frac{(28.3 \text{ EG}) \times .065}{2} = 276 \text{ KSI}$$

BUCKLING IS NOT A PROBLEM

TUBE - BOTTOM WELD : .065" FILLET

$$\downarrow 4986 / 4\pi = 397 \text{ \#/inch}$$

$$\max \tau = \frac{397}{.707 \times .065} = 8634 \text{ psi}$$

$$\text{STRESS INTENSITY} = 2\tau_{\max} = 17,270 \text{ psi} < 20 \text{ KSI}$$

8.0 THERMAL EVALUATION

8.1 INTRODUCTION

The SARP for the CNS 1-13G (ChemNuclear 1992) evaluates the casks for normal transport thermal conditions from -40 °F to 130 °F both with and without solar insolation. Additionally, it analyzed the effects of a half-hour 1475 °F fire on the cask. The SARP determined that the CNS 1-13G could safely transport payloads that generate up to 600 W of heat (ChemNuclear 1992).

The worst case payload that will be transferred in the CNS 1-13G during the transfer of irradiated N Reactor/SPR fuel will be less than 9 W which is considerably below the limit imposed by the SARP. However, during the shipment, the fuel will be in water filled inner containers, which is not accounted for the SARP analysis. The SARP analysis indicates that temperatures within the inner cavity of the CNS 1-13G will exceed 400 °F. This will cause boiling of the water in the SFECs potentially causing the internal pressure within the CNS 1-13G to exceed its 15 psig test pressure.

8.2 SUMMARY OF RESULTS

For the analysis of the CNS 1-13G under normal conditions, the SARP states:

Packaging components, i.e., steel shells and lead shielding are unaffected by temperature extremes of -40°F and 130°F. Package contents, at least singly-encapsulated, or contained in specification 2R containers, or other inner containers, but not limited to special form, will not be affected by these temperature extremes.

For the analysis of the CNS 1-13G under fire conditions, the SARP states:

A major objective of this accident evaluation is to determine if any shielding lead melted as a result of the event. The [thermal analysis] computations show a maximum lead temperature of 479°F for the cask without liner case. The calculated maximum lead temperature for the cask with liner case is 474°F. These maximum lead temperatures are well below the 621°F melting point of lead.

The SARP fire analysis indicates that the maximum inner cask temperature will be 410 °F, which is more than sufficient to boil the inner container water. In the case of a fire based accident, it is assumed that the water will boil and the cask will be pressurized beyond the 15 psig test pressure causing failure of the cask gasket system and a release of radioactive

material to the environment. Part B, Sections 3.0 and 4.0 evaluates the frequency of such a fire based release and the dose consequence of a release, respectively.

8.3 THERMAL SOURCE SPECIFICATION

Table B8-1 contains the decay heat generation for the worst case payload. The maximum possible heat generation is 8.45 W (28.84 Btu/h), which is less than the 600 W permitted by the SARP (ChemNuclear 1992).

Table B8-1. Thermal Source for CNS 1-13G with 16% ^{240}Pu and 500 g Fissile Content.

Nuclide	Curies of Material	Heat (W)	Nuclide	Curies of Material	Heat (W)
^3H	2.17 E+00	5.85 E-05	^{144}Ce	1.17 E+02	9.44 E-01
^{55}Fe	1.71 E+00	6.05 E-05	$^{144\text{m}}\text{Pm}$	1.40 E+00	
^{60}Co	5.70 E-01	8.88 E-03	^{147}Pm	3.58 E+02	1.28 E-01
^{63}Ni	9.00 E-02	9.64 E-06	^{151}Sm	4.26 E+00	5.20 E-04
^{85}Kr	3.84 E+01	6.12 E-02	^{152}Eu	6.00 E-02	0.00
^{90}Sr	3.59 E+02	2.45 E+00	^{154}Eu	1.23 E+01	1.04 E-01
^{99}Tc	7.00 E-02	0.00	^{155}Eu	3.95 E+00	2.80 E-03
^{106}Ru	1.04 E+02	1.03 E+00	^{235}U	6.46 E-04	1.82 E-05
$^{110\text{m}}\text{Ag}$	5.00 E-02	0.00	^{238}Pu	4.94 E+00	1.62 E-01
$^{113\text{m}}\text{Cd}$	2.00 E-01	0.00	^{239}Pu	5.80 E+00	1.83 E-01
$^{119\text{m}}\text{Sn}$	9.00 E-02	0.00	^{240}Pu	4.11 E+00	1.30 E-01
^{125}Sb	2.75 E+01	8.65 E-02	^{241}Pu	4.66 E+02	1.53 E-02
$^{125\text{m}}\text{Te}$	6.73 E+00	0.00	^{241}Am	4.48 E+00	1.52 E-01
^{134}Cs	5.81 E+01	5.99 E-01	^{244}Cm	1.45 E+00	5.11 E-02
^{137}Cs	4.80 E+02	2.34 E+00			
Total Payload Heat Generation					8.45

