71-9001

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17 February, 2000
579-028-00

Mr. Timothy J. Kobetz, Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards, NMSS
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Kobetz:

Subject: Response to Request for Additional Information for Model No. IF-300, CofC No. 9001
Reference: 6 Jan 2000 RAI Response from Chem-Nuclear

As we discussed, Chem-Nuclear made a number of editorial errors in the 6 Jan RAI response. Please make the following changes to correct these errors:

Attachment 7 (Proprietary CSAR pages) of the 6 Jan submittal incorrectly included five pages that do not contain proprietary data. Please remove and discard pages D-6-105, D-6-114, D-6-122, D-6-123, and D-6-132 from Attachment 7.

Please replace pages in the Non-Proprietary RAI Response (6 Jan submittal, Attachment 1) with those in Attachment 1 to this letter. Also, please replace pages in the Proprietary RAI Response (6 Jan submittal, Attachment 2) with those in Attachment 2 to this letter. In preparing the RAI Response and the revised CSAR submitted on 6 Jan, some editorial changes were made in the CSAR that were not made to what was intended to be identical text in the RAI Response. Replacement of the pages in Attachment 1 and 2 will correct these discrepancies; changes on these pages are noted with revision bars in the margins. No changes were made to the technical content on the RAI Response; all changes were editorial. Also note when reviewing the RAI Response, any reference to a particular CSAR page is referring to the page from the CSAR included with the original amendment request submitted in 8 June 1999.

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

Sincerely,

Patrick L. Paquin

Patrick L. Paquin
General Manager – Engineering and HLW

Attachments:
Attachment 1 – Replacement pages for the Non-Proprietary RAI Response
Attachment 2 – Replacement pages for the Proprietary RAI Response

NMSS01 Prop

ATTACHMENT 1

Replacement Pages for
Non-Proprietary RAI Response

STRUCTURAL RAI

Identify the gross weights of the cask when the proposed additional fuel types are included as contents.

Response:
In response to this RAI, Section D-2.1 was revised as shown below. Original paragraph 1, sentence 2 was revised, and Paragraph 1, sentences 3 to 6 below were added. A new paragraph was created from the remaining 2 sentences in the original paragraph.

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 (GIII) 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 an GIII PWR fuel assembly is 653.2 kg, and the maximum assembly weight for the GIII BWR fuel designs 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.

CRITICALITY RAI 1 AND 6

1) Justify modeling the fuel rod spacers as running the full length of the active fuel.
6) Demonstrate that modeling the partial length BWR fuel rods as full length fuel rods is "conservative."

Response:
Additional analysis shows that modeling the spacers as full length is not justified. The following sections were revised to discuss revised models that include discrete axial spacers for the PWR and BWR fuel designs and modeling of the GE-13 fuel with 8 part length rods.

- *The first paragraph of Section D-6.2.1 (p. D-6-7) was revised as follows: (Note: highlighted data is proprietary to Siemens Power Corporation. See the proprietary version of the response to the RAIs for the information.)*

D-6.2.1 Group III PWR Fuel Design Description

The Siemens Power Corporation 15x15 fuel assembly design is the GIII PWR fuel that is evaluated for shipment in the IF-300 cask. This fuel design incorporates GdO_2 integral burnable absorbers and 6 inch top and bottom axial blankets containing natural UO_2 pellets (0.72 wt% ^{235}U) with a [redacted]% theoretical fuel density. The GIII PWR assembly contains 204 fuel rods, 20 guide tubes, and 1 instrument tube. There are also 10 spacer grids located axially along the fuel assembly, with spacer lengths varying from 0.75 in to 2.25 in. The geometry of a typical fuel rod cell is shown in Figure D-6.2-1. Figures D-6.2-2a and D-6.2-2b illustrate the GIII PWR fuel assembly models used in the evaluation, and Tables D-6.2-2 and D-6.2-2-1 summarize the fuel design and axial spacer grid data.

• *Inserted new Table D-6.2-2-1.*

Table D-6.2-2-1

Dimensions of Axial Spacers for Group III Fuel Design

Spacer No.	Spacer Length Inches	Spacer Location Inches[a]
10 (Top)	1.75	152.375
9	1.75	133.745
8	0.75	121.125
7	1.75	107.555
6	0.75	94.935
5	1.75	81.365
4	0.75	68.745
3	1.75	55.175
2	1.75	28.985
1 (Bottom)	2.25	4.52
Total Spacer length	15	

[a] Distance from bottom of fuel assembly lower tie plate to bottom of spacer.

D-6.2.2 Group III BWR Fuel Design Description

• *The 2nd paragraph of Section D-6.2.2 (p. D-6-8) was revised as follows:*

The GE-7 fuel assemblies contain two small water rods offset diagonally in the center of the lattice (Figure D-6.2-3). The GE-8 fuel assemblies contain four small water rods in the center of the lattice (Figure D-6.2-4). Note that two of the four water rods for the GE-8 design are normal water rods, and the other two water rods are empty fuel rods serving as water rods. The GE-9 and GE-10 fuel assemblies have one large water rod located in the center of the lattice (Figure D-6.2-5). The GE-13 fuel assemblies contain two large water rods offset diagonally in the center of the lattice (Figure

D-6.2-5). All of the GIII BWR fuel designs have 7 spacer grids located axially along the fuel assembly except for the GE-13 fuel design which has 8 spacer grids. The spacers have axial lengths of 1.64 in (GE-7, 8), and 1.2 in (GE-9, 10, 13). Table D-6.2-3-1 summarizes the axial spacer grid data for each fuel design.

• *Inserted new Table D-6.2-3-1. (Note: highlighted data is proprietary to General Electric. See the proprietary version of the response to the RAIs for the information.)*

Table D-6.2-3-1

Dimensions of Axial Spacers for the Group III BWR Fuel Designs, inches

Spacer No.	GE-7		GE-8		GE-9		GE-10		GE-13	
	length	Location	length	Location	Length	location	Length	Location	Length	location
8	NA	NA	NA	NA	NA	NA	NA	NA	1.2	[redacted]
7	1.64	146.465	1.64	146.465	1.2	146.675	1.2	146.675	1.2	[redacted]
6	1.64	126.315	1.64	126.315	1.2	126.525	1.2	126.525	1.2	[redacted]
5	1.64	106.165	1.64	106.165	1.2	106.375	1.2	106.375	1.2	[redacted]
4	1.64	86.015	1.64	86.015	1.2	86.225	1.2	86.225	1.2	[redacted]
3	1.64	65.865	1.64	65.865	1.2	66.075	1.2	66.075	1.2	[redacted]
2	1.64	45.715	1.64	45.715	1.2	45.925	1.2	45.925	1.2	[redacted]
1	1.64	25.565	1.64	25.565	1.2	25.775	1.2	25.775	1.2	[redacted]
Total Spacer length	11.48		11.48		8.4		8.4		9.6	

[a] Distance from bottom of fuel assembly nozzle to bottom of spacer.

