71-9001

CHEM-NUCLEAR SYSTEMS, LLC

140 Stoneridge Drive • Columbia, South Carolina 29210 • (803) 256-0450

11 October, 2000
579-190-00

Mr. Timothy J. McGinty, Project Manager
Addressee Only
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards, NMSS
U.S. Nuclear Regulatory Commission
Mail Stop O-13-D-13
Washington, DC 20555

Subject: Response to Request for Additional Information for Model No. IF-300, CofC No. 9001; Docket 71-9001; TAC Number L22931

Reference: 1) 16 June 2000 letter (579-103-00) from P. Paquin to T. McGinty including revised submittal
2) 22 September 2000 Conference Call chaired by T. McGinty including NRC staff and representatives from Chem-Nuclear Systems(Chem-Nuclear) and Carolina Power and Light (CP&L)

Dear Mr. McGinty:

In the 22 September 2000 conference call (Reference 2), the NRC staff, Chem-Nuclear, and CP&L discussed the Group III PWR fuel loading restriction proposed in the revised submittal (Reference 1). Since the center cell in the PWR fuel basket must remain empty for Group III fuel to satisfy the criticality limit, the NRC staff requested a more positive control than the proposed administrative controls to prevent loading the center cell. After discussion, CNS and CP&L agreed to evaluate a physical control mechanism and provide a further revision to the previous submittal (Reference 1).

Enclosed is the submittal which addresses the physical controls to prevent loading the center cell of the PWR basket when shipping Group III fuel. The submittal consists of:

- Attachment 1 – Replacement page iii-4, a revised page iii-4 Revision Control Sheet
- Attachment 2 – Replacement page 4-i/4-ii (revised Table of Contents) and New page 4-22, the drawing of a PWR Fuel Basket Center Plug, for inclusion in Section 4 of Volume 1 of the CSAR
- Attachment 3 – Replacement pages 10-i through 10-15 (New Section 10), a revision of the operating procedures in Section 10 of Volume 1 of the CSAR to address use of the plug
- Attachment 4 – Replacement pages D-1-2/D-1-3 and D-6-29/D-6-30, a revision of two pages of Appendix D to address the impact on criticality of use of the plug.

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Attachments 1, 3, and 4 contain replacement pages for pages previously submitted (Reference 1); please discard the previously submitted pages and replace them with the pages provided in this submittal. Attachment 2 contains a replacement Table of Contents (TofC) and a new drawing for Section 4 of the current CSAR; replace the TofC pages and insert the new drawing. Changes on a page are noted by a mark in the margin.

Should you or members of your staff have questions about the responses, please contact Mark Whittaker at (803) 758-1898.

Sincerely,

Patrick L. Paquin

Patrick L. Paquin
General Manager – Engineering and Licensing

Attachments: As stated

**ATTACHMENT 1**
**REPLACEMENT PAGE**
**REVISION CONTROL SHEET**

# REVISION CONTROL SHEET

TITLE: Consolidated Safety Analysis Report for IF-300 Shipping Cask

DOCUMENT NO.: NEDO-10084

| AFFECTED PAGE(S) | DOC. REV. | REMARKS |
|---|---|---|
| 1-1 - 1-3 | 5 | First revision prepared by CNS, Inc. |
| 2-1 - 2-3a | " | Incorporates C of C 9001, Revision xx references |
| 2-10 - 2-12a | " | From xx/xx/xx through xx/xx/2000. |
| 2-14 & 2-15 | " | A vertical line on the left hand margin indicates a |
| 3-1 & 3-1a | " | Revision. |
| 4-ii & 4-22 | " | |
| 5-3 & 5-4 | " | |
| 6-1 - 6-2 | " | |
| 8-1 & 8-2 | " | |
| 9-1 -& 9-2 | " | |
| 10-i - 10-ii | " | |
| 10-1 - 10-15 | " | |
| D-i - D-vii | " | |
| D-1-1 - D-1-2 | " | |
| D-2-1 - D-2-3 | " | |
| D-3-1- D-3-15 | " | |
| D-4-1- D-4-16 | " | |
| D-5-1- D-5-13 | " | |
| D-6-1 - | " | |
| D-6-170 | " | |
| D-7-1- D-7-11 | " | |
| D-8-1 - D-8-3 | " | |
| D-9-1- D-9-46 | " | |
| | | PAGE 4 of 4 |

ATTACHMENT 2
REPLACEMENT PAGES
VOLUME 1, SECTION 4

# TABLE OF CONTENTS

## IV. EQUIPMENT DESCRIPTION

# LIST OF ILLUSTRATIONS

FIGURE WITHHELD UNDER 10 CFR 2.390

| | | |
|---|---|---|
| GTS DURATEK | | |
| PWR FUEL BASKET CENTER PLUG FOR IF-300 CASK | | |
| SIZE B | DRAWING NUMBER C-110-B-57915-001 | REV. 0 |
| SCALE 1/6 | WT N / A | SHEET 1 OF 1 |

ATTACHMENT 3
REPLACEMENT PAGES
VOLUME 1, SECTION 10

# TABLE OF CONTENTS

## X. OPERATION, MAINTENANCE AND TESTING

## LIST OF TABLES

# X. OPERATION, MAINTENANCE AND TESTING

## 10.1 OPERATING PROCEDURES

Instructions for use of the IF-300 Transportation System are published in the Chem-Nuclear document "Operating Instructions, IF-300 Irradiated Fuel Transportation System", GEI-92817. This manual describes the complete handling sequence for preparation, loading, transport, and unloading. The manual is used for operator training as well as on-the-job direction. During actual operation of the cask the manual may be supplemented with Chem-Nuclear technical advisor, training classes, and site specific procedures, as applicable.