8.4 THERMAL EVALUATION OF PACKAGING

The thermal analysis of CNS 1-13G cask is contained in the CNS 1-13G SARP (ChemNuclear 1992). The N Reactor/SPR fuel characterization shipping campaign will not impose any thermal conditions that will depart from those analyzed in the SARP. Therefore, no additional thermal analyses are required.

8.5 APPENDIX

8.5.1 Reference

ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()*F, ChemNuclear, Inc, Columbia, South Carolina.

9.0 PRESSURE AND GAS GENERATION

9.1 GAS GENERATION

9.1.1 Normal Conditions of Transport

It is expected that gas will be generated within SFECs during normal conditions of transport. This gas generation will consist primarily of hydrogen and oxygen created by corrosion of breached fuel and to a lesser extent radiolysis. The gas generation will also include trace amounts of noble gasses due to radionuclide decay.

The limits for hydrogen concentration during transport are 5% by volume with a factor of safety of two, or, assuming a linear hydrogen generation rate 2.5% by volume (NRC 1984). Prior to shipment, the gas generation rate within the inner canister shall be determined via a pressure measurement lid for the SFEC (H-1-80395). It shall be conservatively assumed that all gas generated is hydrogen. The calculated gas generation rate will determine the amount of time that the cask can be sealed before the H_2 concentration exceeds 2.5%.

9.1.2 Accident Conditions

It has not been established that under severe accident conditions that the SFEC would maintain its integrity. If the single fuel element canister lost its integrity, the fuel element segments would be exposed to the air within the sealed cask. Some of the fuel samples could have significant concentrations of UH_3 (Baker et al. 1966), which can react with oxygen at relatively low temperatures.

The atmosphere within the cask will be air at atmospheric pressure, which consists of 0.126 g-mol O_2 and 0.472 g-mol N_2 . To insure that the atmosphere within the cask will react to as great a degree as possible, it is assumed that 50% of the uranium metal available in the fuel element has reacted during storage to form UH_3 . In reality such high concentrations of UH_3 are extremely unlikely as UH_3 -water reactions would occur before such levels were reached. The total amount of UH_3 will be 80.7 g-mol. The WCET chemical analysis code (McLaughlin and Slater 1993) was utilized to evaluate this potential reaction. The input and output files for this calculation is in Part A, Section 9.3.2.

The code predicted that 40,000 cal (159 Btu) would be released during the course of the reaction, which would raise the temperature of the atmosphere within the cask 6.6 °F. It predicted that the products would consist of 1.605 g-mol of H_2 (gas), 0.944 g-mol of UN (solid), 0.126 UO_2 (solid), and 79.63 g-mol UH_3 (solid). The gas pressure developed by this reaction was found to be sufficient to exceed the 15 psig rating of the cask (Section 9.2.2). The effects of a possible reaction between the fuel and the atmosphere external to the cask are beyond the scope of this analysis.

9.2 PACKAGE PRESSURE

9.2.1 Normal Conditions of Transport

The acceptance criteria for the SARP (ChemNuclear 1992) mandates that periodically, "The cask shall be pneumatically pressurized to 15 psi and while under pressure, seals are soap bubble checked for leakage acceptance criteria - no visible bubbles." The maximum allowable pressure for the cask under normal conditions of transport is therefore 15 psig. If it is conservatively assumed that all hydrogen is generated by radiolysis, so oxygen will be created as well. If the concentration of hydrogen in the cask reached 2.5% by volume, then the increase in pressure is

$$0.025 \underset{(H_2)}{x} 14.7 \text{ psi} + 0.0125 \underset{(O_2)}{x} 14.7 \text{ psi} = 0.55 \text{ psig}$$

Therefore, the maximum package pressure expected during the shipment of N Reactor/SPR fuel is acceptable.

9.2.2 Accident Conditions

9.2.2.1 Fire Based Accident. During a fire based accident, it is assumed since the temperatures within the cask inner cavity reach temperatures in excess of 400 °F that sufficient water boils to overpressurize the cask causing a release of radioactive material to the environment. Part B, Section 3.0 calculates the frequency of such a release and Part B, Section 4.0 assesses the maximum dose that could be received from such a release.

