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September 18, 2008

U.S. Nuclear Regulatory Commission
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Attn: Document Control Desk

Subject: NAC-UMS® Final Safety Analysis Report (FSAR) – Changes to Chapter 5, Shielding Evaluation

Docket No. 72-1015

Reference:

1. Request for an Amendment of Certificate of Compliance (CoC) No. 1015 for the NAC-UMS® Universal Storage System to Incorporate High Burnup PWR Fuel as Approved Contents and Implement Changes to the Technical Specifications, NAC International, September 22, 2006
2. Request for Additional Information for Review of an Amendment for the Certificate of Compliance No. 1015 for the NAC-UMS® Universal Storage System (Docket 72-1015), U.S. Nuclear Regulatory Commission, May 30, 2007
3. Response to Request for Additional Information for an Amendment for Certificate of Compliance No. 1015 for the NAC-UMS® Universal Storage System (TAC No. L24032), Docket No. 72-1015, NAC International, September 6, 2007
4. Submittal of Supplemental Information to the Response to Request for Additional Information for an Amendment for Certificate of Compliance No. 1015 for the NAC-UMS® Universal Storage System, NAC International, September 26, 2007
5. NRC/NAC May 29, 2008 conference call to discuss three additional shielding questions emailed by Randy Hall on May 28, 2008
6. NRC/NAC June 6, 2008 conference call to discuss outstanding shielding questions from the May 29, 2008 conference call

This communication confirms NAC International's (NAC) commitment to incorporate clarifications to Chapter 5 of the UMS FSAR that were discussed in the June 6, 2008 NRC/NAC conference call (Reference #6). At that time, NAC agreed to change text in Chapter 5 to clarify how one-dimensional and three-dimensional analyses for shielding (dose rates) were used in the UMS FSAR and provide a copy of the changed FSAR pages in advance of the regularly scheduled 24-month update. Changes were made to the following: Section 5.1, Discussion and Results; Section 5.5.4.1, PWR and BWR Assembly Minimum Cooling Times; and Table 5.5-2, Design Basis Assembly Dose Rate Limit (mrem/hr). In addition, an editorial change was made

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to Section 5.4.3.1, Vertical Concrete Cask Dose Rates, the subsection titled Three-Dimensional Dose Rates for Concrete Cask Containing BWR Fuel, since the dose rate previously listed and the conclusion statement were inconsistent with the remainder of the chapter. As agreed in the June 6, 2008 conference call, NAC has made these changes to the UMS FSAR via the 10 CFR 72.48 Determination process. A summary of the 10 CFR 72.48 Determination prepared in order to make these changes is included as Attachment 1 to this letter.

Enclosed for your information are the changed pages for Chapter 5. The pages updated via the 72.48 process are dated August 2008 and the header reads "DCR(L) 790-FSAR-6F." Revision bars mark the changes made to these pages. Please note that there are several pages of text flow that contain no revision bars. Also note that there are two Revision 3 pages dated March 2004 that also contain revision bars, which are provided as the front or back of the revised pages. These pages will fit correctly into the copy of the UMS FSAR currently held by the NRC.

Upon approval of the UMS high burnup amendment (Revision 5 to the UMS Certificate of Compliance), NAC will prepare Revision 7 of the UMS FSAR and incorporate these changes that have been made to Chapter 5 via the 72.48 process.

If you have any comments or questions, please contact me on my direct line at (678) 328-1274.

Sincerely,

Anthony L. Patko
Director, Licensing
Engineering

Enclosures

NAC INTERNATIONAL

Attachment 1

72.48 Determination ID #NAC-08-UMS-014

Change Description:

Revises Chapter 5 of the FSAR to clarify how one-dimensional and three-dimensional analyses for shielding (dose rates) were used in the UMS® FSAR. Changes have been made to Section 5.1, Discussion and Results; Section 5.5.4.1, PWR and BWR Assembly Minimum Cooling Times; and Table 5.5-2, Design Basis Assembly Dose Rate Limit (mrem/hr). In addition, an editorial change has been made to Section 5.4.3.1, Vertical Concrete Cask Dose Rates, the subsection titled Three-Dimensional Dose Rates for Concrete Cask Containing BWR Fuel.

Chapter 5 pages changed: 5.1-2 through 5.1-5, 5.4-5, 5.5-3 and 5.5-4

Source of change: 72.48 Determination ID #NAC-08-UMS-014

Originating document: DCR(L) 790-FSAR-6F

This DCR(L) clarifies how the results of one-dimensional and three-dimensional analyses for shielding (dose rates) were used in the UMS® FSAR. Editorial corrections were also made on page 5.4-5, as the dose rate previously listed and the conclusion statement were inconsistent with the remainder of the chapter.