The operating procedures are summarized below:

### 10.1.1 Procedures for Cask Loading

Operations at the loading facility include the span of activities from receiving and inspecting the cask to preparing the loaded cask for shipment. Each loading facility must provide fully trained personnel and detailed operating procedures to cover all of the activities.

#### 10.1.1.1 Cask Receiving and Inspection

a. The IF-300 railroad car is oriented, chocked, and braked.
b. A visual inspection for damage and leakage is made and a radiological survey of the cask is initiated in accordance with the requirements of 10 CFR 20.

#### 10.1.1.2 Preparing for Cask Removal from the Rail Car

a. The cask enclosures are opened.
b. The valve box covers are removed.
c. Cask tie-down pins are removed and the lifting trunnions are installed.
d. The cask lifting yoke is picked up and engaged with the cask trunnionsl
e. Proper engagement of the yoke hooks and trunnions is verified.

#### 10.1.1.3 Moving the Cask to the Preparation Area

a. The cask is rotated to the vertical position, lifted free of the tilting cradle, moved to the preparation area, and set down.

#### 10.1.1.4 Preparing to Load the Cask

a. The cask lifting yoke is removed from the cask and set aside.
b. The cask exterior is cleaned and the inner cavity filled with water.
c. The cask head sleeve nuts are loosened and removed.
d. The yoke is repositioned on the cask and the head removal cables are inspected, attached and adjusted.
e. The cask is lifted and lowered into the loading basin.

f. The cask is lowered to the basin floor and the yoke is disengaged.
g. The cask closure head is removed.
h. The head is raised out of the basin, rinsed, inspected, and stored.
i. The cask cavity is inspected to verify, for irradiated fuel shipments, that the proper fuel baskets are in place or, for irradiated hardware shipments, that the inner cavity is empty.
j. If Group III PWR fuel is to be loaded, remove the center spacer from the cask closure head, unless previously removed, and install the plug in the center cell of the PWR fuel basket, unless previously itnstalled.

10.1.1.5 Loading Irradiated Fuel into the cask

a. The list of irradiated fuel bundles, transfer procedure, and cask loading diagrams are obtained. For Group III PWR fuel, since only six (6) bundles will be loaded, verify that the fuel basket center plug is installed in the center cell.
b. Fuel bundles are grappled one at a time and moved to the appropriate cell in the basket. Fuel assembly seating is verified.
c. The identification marking is verified for each fuel bundle moved and the records are correspondingly marked.
d. After fuel loading is complete, the fuel basket center plug may be removed.

10.1.1.6 Loading Irradiated Hardware

a. A cask liner for the hardware to be transported is placed in the loading basin.
b. The hardware is loaded into the liner using appropriate component spacers to limit the movement of the hardware.
c. The liner cover is installed and the liner lifted and placed in the IF-300 cask.

10.1.1.7 Installing the Cask Closure Head

a. The cask closure head is lifted and the gasket and gasket retaining clips are inspected for damage or looseness. If Group III PWR fuel was loaded, verify that the center head spacer is removed.
b. The head is slowly lowered onto the cask over the guide pins. This operation is closely watched to assure that the head is properly aligned.

10.1.1.8 Returning the Cask to the Preparation Area

a. The yoke is re-engaged with the cask trunnions.
b. The connection is visually inspected to verify proper engagement.
c. The cask is slowly raised (while monitoring radiation levels) until the top of the cask reaches the level of the fuel pool curb.
d. Four cask closure head sleeve nuts are installed, hand tight.
e. The cask is removed from the pool (while again monitoring radiation levels), washed, and placed in the preparation area.
f. The yoke is removed and set aside.

10.1.1.9 Securing the Cask Closure Head

a. Parallelism of the head and cask flanges is tested and head sleeve nuts are torqued to 370 ft-lbs. minimum.
b. After metal-to-metal contact (.007 inch gap or less) is achieved between the head and cask flanges, the head sleeve nuts are lockwired for security.

10.1.1.10 Flushing of the Cask Inner Cavity

a. When desired, the cask inner cavity may be flushed with demineralized water until sample analysis conforms with pre-determined limits. This step is not mandatory.

10.1.1.11 Draining of the Cask Inner Cavity

a. A pressure regulated helium supply connected to the cask cavity vent valve.
b. A drain hose is connected to the cask cavity fill/drain valve and directed into a radwaste drain or back into the pool.
c. After opening the cask cavity vent and fill/drain valves, helium is introduced through the vent valve up to 20 psig.
d. When helium is observed to flow out of the cask cavity drain hose, the fill/drain valve is closed and the cask cavity pressurized to 15 psig.
e. The drain hose is removed.
f. The cask cavity vent valve is closed and the helium supply removed.