D-6.4.1 IF-300 Cask with Group III PWR Fuel

• *The 4th paragraph, Section D-6.4.1 (p. D-6-20) was replaced with the following:*

The GIII PWR fuel has 10 spacer grids located axially along the fuel assembly, with spacer lengths varying from 0.75 in to 2.25 in. Table D-6.2-2-1 lists the spacer dimensions and axial locations. The axial spacers were explicitly modeled as a thin layer of Zr-4 surrounding each of the 15x15 lattice locations. The thickness of the spacers was calculated using the total weight of 7.78 kg for all of the spacers, the number of lattice elements (15x15=225), the fuel pin pitch (0.563 in), the total length of the spacers in the axial direction (15 in), and the Zr-4 density (6.565 g/cc). The total weight of the Zr-4 spacers was taken from Table D-6.2-2. This method of modeling assumes that each spacer has the same thickness in the X-Y plane, whereas the spacers actually vary slightly on a mass per unit length basis (i.e., grams per inch of spacer in the axial direction). The model accounts for the total amount of spacer material in the active core region and maintains the axial and radial heterogeneity of the spacers.

D-6.4.2.1 <u>Evaluation of Most Reactive Group III BWR Fuel Design used in CP&L's Brunswick Power Plant</u>

• *Added following sentence after 7th sentence in 3rd paragraph, Section D-6.4.2.1 (p. D-6-34):*

• 7th sentence - "*These fuel rods extend from the lower tie plate (LTP) up to the 6th axial spacer grid." Add the following sentence:* The MCNP model accounts for these 8 part length fuel rods.

• *Replaced 5th paragraph, Section D-6.4.2.1 (p. D-6-34), with the following:*

All of the GIII BWR fuel designs have 7 spacer grids located axially along the fuel assembly except for the GE-13 fuel design which has 8 axial spacers. The spacers have axial lengths of 1.64 in (GE-7, 8), and 1.2 in (GE-9, 10, 13) as shown in Table D-6.2-3-1. The spacers were explicitly modeled as a thin layer of Zr-2 or Zr-4 (see Table D-6.2-3) surrounding each of the 8x8 or 9x9 lattice locations. The thickness of the spacers was calculated using the total weight for all of the zirconium spacers (Table D-6.2-3), the fuel rod pitch (Table D-6.2-3), the number of lattice elements (appropriately reduced for large water holes in GE 9, 10, & 13), the length of the active fuel, and the zirconium density (6.565 g/cc). Note that the weight of inconel in the active region was not included in the calculation of the spacer thickness.

• *Replaced 6th and 7th paragraph, Section D-6.4.2.1 (p. D-6-35), with the following:*

The infinite array cases for the five GE fuel designs were run in MCNP using 2000 neutrons per cycle for a total of

230 cycles. The first 30 cycles were skipped to ensure that the source distribution was reasonably converged. The MCNP runs produced what are essentially k_∞ values for each fuel design since the model used mirror reflective boundary conditions in the radial direction and a 12 inch reflector in the axial directions. Calculations were performed for a water density of 1 g/cc. This water density was selected because preliminary calculations for the cask indicated that the effective multiplication of the cask system is maximized when the cask containment is flooded with full density water (1 g/cc). Calculations were made for both ends of the fuel density range (94.4% - 98% TD) to assess the impact of the fuel density on the effective multiplication of the infinite array of fuel assemblies. An example MCNP input file for the GE-7 fuel type is included in Section D-6.9-Appendix.

The results in Table D-6.4-4 indicate that the GE-7 fuel has the highest k_∞ value, and this occurs for the higher end of the fuel density range (98% TD). The differences between the higher k_∞ values are statistically insignificant, which indicates that the reactivity of these fuel designs are nearly the same. However, the GE-7 fuel with a 98% TD has the highest k_∞ value, therefore, this fuel design will be used for the BWR cask criticality calculations.

• *Revised Table D-6.4-4 with spacers modeled explicitly as follows.*

Table D-6.4-4

Infinite Multiplication Factors for Group III Fuel Designs

Fuel Type	%TD	k_∞	σ	MCNP input file
GE-7	94.4	1.19653	0.00091	ge7ins1
GE-7	98	**1.20119**	0.00092	ge7inas1
GE-8	95	1.19594	0.00081	ge8ins1
GE-8	98	1.19817	0.00082	ge8inas1
GE-9	94.4	1.19535	0.00090	ge9ins1
GE-9	98	1.19976	0.00084	ge9inas1
GE-10	94.4	1.19249	0.00086	ge10ins1
GE-10	98	1.19742	0.00090	ge10ias1
GE-13	94.4	1.19627	0.00089	ge13ins1
GE-13	98	1.20104	0.00085	ge13ias1

CRITICALITY RAI 2

Explain and justify the method for establishing an equivalent poison rod for use in the PWR basket analysis model.

Response:
Additional analysis showed that using an equivalent poison rod was not justified. The revisions below describe the revised basket model where the poison rods are modeled as explicit segments in the axial direction instead of continuously.

- *Revised 5th paragraph (p. D-6-21) of Section D-6.4.1, as follows:*

The poison rods in the PWR basket occupy the space between the 9 spacer discs. The distance between the 9 spacer discs varies and subsequently the length of the poison rods in the PWR basket vary from 8.75 in to 19.62 in, as shown in Table D-6.4-1-1 (page 4-16, DWG 159C523B, sheet 6 of 7). From bottom to top, the PWR basket uses poison rod lengths as follows: one 19.62 in long poison rod, seven 17.31 in poison rods, and one 8.75 in poison rod. Note that the poison rod groups not used in the PWR basket (i.e., groups 2, 3, 5, 6, and 8) are used in the BWR basket design. The poison rods are 304 SS tubes filled with B_4C having an average compacted density of 1.76 +/- 0.13 g/cc natural boron with 18.3 +/- 0.3 wt% B^{10}. The minimum boron density is then 1.63 g/cc, and the minimum B^{10} enrichment is 18 wt% B^{10}. Each poison rod has a 0.19 in packing and two 1.88 in endcaps (page 4-16, DWG 159C523B, sheet 6 of 7). Table D-6.4-1-1 also shows the total lengths of the poison rods that are occupied by B_4C and the distance between the spacer discs.

- *Added the following two new paragraphs after 5th paragraph (p. D-6-21) of Section D-6.4.1:*

The poison rods cover an overall length of 156″ (396.24 cm) in the assembled PWR basket configuration. The distance between the top of the B4C poison in one rod to the bottom of B4C poison in the rod located in the basket directly above it is 5.138″. This distance accounts for the top end cap (1.88″) of the lower rod, the thickness of the spacer disc (1″), the bottom end cap (1.88″) and the packing

(0.19″) for the rod directly above it, and a small gap (3/16″ = 1/8″ + 1/16″) between the poison rod and the spacer disc. Figure D-6.4-1c is an axial view of the poison rod and spacer disc geometry. Over 90% of this length (4.76″) is occupied by steel from the endcaps (2 x 1.88″) and spacer disc (1″). The remaining length is occupied by the poison rod packing and the small gap between the poison rod and the spacer disc. The entire 5.138″ axial gap between the B4C poison rods is modeled as steel in the MCNP model. The model accounts for the important axial heterogeneities that exist in the basket while not becoming excessively complicated.