9.2.2.2 Non Fire Based Accident. As per the ideal gas law, the pressure within the cask after such an accident can be expressed as

$$P_a = P_n \times \left(\frac{n_a}{n_n} \right) \times \left(\frac{T_a}{T_n} \right)$$

where P_n = Pressure prior to accident (14.7 psia)

P_a = Pressure subsequent to accident

n_n = Gas moles prior to accident (0.598 mol)

n_a = Gas moles subsequent to accident (1.605 mol)

T_n = Absolute temperature prior to accident (40 °F)

T_a = Absolute temperature subsequent to accident (46.6 °F)

The maximum pressure after an accident would be 39.95 psia or 25.2 psig. The SARP for the CNS 1-13G (ChemNuclear 1992) indicates that the cask is tested to 15 psig prior to shipment. Therefore, it is possible that the cask

will be overpressurized subsequent to a collision or rollover based accident and will release radioactive material to the environment. Part B, Section 3.0 addresses the frequency of such a release and Part B, Section 4.0 assesses the maximum dose consequence of such a release.

9.3 APPENDICES

9.3.1 References

Baker, M. McD., L. N. Ness, and S. Orman, 1966, *Uranium and Water Reactions Part 2. Effect of Oxygen and Other Gases*, Transactions Faraday Society, 62:2525-2530.

ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()*F, ChemNuclear, Inc., Columbia, South Carolina.

McLaughlin, D. F., and C. G. Slater, 1993, *WCET/CHEMEQ Chemical Equilibrium Thermodynamics Code and Database*, Westinghouse Science and Technology Center, Pittsburgh, Pennsylvania.

9.3.2 WCET Input and Output Files

Example-1: Burning of the UH3 within the KNSHIP Cask @70F

```
REACTANTS
O 2.000          00      0.12600M    0.0 G 530.00 O 1.000
N 2.000          00      0.47200M    0.0 G 530.00 O 1.000
U 1.000 H 3.000  00      80.7000M    0.0 S 530.00 O 1.000
```

```
NAMELIST
&INPT2
TP=T,
OF=T,
MIX= 0.00,
P = 1.000,
T = 530.0,
&END
STOP
```

1

Example-1: Burning of the UH3 within the KNSHIP Cask @70F
 OERROR IN ABOVE CARD. CONTENTS IGNORED.

REACTANTS													
O	2.0000	.0000	.0000	.0000	00	.0000	.126000	M	.00	G	530.000	O	
N	2.0000	.0000	.0000	.0000	00	.0000	.472000	M	.00	G	530.000	O	
U	1.0000	H	3.0000	.0000	.0000	00	.0000	80.699997	M	.00	S	530.000	O

NAMELIST

```

O ***INPT2***
OKASE = 0 IDEBUG = 0 TRACE = .00000D+00 IONS = F SIUNIT = F
OTP = T HP = F SP = F TV = F UV = F SV = F RKT = F SHOCK = F DETN = F
OTRNSPT = F TRPACC = .999950E+00 NODATA = F
OOF = T FA = F FPCT = F ERATIO = F PHI = F
OSO = .00000000E+00 U = .00000000E+00 H = .00000000E+00
OP = .10000E+01
OT = .53000E+03
OMIX = .00000E+00
OSPECIES BEING CONSIDERED IN THIS SYSTEM
J 3/77 H J 3/77 H2 WE6/87 HN3 J12/65 H2N2
RUS 78 HNO RUS 78 HNO2 RUS 78 HNO3 J 9/78 HO2 J 3/79 H2O L
3/85 H2O2 J 3/77 N RUS 78 N3 RUS 78 NH
RUS 78 NH2 J 6/77 NH3 RUS 78 N2H2 RUS 78 N2H4 RUS 78 N3H
RUS 78 NH2OH RUS 78 NH2NO2 RUS 78 NO RUS 78 NO2 J12/64 NO3
RUS 78 N2O RUS 78 N2O3 RUS 78 N2O4 RUS 78 N2O5 J 3/77 O J
3/77 O2 J 6/61 O3 WE6/87 U WE6/87 UO
WE6/87 UO2 WE6/87 HNO2(cis) WE6/87 HNO2(trans) L 3/81 H2O(S) J
3/79 H2O(L) WE6/87 N2H4(L) WE6/87 U(S) WE6/87 U(L)
WE6/87 UH3(S) WE6/87 UO2(S) Ba6/93 UO2(NO3)2(S) WE6/87 UO3(S)
WE6/87 UN(S) WE6/87 U3O8(S) WE6/87 U4O9(S)
OOF = .000000