10.1.1.12 Assembly Verification Leakage Testing

a. Leakage testing of the cask closure seal, vent valve, fill/drain valve, and rupture disk device is performed with a thermal conductivity sensing instrument. This type of instrument is sensitive to any gas stream having a thermal conductivity different from the ambient air in which the instrument is being used.
b. The test instrument is set up and used according to written procedures and the manufacturer's instructions.
c. With the instrument calibrated to a sensitivity of at least $2 \times 10^{-1}$ $cm^3/sec$ (helium), the vent valve, fill/drain valve, and rupture disk device are checked for indications of leakage.
d. With the instrument calibrated to a sensitivity of at least 2 x 10-2 cm3/sec (helium), the closure seal is checked for indications of leakage. (The sensitivity of this test is increased to account for the dilution, which would occur between a potential point of closure seal leakage and the nearest point of measurement.)
e. If leakage is detected during either of the above checks, the offending components are repaired or replaced and then re-tested for leakage.
f. Valve must be checked to be open if pipe cap or plugs are used.

10.1.1.13 Preparing the cask for Transport of irradiated Fuel

a. Steps 10.1.1.11a thru c are repeated. Nitrogen may be used to supply the third cask volume of inert gas.
b. The supply of helium (nitrogen) is discontinued when at least one additional cask volume has been supplied to the inner cavity. (One cask volume equals 83 cubic feet when shipping irradiated fuel.)
c. The excess helium (nitrogen) within the inner cavity is bled off thru the fill/drain valve until the cavity pressure has decayed to 0 psig. This completes the process of inerting the cask cavity.
d. The vent and fill/drain valve is closed and the connecting hoses and gages are removed.
e. The cask, skid, and rail car are decontaminated in accordance with regulatory requirements.
f. The cask is lifted with the yoke, positioned on the tilting cradle, and lowered to its horizontal position.
g. The yoke is removed.
h. The trunnions are removed and the cask tiedown pins installed.
i. The valve box covers are replaced.
j. The radiological survey of the cask and rail car is completed.

10.1.1.14 <u>Preparing the Cask for Transport of irradiated Hardware</u>

a. A drain hose is connected to the cask cavity fill/drain valve and directed into a radwaste drain or back into the pool.
b. Steps 10.1.1.13c thru j are repeated.

10.1.1.15 <u>Closing the Equipment Skid</u>

a. The cask enclosures are closed, locked, and sealed.

10.1.2 <u>Procedures for Unloading the Package</u>

Operations at the unloading facility are largely the same as loading operations with the major exception being the increased radiological awareness required for receiving a loaded cask. Each unloading facility must provide fully trained personnel and detailed operating procedures to cover all activities.

10.1.2.1 <u>Cask Receiving and Inspection</u>

a. Steps 10.1.1.1a and b are repeated.

10.1.2.2 <u>Preparing for Cask Removal from the Rail Car</u>

a. Steps 10.1.1.2a thru e are repeated.
b. The cask inner cavity temperature may be recorded prior to disconnecting the thermocouple.

10.1.2.3 <u>Moving the Cask to the Preparation Area</u>

a. Step 10.1.1.3a is repeated.

b. If the cask inner cavity temperature was not recorded in step 10.1.2.2b, it is now recorded.

10.1.2.4 Preparing to Unload Irradiated Fuel

a. Steps 10.1.1.4a and b are repeated.
b. A pressure gage is installed on the vent line.
c. The cask cavity is flushed and sampled, giving due consideration to the cask internal temperature and pressure.
d. The cask head sleeve nuts are loosened and all but four are removed.
e. The yoke is repositioned on the cask and the head removal cables are inspected, attached, and adjusted.
f. The cask is lifted from the preparation area and lowered into the loading basin. The last four sleeve nuts are removed while the cask is suspended over the basin with the top of the cask one foot above the water.
g. Steps 10.1.1.4f thru h are repeated.

10.1.2.5 Preparing to Unload Irradiated Hardware

a. If the cask is to be unloaded underwater, steps 10.1.2.4a thru d are followed.
b. If the cask is to be unloaded in air at a waste disposal site, the cask is cleaned and prepared for unloading following a procedure developed by the burial site, reviewed by General Electric, and tested in a dry run at the burial site using unirradiated hardware.
c. The disposal site procedure will specify when and where the cask head sleeve nuts will be loosened and removed.

10.1.2.6 Unloading Irradiated Fuel from the Cask

a. The list identifying fuel bundles to be unloaded is obtained.
b. The fuel bundle identification and location in the cask is verified.
c. The fuel bundles are unloaded one at a time in accordance with the fuel transfer procedure.

10. 1. 2. 7 Unloading Irradiated Hardware from the Cask

a. Unloading of irradiated hardware in air at a disposal site will follow a disposal site procedure.
b. If the irradiated hardware is unloaded underwater, the liner is lifted from the cask and positioned in the water basin as specified by procedure.