Tables D-6.4-1-2 through D-6.4-1-4 describe the calculations used to arrive at the B_4C composition for the MCNP model. Table D-6.4-1-2 lists the basic data used in the calculations, and Table D-6.4-1-3 shows the calculation of the weight fractions of each element (i.e., boron and carbon) contained in the B_4C using the lower end B^{10} weight fraction (18%). Table D-6.4-1-4 contains the calculations for the B_4C density and composition used in the MCNP model when taking credit for only 75% of the poison present.

159.95″ = distance from bottom of cask to top of fuel column (derived in 1.1.4)
1.125″ = length of top spacer
[redacted]″ = length of BWR top plenum

2.3 CONCLUSIONS

As derived in 1.1.2, the distance from the bottom of the cask to the top of the BWR basket poison is 170.63″. Under the worst case HAC drop, the fuel column height is [redacted]″ above the bottom of the cask and would not extend past the basket poison.

CRITICALITY RAI 8

Define the term Maximum Lattice Average Enrichment. Also, provide the maximum weight of enriched uranium in each BWR and PWR fuel assembly.

Response:

Added footnote "b" defining maximum lattice average enrichment to Table D-1.1 and D-6.2-1. Emphasized that 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. Also, revised Sections 6.2.1 and 6.2.2 as shown below.

Table D-1.1
Characteristics of Group III Assemblies

Parameter	PWR Fuel	BWR Fuel
Fuel Type	15x15 GIII 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	50 GWD/MTU	45 GWD/MTU
Maximum Lattice Average Enrichment[b]	4.25 wt% ^{235}U	4.25 wt% ^{235}U
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.

D-6.2.1 Group III PWR Fuel Design Description

- *Replaced 1st sentence in 2nd paragraph of section D-6.2.1 (p. D-6-7) with the following:*

The individual fuel rod enrichments may vary throughout the fuel lattice in the radial and axial directions. Although some fuel rod enrichments may exceed the maximum lattice average enrichment of 4.25 wt%, the average enrichment of every axial slice across the fuel assembly must not exceed the maximum lattice average enrichment. In the MCNP model, all fuel pins in the central enriched zone were assumed to contain UO_2 pellets enriched to 4.25 wt% ^{235}U. This is a conservative assumption since many axial planes will be below this average planar enrichment.

D-6.2.2 Group III BWR Fuel Design Description

- *Inserted the following paragraph in the end of Section D-6.2.2 (p. D-6-8) as follows:*

The individual fuel rod enrichments in the GE fuel designs may vary throughout the fuel lattice in the radial and axial directions. Although some fuel rod enrichments may exceed the maximum lattice average enrichment of 4.25 wt%, the average enrichment of every axial slice across the fuel assembly must not exceed the maximum lattice average enrichment. In the MCNP model, all fuel pins in the central enriched zone were assumed to contain UO_2 pellets enriched to 4.25 wt% ^{235}U. This is a conservative assumption since many axial planes will be below this average planar enrichment.

CRITICALITY RAI 9

Correct the reference in Figure D-6.2-7 (page D-6-15) from Table C-6.2-3 to Table D-6.2-3.

Response:
The reference in the Figure was corrected.

CRITICALITY RAI 10

Revise the cask model to use the correct dimensions.

Response:

Revised the cask models and reran all criticality cases. Revised Figure D-6.4-1b (PWR Single Cask Model - Axial View), Table D-6.4-2 (PWR Cask Geometry and Compositions used in the MCNP Models), and Table D-6.4-6 (BWR Cask Geometry and Compositions used in the MCNP Models) as follows:

- *Revised Table D-6.4-6 (p. D-6-47) as follows:*

Table D-6.4-6

BWR Cask Geometry and Compositions used in the MCNP Models

Axial Regions		
	Zone Thickness	
Zone Material	In	cm
Top of Cask		
304 SST	1.5	3.81
Cast DU	3	7.62
304 SS	1	2.54
Cask Inner Cavity Height	180.25	457.835
304 SST	1.25	3.175
Cast DU	3.75	9.525
304 SST	1.5	3.81
Bottom of Cask		
Radial Regions		
Cask Inner Cavity Radius	18.75	47.625
317 SS Inner Shell	0.5	1.27
Cast DU	4	10.16
317 SS Outer Shell	1.56	3.9624

SHIELDING RAI

Specify the neutron and gamma dose rates at the top and bottom of the IF-300 cask. Specify the neutron and gamma dose rates at the cask enclosure surfaces that surround the top lid and the bottom of the IF-300 cask.

Response:

- *Revised Section D-7.1, paragraph 1, 2nd sentence, and added 3rd sentence as shown below. Also added Section D-7.1-1 to include requested dose rates for the IF-300 PWR Cask.*

D-7.1 Group III PWR Fuel

Table D-7.1-1 summarizes the comparisons of the key evaluation parameters for the analysis-basis fuel versus the GIII PWR fuel with a higher burnup and enrichment. Tables D-7.1-2 and D-7.1-5 compare the NCT and HAC dose rates for the current licensed design against the estimated values for the IF-300 cask loaded with 6 GIII PWR fuel assemblies. Section D-7.1-1 discusses how these estimated doses were calculated. Note that Section D-5.1 addresses additional containment evaluation parameters (i.e., crud spallation, total gases and volatiles, and releases from fuel fines) and qualitatively demonstrates that the GIII PWR fuel is bounded by the analysis basis PWR fuel for these parameters.

These results demonstrate that the GIII PWR fuel from Robinson with a burnup of 50 GWD/MTU, and a cooling time of 5 years can be safely transported in the IF-300 shipping cask. A six fuel assembly loading of GIII PWR fuel with a burnup of 50 GWD/MTU, an initial enrichment between 3.45 and 4.25 wt% ^{235}U, and a cooling time of 5 years will not exceed any of the values that form the basis for the current IF-300 shipping cask C of C considering heat load, shielding, and releases from the cask. The higher burnup of these assemblies, when combined with longer cooling times and lower initial uranium loading, results in all parameters, with the exception of the total neutron source strength, being equal to or less than the currently approved operating envelope of the IF-300 shipping cask. Although the neutron dose rates are higher than the analysis-basis values, the total neutron plus gamma dose rates are below the 10 CFR 71 limits and satisfy the limits stated in the existing C of C. Section 6.0 demonstrates that the IF-300 cask loaded with six 4.25 wt% ^{235}U GIII PWR fuel assemblies in the peripheral basket locations satisfies all of the criticality safety criteria contained in 10 CFR 71.

D-7.1-1 Estimates of NCT and HAC Dose Rates for the IF-300 PWR Cask

NCT Dose Rates at the IF-300 PWR Cask Surface

The neutron and gamma dose rates at the surface of the IF-300 cask are estimated using the method described in Appendix B. This method involves calculating the neutron and gamma dose rates by scaling the dose rates for the analysis basis BWR fuel loading by the ratio of the neutron (or gamma) source terms for the PWR fuel to the neutron (or gamma) source terms for the analysis basis BWR fuel. The dose rates used are from the 17 element channeled BWR fuel basket (CSAR 1995) shielding evaluation (Table A-5.4-5, p. A-5-35) since the CSAR does not report the total neutron and gamma dose rates with dry cask inner cavity PWR fuel shipments. Section D-5.1.2 provides the justification for use of the dose rates from the BWR fuel basket. From Section D-5.1.2, the analysis basis assembly neutron source term is exceeded by a factor of 2.55 for the higher burnup and enrichment assemblies, and the gamma source term for the higher burnup and enrichment assemblies is lower than that for analysis basis assembly by a factor of 0.82. Note that the ratios of the assembly source terms (i.e., 2.55 for neutrons, and 0.82 for gammas) were conservatively used instead of the ratios of the total cask source terms (i.e., 2.18 for neutrons, and 0.70 for gammas) because the analysis basis cask loading consists of 7 PWR assemblies, whereas only 6 GIII higher burnup and enrichment assemblies will be loaded. This conservatism was made because the center assembly does not contribute as much to the dose rate outside the cask as do the peripheral assemblies.