```

ENTHALPY		EFFECTIVE FUEL		EFFECTIVE OXIDANT		MIXTURE	
(KG-MOL)(DEG K)/KG		HPP(2)		HPP(1)		HSUBO	
OKG-FORM.WT./KG		BOP(I,2)		BOP(I,1)		BOP(I)	
O	.12942851E-04	-.57428962E+02	.00000000E+00	.25200000E+00	.12942851E-04		
N	.48484331E-04	.12942851E-04	.94400001E+00	.80699997E+02	.48484331E-04		
U	.41447938E-02	.41447938E-02	.80699997E+02	.24209999E+03	.41447938E-02		
H	.12434381E-01	.12434381E-01	.24209999E+03		.12434381E-01		
OPOINT ITN T O	N	U	H				
1 47 530.00 -140.792	-15.163	93.354	-8.351				
ADD UN(S)							
1 5 530.00 -140.780	-168.529	93.347	-8.346				
ADD UH3(S)							
1 36 530.00 -70.158	-62.193	-12.989	-8.120				
ADD UO2(S)							
1 5 530.00 -121.824	-62.113	-13.069	-8.093				

1
0

THERMODYNAMIC EQUILIBRIUM PROPERTIES AT ASSIGNED
TEMPERATURE AND PRESSURE

TEMP	DENSITY		MOLES	ENERGY	STATE
DEG K	CHEMICAL FORMULA			CAL/MOL	
FUEL	G/CC				
530.00	O 2.00000		.126000	1678.409	G

FUEL N 2.00000 .472000 1625.461 G
 530.00
 FUEL U 1.00000 H 3.00000 80.700 -27546.363 S
 530.00

O O/F= .0000 PERCENT FUEL= 100.0000 EQUIVALENCE RATIO=***** PHI= .0000
 OTHERMODYNAMIC PROPERTIES

P, ATM 1.0000
 T, DEG K 530.00
 RHO, G/CC 2.7893-1
 H, CAL/G -117.45
 U, CAL/G -117.54
 G, CAL/G -167.87
 S, CAL/(G)(K) .0951

M, MOL WT 12130.971
 (DLV/DLP)T -1.00000
 (DLV/DLT)P 1.0000
 CP, CAL/(G)(K) .0549
 GAMMA (S) 1.0030
 SON VEL,M/SEC 19.1

OMOLE FRACTIONS

H2 .01950
 UH3(S) .96750
 UN(S) .01147
 UO2(S) .00153

ADDITIONAL PRODUCTS WHICH WERE CONSIDERED BUT WHOSE MOLE FRACTIONS WERE LESS THAN .50000E-05 FOR ALL
 ASSIGNED CONDITIONS

H	HN3	H2N2	HNO	HN02	HN03
HO2					
H2O	H2O2	N	N2	N3	NH
NH2					
NH3	N2H2	N2H4	N3H	NH2OH	NH2NO2
NO					
NO2	NO3	N2O	N2O3	N2O4	N2O5
O					
O2	O3	OH	U	UO	UO2
UO3					
HN02(cis)	HN02(trans)	H2O(S)	H2O(L)	H2O2(L)	N2H4(L)
U(S)					
U(L)	UO2(NO3)2(S)	UO3(S)	U3O8(S)	U4O9(S)	
1					
STOP					

10.0 PACKAGE TIEDOWN SYSTEM

10.1 SYSTEM DESIGN

The package tiedown system is described in the SARP for the CNS 1-13G (ChemNuclear 1992). The tiedown system consists of bolting the CNS 1-13G cask and overpack to a specially equipped trailer. Additionally, two tiedown lines that are attached at the top of the overpack and anchored to the transport trailer. The tiedown devices for the CNS 1-13G was designed to conform with 10 CFR 71 and can withstand accelerations of 10g in the forward direction, 5g laterally and 2g vertically (ChemNuclear 1992).

10.2 APPENDIX

10.2.1 Reference

ChemNuclear, 1992, *Procedures, License and Safety Analysis Report for ChemNuclear Systems, Inc. CNS 1-13G Type A Radwaste Shipping Cask, USA/9216/B()F*, ChemNuclear, Inc., Columbia, South Carolina.

9.3 CNS 1-13G LOADING PROCEDURE