10.1.2.8 Installing the Cask Closure Head

a. Steps 10.1.1.7a and b are repeated.

10.1.2.9 Returning the Cask to the Preparation Area

a. If the cask has been unloaded underwater, steps 10.1.1.8a thru f are repeated (without radiation monitoring). Step 10.1.1.8d is optional.

b. If the cask has been unloaded dry, disposal site procedures will be followed.

10.1.2.10 Securing the Cask Closure Head

a. Steps 10.1.1.9a and b are repeated.

10.1.3 Transport of an Empty Cask with Type B Contents

The following operations are typically performed subsequent to transport of irradiated fuel:

10.1.3.1 Draining of the Cask Inner Cavity

a. A pressure regulated helium supply is connected to the cask cavity vent valve.
b. A drain hose is connected to the cask cavity fill/drain valve and directed into a radwaste drain or back into the pool.
c. After opening the cask cavity vent and fill/drain valves, helium is introduced through the vent valve at up to 45 psig.
d. When helium is observed to flow out of the cask cavity drain hose, the fill/drain valve is closed and the cask cavity pressurized to 15 psig.
e. The drain hose is removed.
f. The cask cavity vent valve is closed and the helium supply removed.

10.1.3.2 Assembly Verification Leakage Testing

a. Steps 10.1.1.12a thru e are repeated.

10.1.3.3 Preparing the Empty Cask for Transport

a. A drain hose is connected to the cask cavity fill/drain valve and directed into a radwaste drain or back into the pool.
b. Steps 10.1.1.13c thru j are repeated.

10. 1.3.4 Closing the Equipment Skid

a. The cask enclosures are closed, locked, and sealed.

10.1.4 Transport of an Empty Cask with Less Than Type B Contents

The following operations are typically performed after transport of irradiated hardware:

10.1.4.1 Draining of the Cask Inner Cavity

If the cask has been unloaded underwater:

a. Steps 10.1.3.1a through c are repeated with the exception that the use of air may be substituted for the use of helium.

b. When the applied cover gas is observed to flow out of the cask cavity drain hose, the vent valve is closed and the excess pressure within the cavity is allowed to decay to 0 psig.

c. The f ill/drain valve is closed and the connecting hoses and gages are removed.

10.1.4.2 Assembly Verification Leakage Testing

a. Leakage testing is not performed on IF-300 Type B casks when transporting less than Type B quantities of radioactive materials.

10.1.4.3 Preparing the Empty Cask for Transport

a. Steps 10.1.1.13e through j are repeated.

10.1.4.4 Closing the Equipment Skid

a. The cask enclosures are closed, locked, and sealed.

10.2 MAINTENANCE PROCEDURES

The Chem-Nuclear document "Maintenance Instructions, IF-300 Irradiated Fuel Shipping Cask", GEI-91821, provides maintenance procedures for all functional components of the IF-300 Transportation System.

Maintenance procedures affecting the cask and cask components are summarized below:

10.2.1 Annual Inspections

10.2.1.1 Cask Cavity, Exterior, Head, Etc.

a. The cask cavity, cask exterior, closure head, and related components are inspected annually for signs of damage or degradation.

10.2.1.2 Neutron Shielding

a. The neutron shielding liquid is inspected annually for purity, presence of foreign matter or radioactivity, and if applicable, ethylene glycol percentage.

b. The neutron shielding relief valves are inspected annually for functionality and verification of set pressure.

10.2.2 Annual Component Replacement

10.2.2.1 Rupture Disk

a. The rupture disk is replaced on an annual basis, just prior to annual leakage testing. The rupture disk is inspected for corrosion or other defects during the disk replacement process.

10.2.3 Annual Leakage Testing

10.2.3.1 Cask Cavity

a. Leakage testing of the cask closure seal, vent valve, fill/drain valve, and rupture disk device is performed annually with a thermal conductivity sensing instrument or a helium mass spectrometer leak detector.
b. The test instrument is set up and used according to written procedures and the manufacturer's instructions.
c. With-the Instrument calibrated to a sensitivity of at least $3.5 \times 10^{-4}$ $cm^3$/sec (helium), the vent valve, fill/drain valve, and rupture disk device are checked for indications of leakage.
d. With the instrument calibrated to a sensitivity of at least $3.5^{-5}$x $10^3$ cm /sec (helium), the closure seal is checked for indications of leakage. (The increased sensitivity of this test accounts for the dilution which would occur between a potential point of closure seal leakage and the nearest point of measurement)
e. If leakage is detected during either of the above checks, the offending components are repaired or replaced and then re-tested for leakage.

10.2.3.2 Neutron Shielding

a. The neutron shielding containment with vent/fill valves attached is hydrostatically tested annually at a pressure of 80-100 psig.

10.3 TESTING

This subsection discusses or references the tests which are or have been applied to the cask or to selected cask components. These tests may be initial determinations or they may be periodic.