Table D-7.1-2 lists the IF-300 Cask surface dose rates for the analysis basis fuel loading and for the cask loaded with 6 GIII PWR fuel assemblies.

Figure A-5.3-1 (p. A-5-22) illustrates the dose point locations (i.e., R_o, R'_o, T_0, B_0).

NCT Dose Rates at the Vehicle Surface and Aluminum Enclosure

The exclusive use dose rate limit of 200 mrem/h applies to the top of the enclosure, the bottom of the rail car, and the vertical planes represented by the outer lateral surfaces of the closed transport vehicle (i.e., edge of the conveyance bed).

Dose attenuation factors (DAFs) for various source locations and dose points are calculated in Section A-5.5.3. These DAFs are multipliers on the surface dose rates to obtain dose rates at distances away from the cask by adjusting for attenuation and geometric effects. From Section A-5.5.3 (p. A-5-52), the distance between the outermost fins of the cask to the aluminum enclosure is 16.125 inches in the cask radial direction, and the distance between the aluminum enclosure and the edge of the conveyance bed is 15 in. The distance between the cask top head region and the surface of the aluminum enclosure in the cask radial direction is 22.41 in. The dose attenuation factor (DAF) for sources in the top nozzle region that contribute to the dose rate at the enclosure is 0.219 (p. A-5-52), and the DAF for sources in the active fuel region that contribute to the dose rate at the enclosure is 0.451 (p. A-5-53). From Table D-7.1-2, the surface dose rate is approximately 15.7 mrem/h on the side of the cask at the cask midplane, and is 666.4 mrem/h on the side of the cask adjacent to the top nozzle region. Applying the DAFs above results in a dose rate of approximately 146 mrem/h (666.4 mrem/h x 0.219) at the enclosure adjacent to the top nozzle region and 7.1 mrem/h (15.7 mrem/h x 0.451) at the enclosure adjacent to the cask midplane, both of which are well below the 200 mrem/h exclusive use dose rate limit. Note that the distance from the cask to the bottom of the rail platform is 33 in, therefore the dose rate at the top of the enclosure, which is only 16.125 in from the cask, bounds the dose rate at the bottom of the rail.

Because the estimated dose rates at the top (125.70 mrem/h) and bottom (192.09 mrem/h) of the cask surface are already below 200 mrem/h, the dose rates at the edge of the conveyance bed will be less than the 200 mrem/h limit for the top and bottom ends of the cask. Table D-7.1-3 summarizes these dose rates.

NCT Dose Rates at 2 m from the Vehicle Surface

Table D-7.1-4 summarizes the dose rates at 2 m from the edge of the conveyance bed. These dose rates were calculated using the same method described above for the cask surface dose rates, and are within the dose rate limit of 10 mrem/h at 2 m from the vehicle surface.

Accident Dose Rates at 1 m from the Cask Surface

Table D-7.1-5 summarizes the dose rates at 1 m from the cask for accident conditions using the same method described above for the cask surface dose rates, and are well within the dose rate limit of 1000 mrem/h at 1 m from the cask.

Table D-7.1-2

IF-300 PWR Cask NCT Dose Rates on the Cask Surface

Location	Current Licensed Design (CSAR 1995), mrem/h			Six Group III PWR Assemblies mrem/h		
	Neutron	Gamma	Total n+g	Neutron[1]	Gamma[2]	Total n+g
Side(cask midplane), R_o	3.03	9.77	12.8	7.72	7.99	15.71
Side(top nozzle), R'_0	44.93	674.6	719.53	114.49	551.92	666.41
Top, T_0	44.93	13.7	58.63	114.49	11.21	125.70
Bottom, B_0	74.11	3.96	78.07	188.85	3.24	192.09
Dose Rate Limit			1000			1000

[1] Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the GIII assembly source term to the analysis basis assembly source term (from Section D-5-3, 2.55 = 6.8E8/2.67E8 n/s).

[2] Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the total gamma source term for six GIII assemblies to that for the design basis BWR fuel (from Section D-5-3, 0.82 = 9.23E15/1.13E16 g/s).

Table D-7.1-3

IF-300 PWR Cask Estimated NCT Dose Rates at the Aluminum Enclosure and the Edge of the Conveyance Bed

Location	mrem/h
Side(cask midplane)	7.1[1]
Side(top nozzle)	146[1]
Top	<125.7[2]
Bottom	<192.1[2]
Dose Rate Limit	200

[1] Estimated dose rate at the top of the aluminum enclosure adjacent to the cask midplane or top nozzle.
[2] Estimated dose rate at the edge of the conveyance bed for the top and bottom of the cask.

Table D-7.1-4

IF-300 PWR Cask NCT Dose Rates at 2 m from the Vehicle Surface

Location	Current Licensed Design (CSAR 1995), mrem/h			Six Group III PWR Assemblies mrem/h		
	Neutron	Gamma	Total n+g	Neutron[1]	Gamma[2]	Total n+g
Side(cask midplane), R_0	0.71	2.33	3.04	1.81	1.91	3.72
Side(top nozzle), R'_0	0.92	8.5	9.42	2.34	6.95	9.30
Top, T_0	1.8	2.82	4.62	4.59	2.31	6.89
Bottom, B_0	2.97	0.95	3.92	7.57	0.78	8.35
Dose Rate Limit			10			10

[1] Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the GIII assembly source term to the analysis basis assembly source term (from Section D-5-3, 2.55 = 6.8E8/2.67E8 n/s).
[2] Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the assembly gamma source term for six GIII assemblies to that for the design basis BWR fuel (from Section D-5-3, 0.82 = 9.23E15/1.13E16 g/s).

Table D-7.1-5

IF-300 PWR Cask Accident Condition Dose Rates at 1 m from the Cask

Location	Current Licensed Design (CSAR 1995), mrem/h			Six Group III PWR Assemblies mrem/h		
	Neutron	Gamma	Total	Neutron[1]	Gamma[2]	Total
Radial CL	163.06	18.10	181.16	415.80	14.84	430.65
10 CFR 71 Limit			1000			1000

[1] Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the GIII assembly source term to the analysis basis assembly source term (from Section D-5-3, 2.55 = 6.8E8/2.67E8 n/s).
[2] Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the assembly gamma source term for six GIII assemblies to that for the design basis BWR fuel (from Section D-5-3, 0.82 = 9.23E15/1.13E16 g/s).

• Revised Section D-7.2, paragraph 1, 2nd sentence, and added 3rd sentence as shown below. Also added section D-7.2-1 to include requested dose rates for the IF-300 BWR Cask.