10.3.1 Tests at Fabrication

10.3.1.1 Cask Inner Cavity

a. The cask inner cavity, closure, closure seal, piping and valves were hydrostatically tested at 600 psig at room temperature.

10.3.1.2 Neutron Shielding Cavity

a. The neutron shielding cavity, piping, vent/fill valves, and closures have been hydrostatically tested at 200 psig at room temperature. Both neutron shielding cavity sections were tested simultaneously.

10.3.1.3 Rupture Disk Device

a. See Section 6.6.

10.3.1.4 Neutron Shielding Containment Relief Valve

a. See Section 6.6 (200 Psig Pressure Relief Valve).

10.3.1.5 Inner Cavity/Neutron Shielding Cavity Fill, Drain, and Vent Valves

a. See Section 6.6 (I-Inch Globe Valve).

10.3.1.6 Thermal Testing

a. See Section 6.7

10.3.1.7 Gamma Shielding

a. During fabrication, the uranium castings were radiographed and then checked, after stacking, by gamma scan techniques to assure that there are no radiation leaks and the uranium material is sufficiently sound such that the requirements in Section 8.5 can be satisfied.

10.3.1.8 Functional Testing

a. Prior to delivery for use, the IF-300 casks were given a complete functional test. This test involved the removal and replacement of the two irradiated fuel baskets, the two heads, rotation and removal of the cask from the equipment skid, operation of the cooling systems, operation of the enclosures, and remote engagement and disengagement of the lifting systems.

10.3.2 Leakage Testing

Leakage testing on the IF-300 cask is done in accordance with the requirements of 10 CFR 71, Regulatory Guide 7.4, and ANSI Std. N14.5.

10.3.2.1 Package Containment Requirements

a. For normal conditions of transport, the containment criteria is "... no loss or dispersal of radioactive contents, as demonstrated to a sensitivity of $10^{-6}$ $A_2$ per hour,..." (Ref. 10.1).

b. For hypothetical accident conditions, the containment criteria is "... no escape of $Kr^{85}$ exceeding 10,000 curies in one week, no escape of other radioactive materials exceeding a total amount $A_2$ in one week..." (Ref. 10.1).

10.3.2.2 Release Rate Limits

a. Using the assumptions and methods listed below, an $A_{2e}$ ($A_2$ equivalent) of 162 curies is calculated for the contents of the IF-300 cask (excluding the $Kr^{85}$ in fuel rod plenums):

- Data from PWR fuel assemblies (Ref.10.2) is used to identify the radionuclides in the crud on LWR fuel assemblies, and estimate the activity associated with each radionuclide at fuel assembly discharge. (Use of PWR crud data is conservative.)

- A decay period of two years from fuel assembly discharge is assumed in calculating $A_{2e}$. Two years corresponds approximately to the minimum cooling time needed for a fuel assembly with normal burn-up to comply with the heat load requirements of the IF-300 cask.

- The following formula (Ref. 10.3) is used to calculate the value of $A_{2e}$:

$$A_{2e} = 1 / \sum \frac{f'}{A_{2e}} \qquad \text{Eq 'n 10-1}$$

where: $A_{2e}$ = Equivalent $A_2$ of the mixture of radionuclides
$f'$ = Activity fraction (at two years) of the individual radionuclides in the mixture.
$A_2$ = Tabulated $A_2$ values for the individual radionuclides in the mixture.

b. Therefore, the release rate limits for the conditions of interest are:

$R_N$ = $A_2$ x 10-6 Ci/Hr
= 4.51 x $10^{-8}$ Ci/Sec (Normal Conditions of Transport)

$R_{A1}$ = $A_2$ Ci/Wk
= 2.68 x $10^{-4}$ Ci/Sec (Hypothetical Accident Conditions) (Excluding $Kr^{85}$)

$R_{A2}$ = 10,000 Ci/Wk
= 1.65 x $10^{-2}$ Ci/Sec (Hypothetical Accident Conditions) ($Kr^{85}$ only)

10.3.2.3 Radionuclide Concentrations

a. Based on the data of Reference 10.2, the activity due to the crud of one 2 year cooled PWR fuel assembly is 1074 Ci. (use of PWR crud data is conservative.)

b. Based on data from W5 NEDO-10084-2, the activity due to the fission gas inventory ($Kr^{85}$) of one 2 year cooled BWR fuel assembly is 727 Ci. (Use of BWR fission gas data is conservative).

c. The minimum cavity free volume for the IF-300 cask is 82.2 $ft^3$ (PWR configuration, Section 6.5).

d. The release fractions of Table X-1 are used in calculating radionuclide concentrations.

For normal conditions, a.) the release fraction for crud is based on an upper bound estimate of the percentage of crud that could remain airborne (i.e. available for release) during transport, and b.) the release fraction for fission gas is based on the thermal calculations of Section 6 and General Electric's extensive experience in shipping irradiated fuel assemblies.

For hypothetical accident conditions, a.) the release fraction for crud is based on the values reported in SAND80-2124 (Ref.10.4), and b.) the release fraction for fission gas is based on the very conservative assumption that all of the fuel assemblies fail as a result of hypothetical accident conditions.

TABLE X-I
CRUD AND FISSION GAS RELEASE FRACTIONS

| | Normal Conditions | Accident Conditions |
|---|---|---|
| Crud, % | 1 | 25 |
| Fission Gas, % | 0 | 100 |

e. Radionuclide concentrations are calculated with the following equation:

$$C_x = Act_x \;/\; V \qquad \text{Eq'n 10-2}$$

where: $C_x$ = Radionuclide concentration for condition "x"
$Act_x$ = Activity available for leakage for condition "x", including the effect of release fraction
$V$ = Cavity free volume

Therefore, the radionuclide concentrations for the conditions of interest are:

$C_N = \underline{3.23 \times 10^{-5}\ Ci/cm^3}$ (Normal Conditions of Transport)

$C_{A1} = \underline{8.08 \times 10^{-4}\ Ci/cm^3}$ (Hypothetical Accident Conditions) (Excluding $Kr^{85}$)

$C_{A2} = \underline{5.61 \times 10^{-3}\ Ci/cm^3}$ (Hypothetical Accident Conditions) ($Kr^{85}$ only)

10.3.2.4 Leakage Rate Limits

a. Leakage rate limits are calculated with the following equation:

$$L_x = R_x / C_x \qquad \text{Eq'n 10-3}$$

where: $L_x$ - Leakage rate limit at condition "x"
$R_x$ - Release rate limit at condition "x"
$C_x$ - Radionuclide concentration at condition "x"

b. Therefore, leakage rate limits for the conditions of interest are:

$L_N$ = $\underline{1.40 \times 10^{-3}\ cm^3/sec}$ (Normal Conditions of Transport)

$L_{A1}$ = $\underline{3.32 \times 10^{-1}\ cm^3/sec}$ (Hypothetical Accident, Conditions) (Excluding $Kr^{85}$)

$L_{A2}$ = $\underline{2.94\ cm^3/sec}$ (Hypothetical Accident Conditions) ($Kr^{85}$ Only)

10.3.2.5 Limiting Leakage Path Diameter

a. Gas leakage for laminar, transitional, or molecular flow modes can be estimated with the following equation (Ref. 10.5):

$$L = 3810\frac{D^3}{a}(323\frac{D}{\eta}(P_u^2 - P_d^2) + \sqrt{\frac{T}{M}}(P_u - P_d)) \qquad \text{Eq'n 10-4}$$

where:
L = leakage rate, $cm^3/sec$
D = leak path diameter, cm
a = leak path length, cm
$P_u$ = upstream pressure, atm-abs
$P_d$ = downstream pressure, atm-abs
$\eta$ = gas Viscosity, cp
T = gas temperature, °K
M = gas molecular wt., amu

b. Upstream pressures and gas temperatures are obtained from the thermal analyses of Section 6.3. LOMC data is used for normal conditions of transport and PFE data is used for hypothetical accident conditions.

c. By assuming a leak path length of 1 cm, leak path diameters associated with the leakage rates of interest ( $L_N$, $L_{A1}$, $L_{A2}$) can be calculated. Table X-2 documents the inputs us 3.n calculating the following leak path diameters:

$D_N$ = $\underline{13.5 \times 10^{-4}\ cm}$ (Normal Conditions of Transport)

$D_{A1}$ = $\underline{21.5 \times 10^{-4}\ cm}$ (Hypothetical Accident Conditions) (Excluding $Kr^{85}$)

$D_{A2}$ = $\underline{37 \times 10^{-4}\ cm}$ (Hypothetical, Accident Conditions) ($Kr^{85}$ only)

Comparing the above leak path diameters, it is concluded that normal conditions of transport (i.e. LOMC) are the most limiting leakage conditions.

TABLE X-2
INPUTS TO LEAKAGE PATH DIAMETER CALCULATIONS

| Inputs | Normal Conditions Of Transport | Hypothetical Accident Conditions (Excluding $Kr^{85}$) | Hypothetical Accident Conditions ($Kr^{85}$ only) |
|---|---|---|---|
| L, $cm^3$/sec | $1.4 \times 10^{-3}$ | $3.32 \times 10^{-1}$ | 2.94 |
| a, cm | 1.0 | 1.0 | 1.0 |
| $P_u$, atm-abs | 2.99 | 19.1 | 19.1 |
| $P_d$, atm-abs | 1.0 | 1.0 | 1.0 |
| $\eta$, cp | 0.025 | 0.029 | 0.029 |
| T, °K | 469 | 548 | 548 |
| M, amu | 28.71 (air) | 28.71 (air) | 28.71 (air) |

10.3.2.6 Reference Air Leakage Rate

a. Using equation $10^{-4}$, a leak path length and diameter of 1 cm and 13.5 microns, respectively, and standard air conditions (25°C and 1 atm-abs), the reference air leakage rate is:

$L_{Ref} = 2.45 \times 10^{-4}$ atm-cm$^3$/sec

This leakage rate is equivalent to $L_N$.