D-7.2 <u>Group III BWR Fuel</u>

Table D-7.2-1 summarizes the comparisons of the key evaluation parameters for the analysis-basis fuel versus the GIII BWR fuel with a higher burnup and enrichment. Tables D-7.2-2 and D-7.2-5 compare the NCT and HAC dose rates for the current licensed design against the estimated values for the IF-300 cask loaded with 17 GIII BWR fuel assemblies. Section D-7.2-1 discusses how these estimated doses were calculated. Note that Section D-5.2 addresses additional containment evaluation parameters (i.e., crud spallation, total gases and volatiles, and releases from fuel fines) and qualitatively demonstrates that the GIII BWR fuel is bounded by the analysis basis BWR fuel for these parameters.

These results demonstrate that the GIII BWR fuel with a burnup of 45 GWD/MTU, and a cooling time of 4 years can be safely transported in the IF-300 shipping cask. A seventeen fuel assembly loading of GIII BWR fuel with a burnup of 45 GWD/MTU, an initial enrichment between 3.19 and 4.25 wt% ^{235}U, and a cooling time of 4 years will not exceed any of the values that form the basis for the current IF-300 shipping cask C of C considering heat load, shielding, and releases from the cask. The higher burnup of these assemblies, when combined with longer cooling times and lower initial uranium loading, results in all parameters, with the exception of the total neutron source strength, being equal to or less than the currently approved operating envelope of the IF-300 shipping cask. Although the neutron dose rates are higher than the analysis-basis values, the total neutron plus gamma dose rates are below the 10 CFR 71 limits and satisfy the limits stated in the existing C of C. Section 6.0 demonstrates that the IF-300 cask loaded with seventeen 4.25 wt% ^{235}U GIII BWR fuel assemblies satisfies all of the criticality safety criteria in 10 CFR 71.

D-7.2-1 Estimates of NCT and HAC Dose Rates for the IF-300 BWR Cask

NCT Dose Rates at the IF-300 BWR Cask Surface

The neutron and gamma dose rates at the surface of the IF-300 cask are estimated using the method described in Appendix B. This method involves calculating the neutron and gamma dose rates by scaling the dose rates for the analysis basis BWR fuel loading by the ratio of the neutron (or gamma) source terms for the GIII BWR fuel to the neutron (or gamma) source terms for the analysis basis BWR fuel. From Section D-5.2.2, the analysis basis assembly neutron source term is exceeded by a factor of 2.69 for the higher burnup and enrichment assemblies, and the gamma source term for the higher burnup and enrichment assemblies is lower than that for the analysis basis assembly by a factor of 0.89.

Table D-7.2-2 lists the IF-300 Cask surface dose rates for the analysis basis fuel loading and for the cask loaded with 17 channeled GIII BWR fuel assemblies. Figure A-5.3-1 (p. A-5-22) illustrates the dose point locations (i.e., R_o, R'_o, T_0, B_0).

NCT Dose Rates at the Vehicle Surface and Aluminum Enclosure

From Table D-7.2-2, the surface dose rate is approximately 16.9 mrem/h on the side of the cask at the cask midplane, and is 723.6 mrem/h on the side of the cask adjacent to the top nozzle region. Applying the DAFs discussed in Section D-7.1-1 results in a dose rate of approximately 158 mrem/h (723.6 mrem/h x 0.219) at the enclosure adjacent to the top nozzle region and 7.6 mrem/h (16.9 mrem/h x 0.451) at the enclosure adjacent to the cask midplane, both of which are well below the 200 mrem/h exclusive use dose rate limit.

The estimated dose rate of 133 mrem/h at the top of the cask surface is below 200 mrem/h. The estimated dose rate of 202.7 mrem/h at the bottom of the cask is slightly greater than 200 mrem/h. However, because the distance from the bottom of the cask to the edge of the conveyance bed is over 75 inches, the dose rates at the edge of the conveyance bed will be less than the 200 mrem/h limit for both the top and bottom ends of the cask. Table D-7.2-3 summarizes these dose rates.

NCT Dose Rates at 2 m from the Vehicle Surface

Table D-7.2-4 summarizes the dose rates at 2 m from the edge of the conveyance bed. These dose rates were calculated using the same method described above for the cask surface dose rates. Although the estimated dose rate of 10.07 mrem/h slightly exceeds the dose rate limit of 10 mrem/h at 2 m from the vehicle surface, the dose rate estimate is based on a very conservative source term and is not expected to occur during cask use. Note that prior to shipment, dose rate measurements are taken to ensure that the cask meets all applicable transportation dose rate limits.

Accident Dose Rates at 1 m from the Cask Surface

Table D-7.2-5 summarizes the dose rates at 1 m from the cask for accident conditions using the same method described above for the cask surface dose rates, and are well within the dose rate limit of 1000 mrem/h at 1 m from the cask.

Table D-7.2-2

IF-300 BWR Cask NCT Dose Rates on the Cask Surface

Location	Current Licensed Design (CSAR 1995), mrem/h			17 Group III BWR Assemblies mrem/h		
	Neutron	Gamma	Total n+g	Neutron[1]	Gamma[2]	Total n+g
Side(cask midplane), R_0	3.03	9.77	12.8	8.14	8.73	16.87
Side(top nozzle), R'_0	44.93	674.6	719.53	120.74	602.87	723.61
Top, T_0	44.93	13.7	58.63	120.74	12.24	132.99
Bottom, B_0	74.11	3.96	78.07	199.16	3.54	202.70
Dose Rate Limit			1000			1000

[1] Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the total neutron source term for 17 GIII assemblies to that for the design basis BWR fuel (2.687 = 5.02E9/1.87E9 n/s).
[2] Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the total gamma source term for 17 GIII assemblies to that for the design basis BWR fuel (0.894 = 7.06E16/7.9E16 g/s).

Table D-7.2-3

IF-300 BWR Cask Estimated NCT Dose Rates at the Aluminum Enclosure and the Edge of the Conveyance Bed

Location	mrem/h
Side(cask midplane)	7.6[1]
Side(top nozzle)	158[1]
Top	<133[2]
Bottom	<200[2]
Dose Rate Limit	200

[1] Estimated dose rate at the top of the aluminum enclosure adjacent to the cask midplane or top nozzle.
[2] Estimated dose rate at the edge of the conveyance bed for the top and bottom of the cask.

Table D-7.2-4

IF-300 BWR Cask NCT Dose Rates at 2 m from the Vehicle Surface

Location	Current Licensed Design (CSAR 1995), mrem/H			17 Group III BWR Assemblies mrem/h		
	Neutron	Gamma	Total n+g	Neutron[1]	Gamma[2]	Total n+g
Side(cask midplane), R_0	0.71	2.33	3.04	1.91	2.08	3.99
Side(top nozzle), R'_0	0.92	8.5	9.42	2.47	7.60	10.07[3]
Top, T_0	1.8	2.82	4.62	4.84	2.52	7.36
Bottom, B_0	2.97	0.95	3.92	7.98	0.85	8.83
Dose Rate Limit			10			10

[1] Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the total neutron source term for 17 GIII assemblies to that for the design basis BWR fuel (2.687 = 5.02E9/1.87E9 n/s).
[2] Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the total gamma source term for 17 GIII assemblies to that for the design basis BWR fuel (0.894 = 7.06E16/7.9E16 g/s).
[3] Although the estimated dose rate of 10.07 mrem/h slightly exceeds the dose rate limit of 10 mrem/h at 2 m from the vehicle surface, the dose rate estimate is based on a very conservative source term and is not expected to occur during cask use. Note that prior to shipment, dose rate measurements are taken to ensure that the cask meets all applicable transportation dose rate limits.