10.3.2.7 Annual Leakage Test Requirements

a. Type B packages must be leakage tested within the preceding 12-month period. The test procedure sensitivity for this "annual" test must be less than or equal to one-half the reference air leakage rate, $L_{Ref,}$ or its equivalent. (Ref. 10.5)

b. For the IF-300 cask, helium at room temperature and 1 atm-gage is used for the annual leakage test. Using equation $10^{-4}$ and a leak path length and diameter of 1 cm and 13.5 microns, respectively, a helium leakage rate of $7.1 \times 10^{-4}$ cm/sec is calculated.

This leakage rate is equivalent to $L_N$ and $L_{Ref,}$

c. Since the leakage test procedure sensitivity must be less than or equal to one-half the calculated leakage rate, the required test procedure sensitivity for the annual leakage test is $3.5 \times 10^{-4}$ cm$^3$/sec (helium) or less.

10.3.2.8 Assembly Verification Leakage Testing

a. Type B packages must also be leakage tested prior to each shipment. The required test procedure sensitivity in this instance, however, is less stringent than that of the annual leakage test. Per Reference 10.6, leakage testing prior to each shipment ...

"should be sensitive enough to preclude the release of an $A^2$ quantity in 10 days, but need not be more sensitive than $10^{-3}$ atm-cm$^3$/sec and can be no less sensitive than $10^{-1}$ atm-cm$^3$/sec."

b. By using the methods presented in 10.3.2.2 thru 10.3.2.4 above, the following parameters are calculated for a release rate of $A_2$ Ci in 10 days:

$$R_A = 1.88 \times 10^{-4} \text{ Ci/sec}$$
$$C_A = C_N = 3.23 \times 10^{-5} \text{ Ci/cm}^3$$
$$L_A = 5.80 \text{ cm}^3\text{/sec}$$

Substituting $L_A$, a leak path length of 1 cm, and LOMC conditions into equation $10^{-4}$, the resulting leak path diameter is 110 microns. Standardized to air at 25°C and 1 atm-abs, this leak path diameter would result in a leakage rate of $9.65 \times 10^{-1}$ atm-cm$^3$/sec (air).

c. For a minimum leakage test procedure sensitivity of $10^{-1}$ atm-cm$^3$/sec (at standard air conditions), the use of equation $10^{-4}$ results in a leak path diameter of 62 microns when a leak path length of I cm is assumed.

d. Helium at 1 atm-gage is used for assembly verification testing. For a leak path length and diameter of 1 cm and 62 microns, respectively, the use of equation $10^{-4}$ and test conditions results in a leakage rate of $2.88 \times 10^{-1}$ cm$^3$/sec (helium).

Thus, for the IF-300 cask, the assembly verification test procedure must have a sensitivity of 2.88 x-$10^{-1}$(helium) to be equivalent to a sensitivity of $1 \times 10^{-1}$ atm cm$^3$/sec at standard air conditions.

## 10.4 REFERENCES

1. 10 CFR 71.51

2. EPRI NP-2735, Expected Performance of Spent LWR Fuel Under Dry Storage Conditions, Battelle, Colombus Laboratories, Dec. 1982.

3. R. H. Jones, R.T. Reese, A Method for Determination of An for a Mixture of Radionuclides, presented at PATRAM '83, New Orleans, LA, May 15-20, 1983.

4. E.L. Wilmot, Transportation Accident Scenarios for Commercial Spent Fuel, SAND80-2124, Sandia National Laboratories, February 1981.

5. ANSI Std. N14.5-1977, American National Standard for Leakage Tests an Packages for Shipment of Radioactive Materials.

6. Letter, dated Nov. 14, 1984, C.E. MacDonald to J.E. VanHoomissen, regarding GE's application for renewal of C of C 9001.

ATTACHMENT 4
REPLACEMENT PAGES
VOLUME 4, APPENDIX D

Table D-1.1

Characteristics of Group III Assemblies

| Parameter | PWR Fuel | BWR Fuel |
|---|---|---|
| Fuel Type | 15x15 SPC PWR for Westinghouse Class 15x15 Reactors | GE-7, 8, 9, 10 & 13 BWR Fuel for General Electric BWR/4 Plant Design |
| Uranium Weight | 437 kg/assembly | 187 kg/assembly |
| Number of Assemblies | 6[a] | 17 channeled |
| Maximum Assembly Average Burnup | 45 GWD/MTU | 45 GWD/MTU |
| Maximum Lattice Average Enrichment[b] | 4.25 wt% $^{235}U$ | 4.25 wt% $^{235}U$ |
| Minimum Blanket Length[c] | 6 inches | 6 inches |
| Minimum Cooling Time | 5 years | 4 years |

[a] The center location in the PWR basket must be left empty. If an assembly is loaded in the center PWR basket location and a peripheral location is instead left empty, the cask will not meet the criticality safety criteria with 6 assemblies at 4.25 wt% $^{235}U$.

[b] The maximum lattice average enrichment is defined as the maximum planar average enrichment of any axial plane from the bottom to the top of the fuel assembly. Although some individual fuel rod enrichments may exceed this enrichment, the average enrichment for every axial slice across the fuel assembly must not exceed the maximum lattice average enrichment.

[c] The minimum length of natural $UO_2$ fuel above and below the enriched portion of the active fuel.

A plug was designed for the PWR fuel basket to prevent inadvertent loading of the center basket location. The plug has an outer lip that rests on top of the basket when the plug is inserted into the basket. The plug extends above the basket 3.625 in and down into the basket approximately 8 in to prevent it from accidentally coming out during fuel loading or transport. The bottom of the plug is approximately 7 in above the top of the active fuel and the plug itself adds only a small amount of steel in a region of low importance from a criticality standpoint. Also, because the plug is located above the empty center basket location, the reactivity effect due to reflection of neutrons back into the fuel is negligible.

- Close full reflection of the containment system by water on all sides or such greater reflection of the system as may be provided by the material of the packaging.

This conclusion is reached with the following conservative assumptions:

1) The fuel is unirradiated (i.e., has 0 MWD/MTU burnup).

2) No credit is taken for the gadolinia present in the fuel before irradiation.

3) Credit is only taken for 75% of the poison present in the basket.

A plug was designed for the PWR fuel basket to prevent inadvertent loading of the center basket location. The plug has an outer lip that rests on top of the basket when the plug is inserted into the basket. The plug extends above the basket 3.625 in and down into the basket approximately 8 in to prevent it from accidentally coming out during fuel loading or transport. The bottom of the plug is approximately 7 in above the top of the active fuel and the plug itself adds only a small amount of steel in a region of low importance from a criticality standpoint. Also, because the plug is located above the empty center basket location, the reactivity effect due to reflection of neutrons back into the fuel is negligible.

## D-2.0 ANALYSIS METHODOLOGY

The approach used in this appendix to address the effect of the change on the pre-1995 IF-300 certification is described in this section.

### D-2.1 Structural Assessment

The requested change deals with increasing the permissible burnup and enrichment of fuel to be shipped in the IF-300 package for specific types of PWR and BWR fuel, which are designated as Group III fuel assemblies. The Group III fuel assemblies have outer dimension envelopes that are compatible with the Group I and II fuel designs, and fuel assembly weights that are bounded by the Group I and II fuel designs already permitted under the current IF-300 Cask C of C (C of C No. 9001). The total weight of a PWR Group III fuel assembly is 653.2 kg, and the maximum assembly weight for the BWR Group III fuel designs is 301.2 kg (GE-7). This results in a total fuel assembly loading of 3,920 kg (6 x 653.2 kg) for the PWR cask and 5,120 kg (17 x 301.2 kg) for the BWR cask. This is below the structural evaluation design basis assembly loading of 4,920 kg for the PWR cask and 5,205 kg for the BWR Cask with the channeled fuel assembly basket. The PWR cask analysis basis fuel loading of 4,920 kg was obtained by subtracting the PWR basket weight (2,050 kg from p. 5-100) from the maximum PWR basket weight with fuel (6,970 kg from p. 5-99). The BWR cask analysis basis fuel loading of 5,205 kg was obtained from Table A-2.2-1 (p. A-2-8) for the channeled BWR fuel basket. Therefore, there is no change in the structural aspects of the contents to be shipped.

Furthermore, since Section 4.0 indicates that the number of moles of residual gas available for release in the cask cavity and the total decay heat load will remain within the pre-1994 licensed limit of 0.5 moles and 40,000 Btu/h, respectively, no change in the internal pressure will occur. Therefore, the proposed change has no effect on the structural analysis of the cask upon which the current C of C (C of C No. 9001) is based.

D-6.4.1.1-B **Summary of Single Package Evaluation Results for the IF-300 Cask with SPC PWR Fuel**

Table D-6.4-3 summarizes the results of the calculations made for the single IF-300 cask with a 12 in close-fitting water reflector around the cask. Most of the MCNP cases were run using 3000 neutrons per cycle for a total of 220 cycles with the first 30 cycles being skipped to ensure that the source distribution was reasonably converged. The maximum $k_{eff}$ including bias and uncertainties is 0.94427 and occurs for full density water conditions in the cask containment with the fuel assemblies shifted outward to maximize the assembly-to-assembly spacing. Note that the effective multiplication of the system decreases rapidly with decreasing containment water density.

An additional MCNP case was run with 6 GIII PWR fuel assemblies, however, in this case an assembly was loaded in the center PWR basket location and a peripheral basket location was left empty instead of the center location. This calculation indicates that the cask will not meet the criticality safety criteria if an assembly is placed in the center location and 6 assemblies at 4.25 wt% $^{235}U$ are loaded into the cask.

It is concluded that a single IF-300 Cask will meet the criticality safety limit of $k_{eff}$ < 0.95 when the cask contains six GIII PWR fuel assemblies with a maximum lattice average enrichment of 4.25 wt% $^{235}U$ loaded in the peripheral PWR basket locations (i.e., with the center basket location empty). Therefore, the IF-300 Cask meets the requirement that a single package must be subcritical if water were to leak into the containment system under the following conditions:

- Most reactive credible configuration consistent with the chemical and physical form of the material.
- Moderation by water to the most reactive credible extent.