Table D-7.2-5

IF-300 BWR Cask Accident Condition Dose Rates at 1 m from the Cask

Location	Current Licensed Design (CSAR 1995), mrem/h			17 GroupIII BWR Assemblies mrem/h		
	Neutron	Gamma	Total	Neutron[1]	Gamma[2]	Total
Radial CL	163.06	18.10	181.16	438.14	16.18	454.32
10 CFR 71 Limit			1000			1000

[1] Estimated neutron dose rate was calculated by multiplying the CSAR neutron dose rate from Table A-5.4-5 times the ratio of the total neutron source term for 17 GIII assemblies to that for the design basis BWR fuel (2.687 = 5.02E9/1.87E9 n/s).
[2] Estimated gamma dose rate was calculated by multiplying the CSAR gamma dose rate from Table A-5.4-5 times the ratio of the total gamma source term for 17 GIII assemblies to that for the design basis BWR fuel (0.894 = 7.06E16/7.9E16 g/s).

Containment RAI 1

Show that the package design, with the requested contents meets the containment provisions of 10 CFR 71.13(a)(1). Specifically, show that the radiological source term for the requested contents is bounded by the radiological source term for the previously approved contents. The evaluation should consider the source term contribution due to the crud on the outside of the fuel rod cladding that can become aerosolized, and fuel fines, volatiles, and gases that are released from a fuel rod in the event of a cladding breach. The evaluation should consider both NCT and HAC.

Response:
Revised Section D-4 as shown below to incorporate the results of the containment source term evaluation using the methodology contained in NUREG/CR-6487. The approach taken in the containment evaluation is to compare the potential releasable source terms for the GIII fuel with that for the analysis basis fuel and to demonstrate that the analysis basis source terms bound that for the proposed GIII fuel designs. Using this approach, both NCT and HAC are addressed without requiring the detailed leak rate calculations for HAC and NCT that are described in NUREG/CR-6487.

D-4.0 DETERMINATION OF ACTIVATION PRODUCTS AND CONTAINMENT EVALUATION

The generation of the activation product and containment evaluation source terms is included in this section. Sections D-4.1 and D-4.2 contain the results for the GIII PWR fuel and the GIII BWR fuel, respectively.

D-4.1 Group III PWR Fuel

D-4.1.1 Upper Nozzle Hardware Activation Product Source

Activation analysis is performed to determine the impact of higher burnup on the ^{60}Co content of the upper nozzle of the Robinson Plant's fuel assemblies with a burnup of 50 GWD/MTU and a cooling time of 5 years.

$$C_{vol\&gas} = \frac{f_B W_R N_R N_A (A_V f_V + A_G f_G)}{V}$$

Where,

f_B = fraction of fuel rods that develop a cladding breach
W_R = mass of fuel in the fuel rod
N_R = number of rods per assembly
N_A = number of assemblies per cask
f_V = fraction of volatiles in a fuel rod released if the rod develops a cladding breach
f_G = fraction of gas that would escape from a fuel rod tht developed a cladding breach
A_V = specific activity of the volatiles in a fuel rod
A_G = specific activity of the gas in a fuel rod

Using the ORIGEN2 computer code, the radioisotope inventory of the gases and volatiles was calculated for each fuel type. The ORIGEN2 models are discussed in Section D-4.1.2.2. Table D-4.1.2.3 lists the radioisotope inventory for gases and volatiles after a 2 year cooling time for a single fuel assembly. This comparison indicates that the GIII PWR fuel cooled 5 years has a lower inventory of gases and volatiles than the analysis basis PWR fuel assembly cooled 2 years. Therefore, the activity of the gases and volatiles from the analysis basis PWR fuel bounds that for the GIII PWR fuel. Note that the radioisotope inventory for the gases from the 4.25% GIII PWR fuel bounds that for the 3.45% fuel, and the radioisotope inventory for the volatiles from the 3.45% GIII PWR fuel bounds that for the 4.25% fuel.

D-4.2 Results for Group III BWR Fuel

D-4.2.1 Upper Nozzle Hardware Activation Product Source

Activation analysis is performed to determine the impact of higher burnup of the ^{60}Co content of the upper nozzle of the Brunswick Plant's fuel assemblies with a burnup of 45 GWD/MTU and a cooling time of 4 years.

An activation analysis based on a unit mass (one kg) of cobalt for a burnup of 35 GWD/MTU at 3 years is contained in Appendix D-9.2, ORIGEN2 run BWRCO265. Also the activation analysis based on a unit mass (1 kg) of cobalt for a burnup of 45 GWD/MTU at 4 years is contained in Appendix D-7.3. ORIGEN2 run BWRCO319. A comparison of the results is included in Section D-5.

D-4.2.2 Containment Evaluation Source Terms

The increase in the fuel assembly burnup limit from 35 GWD/MTU to 45 GWD/MTU and enrichment limit from 4 wt% ^{235}U to 4.25 wt% ^{235}U will impact the releasable source terms from the IF-300 Package. The releasable source terms include 1) crud spallation from fuel rods, 2) release of fines from cladding breaches, and 3) source activity from gases and volatiles released due to cladding breaches. Each of these sources is addressed below. The determination of fission gas product moles is included in Section D-4.2.2.4.

The following subsections contain an evaluation of the containment sources term for fuel with an enrichment from 3.19 to 4.25 wt% ^{235}U, a fuel assembly burnup of 45 GWD/MTU, and a minimum 4 year cooling time.

D-4.2.2.1 Crud Spallation from fuel rods

Following the methodology in NUREG/CR-6487 (NRC 1996) the crud spallation fraction (f_C) is independent of the fuel type. The crud surface activity (S_C) depends only on whether the fuel is PWR or BWR fuel, which leaves the total fuel rod surface area as the only fuel dependent parameter. Therefore, the total fuel rod surface area for the GIII BWR fuel will be compared to that for the analysis basis BWR fuel to demonstrate that the crud density for the GIII BWR fuel is bounded by that for the analysis basis BWR fuel.

Table D-4.2.2.3 lists the radioisotope inventory for gases and volatiles after a 3 year cooling time for a single fuel assembly. This comparison indicates that the GIII BWR fuel cooled 4 years has a lower inventory of gases and volatiles than the analysis basis BWR fuel assembly cooled 3 years. Therefore, the activity of the gases and volatiles from the analysis basis BWR fuel bounds that for the GIII BWR fuel. Note that the radioisotope inventory for the gases from the 4.25% GIII BWR fuel bounds that for the 3.19% fuel, and the radioisotope inventory for the volatiles from the 3.19% GIII BWR fuel bounds that for the 4.25% fuel.

Table D-4.1.2.2 Radioisotope Inventories of Analysis Basis and Group III PWR Fuel to Determine the Specific Activity of the Fuel

Analysis Basis PWR - 3.5%, 2 yr cooled		Group III PWR - 3.45%, 5 yr cooled	
Isotope	Ci[a]	Isotope	Ci[a]
H 3	3.46E+02	H 3	3.72E+02
FE55	2.72E+00	MN54	2.48E+00
CO60	5.79E+01	FE55	2.55E+02
ZN65	7.61E+00	CO60	9.89E+02
KR85	5.34E+03[b]	NI63	6.55E+01
SR89	1.93E+01	KR85	3.97E+03
SR90	3.49E+04	SR90	3.81E+04
Y 90	3.49E+04	Y 90	3.81E+04
Y 91	9.93E+01	ZR93	1.12E+00
ZR95	2.91E+02	TC99	8.04E+00
NB95	6.68E+02	RU106	1.14E+04
NB95M	2.16E+00	RH106	1.14E+04
RU103	2.02E+00	AG110M	2.71E+01
RH103M	1.83E+00	CD113M	3.92E+01
RU106	6.36E+04	SN119M	2.74E+01
RH106	6.36E+04	SB125	4.27E+02
AG110	3.33E+00	TE125M	1.04E+02
AG110M	2.50E+02	SB125	2.69E+03
SN123	3.41E+01	TE125M	6.56E+02
SB125	4.01E+03	CS134	2.78E+04
TE125M	9.78E+02	CS137	6.03E+04
TE127	6.70E+01	BA137M	5.70E+04
TE127M	6.84E+01	CE144	6.19E+03
CS134	4.07E+04	PR144	6.19E+03
CS137	4.87E+04	PR144M	7.43E+01
BA137M	4.61E+04	PM147	1.41E+04
CE144	9.19E+04	SM151	2.33E+02
PR144	9.19E+04	EU152	2.70E+00
PR144M	1.10E+03	EU154	5.80E+03
PM147	3.78E+04	EU155	2.93E+03
SM151	1.77E+02	U 237	1.49E+00
EU154	3.85E+03	NP239	2.49E+01
EU155	2.17E+03	PU238	3.24E+03
U 237	1.28E+00	PU239	1.90E+02
NP239	8.50E+00	PU240	3.41E+02
PU238	1.31E+03	PU241	6.08E+04
PU239	1.55E+02	PU242	1.46E+00
PU240	2.35E+02	AM241	6.36E+02
PU241	5.23E+04	AM242M	1.16E+01
AM241	2.33E+02	AM242	1.15E+01
AM242M	6.83E+00	AM243	2.49E+01
AM242	6.80E+00	CM242	2.55E+01
AM243	8.50E+00	CM243	2.45E+01
CM242	8.82E+02	CM244	4.80E+03
CM243	8.60E+00	Total	3.59E+05
CM244	9.54E+02	Specific Activity, Ci/g	0.82[c]
Total	6.30E+05		
Specific Activity, Ci/g	1.35[c]		

[a] Note that isotopes with activities less than 1 Ci are not included since they have a negligible impact on the total.

[b] The analysis basis Kr-85 inventory available for release from the cask during hypothetical accident conditions is a total of 13,086 Ci. This is equivalent to a Kr-85 fuel assembly inventory of 5.34E+03 Ci (13,086 Ci/0.35/7) for the analysis basis loading of 7 PWR assemblies per cask. A factor of 0.35 is applied to the rod inventory to determine the amount of Kr-85 that is available for release.

[c] Specific activity is based on a U mass loading of 465 kg for the analysis basis PWR fuel and 437 kg for the GIII PWR fuel.

Table D-4.1.2.3 Comparison of Analysis Basis (AB) PWR Fuel and Group III PWR Fuel Gases and Volatiles

Isotope	Gases, Ci		Isotope	Volatiles	
	AB	Group III[a]		AB	Group III[b]
H 3	3.46E+02	3.59E+02	SR89	1.93E+01	4.37E-06
KR85	5.34E+03[c]	4.24E+03	SR90	3.49E+04	3.81E+04
XE133	1.47E-36	0.00E+00	RU103	2.02E+00	8.50E-09
Total	5.69E+03	4.60E+03	RU106	6.36E+04	1.14E+04
			CS134	4.07E+04	2.78E+04
			CS136	4.49E-13	4.19E-38
			CS137	4.87E+04	6.03E+04
			Total	1.88E+05	1.38E+05

[a] The 4.25% GIII PWR fuel has the largest inventory of gases.

[b] The 3.45% GIII PWR fuel has the largest inventory of volatiles.

[c] The analysis basis Kr-85 inventory available for release from the cask during hypothetical accident conditions is a total of 13,086 Ci. This is equivalent to a Kr-85 fuel assembly inventory of 5.34E+03 Ci (13,086 Ci/0.35/7) for the analysis basis loading of 7 PWR assemblies per cask. A factor of 0.35 is applied to the rod inventory to determine the amount of Kr-85 that is available for release.

Table D-4.2.2.2

Radioisotope Inventories of Analysis Basis and Group III BWR Fuel to Determine the Specific Activity of the Fuel

Analysis Basis BWR - 2.65%, 3 yr cooled		Group III BWR - 3.19%, 4 yr cooled	
Isotope	Ci[a]	Isotope	Ci[a]
H 3	1.42E+02	H 3	1.49E+02
CO60	2.32E+01	MN54	3.34E+00
KR85	2.20E+03[b]	FE55	2.00E+02
SR90	1.27E+04	CO60	6.03E+02
Y 90	1.27E+04	NI63	4.04E+01
ZR95	1.48E+00	KR85	1.53E+03
NB95	3.40E+00	SR90	1.45E+04
RU106	1.28E+04	Y 90	1.45E+04
RH106	1.28E+04	RU106	6.83E+03
AG110M	4.91E+01	RH106	6.83E+03
CD113M	1.18E+01	AG110M	2.40E+01
SN123	1.50E+00	CD113M	1.55E+01
SB125	1.29E+03	SB125	1.14E+03
TE125M	3.14E+02	TE125M	2.78E+02
TE127	2.08E+00	CS134	1.33E+04
TE127M	2.12E+00	CS137	2.32E+04
CS134	1.33E+04	BA137M	2.20E+04
CS137	1.99E+04	CE144	4.26E+03
BA137M	1.89E+04	PR144	4.26E+03
CE144	1.06E+04	PR144M	5.11E+01
PR144	1.06E+04	PM147	6.78E+03
PR144M	1.28E+02	SM151	9.91E+01
PM147	9.45E+03	EU152	1.89E+00
SM151	7.81E+01	EU154	2.35E+03
EU154	1.84E+03	EU155	1.32E+03
EU155	1.01E+03	NP239	1.04E+01
NP239	7.14E+00	PU238	1.37E+03
PU238	7.88E+02	PU239	7.32E+01
PU239	6.26E+01	PU240	1.24E+02
PU240	1.03E+02	PU241	2.73E+04
PU241	2.53E+04	AM241	2.46E+02
AM241	1.71E+02	AM242M	7.92E+00
AM242M	5.54E+00	AM242	7.88E+00
AM242	5.51E+00	AM243	1.04E+01
AM243	7.14E+00	CM242	4.20E+01
CM242	1.38E+02	CM243	1.21E+01
CM243	7.82E+00	CM244	2.09E+03
CM244	1.08E+03	Total	1.56E+05
Total	1.69E+05	Specific Activity, Ci/g	0.83
Specific Activity, Ci/g	0.85[c]		

[a] Note that isotopes with activities less than 1 Ci are not included since they have a negligible impact on the total.

[b] The analysis basis Kr-85 inventory available for release from the cask during hypothetical accident conditions is a total of 13,086 Ci. This is equivalent to a Kr-85 fuel assembly inventory of 2.20E+03 Ci (13,086 Ci/0.35/17) for the analysis basis loading of 17 channeled BWR assemblies per cask. A factor of 0.35 is applied to the rod inventory to determine the amount of Kr-85 that is available for release.

[c] Specific activity is based on a U mass loading of 198 kg for the analysis basis BWR fuel and 187 kg for the GIII BWR fuel.

Table D-4.2.2.3 Comparison of Analysis Basis(AB) BWR Fuel and Group III BWR Fuel Gases and Volatiles

Isotope	Gases		Isotope	Volatiles	
	AB	Group III[a]		AB	Group III[b]
H 3	1.42E+02	1.42E+02	SR89	3.03E-02	1.84E-04
KR85	2.20E+03[c]	1.67E+03	SR90	1.27E+04	1.45E+04
Total	2.34E+03	1.81E+03	RU103	9.63E-04	1.46E-06
			RU106	1.28E+04	6.83E+03
			CS134	1.33E+04	1.33E+04
			CS136	7.23E-22	3.96E-30
			CS137	1.99E+04	2.32E+04
			Total	5.88E+04	5.79E+04

[a] The 4.25% GIII BWR fuel has the largest inventory of gases.
[b] The 3.19% GIII BWR fuel has the largest inventory of volatiles.
[c] The analysis basis Kr-85 inventory available for release from the cask during hypothetical accident conditions is a total of 13,086 Ci. This is equivalent to a Kr-85 fuel assembly inventory of 2.20E+03 Ci (13,086 Ci/0.35/17) for the analysis basis loading of 17 channeled BWR assemblies per cask. A factor of 0.35 is applied to the rod inventory to determine the amount of Kr-85 that is available for release.

- *Added reference to Section D-8 (p. D-8-1):*

NRC, 1996, *Containment Analysis for Type B Packages Used to Transport Various Contents,* NUREG/CR-6487, U.S. Nuclear Regulatory Commission, Washington, D.C.

Containment RAI 2

Provide the technical bases and evidence for your assumption that the fission gas product volume varies linearly between 45 and 55 GWD/MTU. (Pg. D-3-13)

Response:

- *Revised the Fission Gas Product Moles calculation (originally Section D-3.3.6, now Section D-4.1.2.4) to conservatively use the fission gas product volume at 55 GWD/MTU for the value at 50 GWD/MTU as follows:*

D-4.1.2.4 Fission Gas Product Moles Calculation

The ^{131m}Xe and ^{85}Kr are the two main fission gas products available for release in the cask inner cavity. The volume of ^{131m}Xe at burnups of 45 GWD/MTU and 55 GWD/MTU for a 15X15 PWR fuel assembly in a Westinghouse 15x15 class reactor is 206 liters/assembly and 259 liters/assembly at STP respectively (DOE/ET/34014-10 1983). Similarly, the volume of ^{85}Kr at burnups of 45 GWD/MTU and 55 GWD/MTU for a 15X15 PWR fuel assembly in a Westinghouse 15x15 class reactor is 21.9 liters/assembly and 27.4 liters/assembly at STP respectively (DOE/ET/34014-10, 1983). Because the variation in fission gas product volume between 45 and 55 GWD/MTU is not known, the higher values at 55 GWD/MTU will conservatively be used in this evaluation. The volume of fission gases for the licensing basis burnup of 35 GWD/MTU is 155 liters/assembly of ^{131m}Xe and 16.7 liters/assembly of ^{85}Kr for a 15X15 PWR fuel assembly in a Westinghouse 15x15 class reactor (DOE/ET/34014-10 1983).

Therefore, the volume of the additional fission gas products due to an increase in burnup from 35 GWD/MTU to 55 GWD/MTU is 259-155 = 104 liters/assembly of ^{131m}Xe and 27.4-16.7 = 10.7 liters/assembly of ^{85}Kr, respectively. For the six PWR fuel assemblies in the cask inner cavity, the total volume of the additional fission gas products due to increase in burnup from 35 GWD/MTU to 55 GWD/MTU is (104 + 10.7) liters/assembly * 6 = 688 L (24.3 ft^3). It is expected that no fuel rods will rupture. If in the extreme case it is assumed that all the fuel rods from all six PWR assemblies in the cask inner cavity rupture, then the residual gases in the fuel rods will increase the cask inner cavity pressure.

The volume of the additional fission gas products at the reactor operating conditions is calculated as follows:

$$\frac{P_1V_1}{T_1} = \frac{P_2V_2}{T_2}$$

$$\frac{14.7 \text{ psia} * 24.3 \text{ ft}^3}{(32+460) \text{ R}} = \frac{2500 \text{ psia}*V_2}{(900+460) \text{ R}}$$

which results in $V_2 = 0.39$ ft^3, where V_2 is the volume of the additional fission gas at the reactor condition of 2500 psia (end-of-life rod pressure) and 900 F (rod gas temperature) (CSAR 1985).

The total fission gas product volume at 50 GWD/MTU available for release is then 1.5 +0.39 = 1.89 ft^3, where 1.5 ft^3 is the total gas volume in all rods available for release at the existing licensed condition of 35 GWD/MTU burnup (CSAR 1985). The number of moles, n, of residual gas that could be released into the cask cavity is estimated to be 0.32 moles which is well below the analysis basis value of 0.5 moles (page 6-50 of CSAR 1985):

$$N = \frac{P_rV_g}{RT_r} = \frac{2500*1.89}{10.73*(900+460)} = 0.32 \text{ moles}$$

- *Also, changed value D-7.1-1 from 0.31 to 0.32 moles.*

Containment RAI 3

Justify the use of 1 cm (input "a" to equation 10-4) as the leakage path length in the containment analysis. Provide a diagram of the leakage path through either the vent and drain valves or the head seal whichever is considered bounding.

Response:
The leakage path length of 1 cm was chosen as a nominal value. It does not correspond to a physical dimension of the cask. In determining the reference leakage rate to determine the corresponding leakage test limit, the leak path length is assigned a nominal value to determine the leakage hole diameter. Using the calculated leakage hole diameter, the reference leak rate is determined. For example, an order of magnitude change in the path length, i.e., to 10 cm or 0.1 cm, changes the reference leakage rate by less than 8%. See Examples 2 and 31 of ANSI N14.5-1997.

Containment RAI 4

Provide the missing section D-3.5.

Response:
Section D-3.5 was incorrectly referenced in Section D-2.3. Section D-2.3 has been revised to reference Section D-4, which contains the fission gas product moles calculations.

ATTACHMENT 2
Replacement Pages for
Proprietary RAI Response