5.0 SHIELDING EVALUATION

Specific dose rate limits for individual casks in a storage array are not established by 10 CFR 72 [1]. Annual dose limit criteria for the independent spent fuel storage installation (ISFSI) controlled area boundary are established by 10 CFR 72.104 and 10 CFR 72.106 for normal conditions and for design basis accidents. These regulations require that, for an array of casks in an ISFSI, the annual dose to an individual outside the controlled area boundary must not exceed 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem to any other organ during normal operations. For a design basis accident, the dose to an individual outside the controlled area boundary must not exceed 5 rem to the whole body. The ISFSI must be at least 100 meters from the owner controlled area boundary. In addition, the occupational dose limits and radiation dose limits established in 10 CFR Part 20 (Subparts C and D) [2] for individual members of the public must be met.

This chapter describes the shielding design and the analysis used to establish bounding radiological dose rates for the storage of various types of PWR and BWR fuel assemblies. The analysis shows that the Universal Storage System meets the requirements of 10 CFR 72.104 and 10 CFR 72.106 when the system is configured and used in accordance with the design basis established by this Safety Analysis Report.

The Universal Storage System compliance with the requirements of 10 CFR 72 with regard to annual and occupational doses at the owner controlled area boundary is demonstrated in Section 10.3 and 10.4.

5.1 Discussion and Results

The transfer cask is provided in either the Standard or Advanced configuration. Canister handling, fuel loading and canister closing are operationally identical for either transfer cask configuration.

The Standard and Advanced transfer casks have a radial shield comprised of 0.75 inch of low alloy steel, 4.00 inches of lead, 2.75 inches of solid borated polymer (NS-4-FR), and 1.25 inches of low alloy steel. An additional 0.625 inch of stainless steel shielding is provided, radially, by the canister shell. Gamma shielding is provided primarily by the steel and lead layers, and neutron

shielding is provided primarily by the NS-4-FR. The transfer cask bottom shield design is a solid section of 7.5 inches of low alloy steel and 1.5 inches of NS-4-FR. The top shielding of the transfer cask is provided by the stainless steel canister shield and structural lids, which are 7 inches and 3 inches thick, respectively. In addition, 5 inches of steel is used as temporary shielding during welding, draining, drying, helium backfill, and other operations related to closing the canister. This temporary shielding is removed prior to storage.

The Advanced transfer cask incorporates a trunnion support plate that allows it to lift a heavier canister. The support plate has no significant shielding impact due to its location above the trunnion. The evaluations and results provided for the Standard transfer cask are, therefore, applicable to the Advanced transfer cask.

The vertical concrete cask radial shield design is comprised of a 2.5-inch thick carbon steel inner liner surrounded by 28.25 inches of concrete. Gamma shielding is provided by both the carbon steel and concrete, and neutron shielding is provided primarily by the concrete. As in the transfer cask, an additional 0.625 inch thickness of stainless steel radial gamma shielding is provided by the canister shell. The concrete cask top shielding design is comprised of 10 inches of stainless steel from the canister lids, a shield plug containing a 1-inch thickness of NS-4-FR or 1.5 inches of NS-3 and 4.1 inches of carbon steel, and a 1.5-inch thick carbon steel lid. Since the bottom of the concrete cask rests on a concrete pad, the cask bottom shielding is comprised of 1.75 inch of stainless steel from the canister bottom plate, 2 inches of carbon steel (pedestal plate) and 1 inch of carbon steel cask base plate. The base plate and pedestal base are structural components that position the canister above the air inlets. The cask base supports the concrete cask during lifting, and forms the cooling air inlet channels at the cask bottom. An optional carbon steel supplemental shielding fixture, shown in Drawing 790-613, may be installed to reduce the radiation dose rates at the air inlets.

The spent fuel that may be stored in the Universal Storage System is divided into five classes, three PWR and two BWR, depending on the length of the fuel assembly. The transportable storage canister, transfer cask, and vertical concrete cask are provided in five lengths, corresponding to the lengths of the fuel assemblies.

The shielding analysis is based on the use of bounding dose rates for the design basis PWR and BWR fuel assembly, and its associated canister, transfer cask, and concrete cask. Shielding evaluations are performed for the transfer cask with both wet and dry canister cavities. The wet

canister cavity condition occurs during the welding of the shield lid. During the welding of the structural lid, the canister cavity is assumed to be completely dry. Note that in the wet canister condition, the modeled water level is the base of the upper end fitting. Shielding evaluations for the concrete cask assume a dry cavity.

Site-specific fuels, which may have configurations or parameters that are not considered in the design basis fuels, are described in Section 5.6. As described in Section 5.6, the site-specific fuels must either be shown to be bounded by the evaluation of the design basis fuel or be separately evaluated to establish limits that are maintained by administrative controls.

Shielding evaluations to determine maximum system dose rates for the range of fuel types and allowable enrichment and burnup combinations rely on a three-step approach. In the initial step, one-dimensional evaluations consider all assembly types intended for storage in the Universal Storage System at a fixed burnup, initial enrichment (referred to as shielding design basis) to determine a bounding, i.e., shielding design basis assembly design. In the second step, the design basis assembly design is evaluated using a three-dimensional Monte Carlo code to determine maximum licensing (design basis) dose rates. In the third analysis step, one-dimensional shielding evaluations are added to extend the burnup and enrichment range from the design basis values for each of the primary fuel types.

Shielding Evaluations to Determine Bounding Fuel Type at Fixed (Design Basis) Burnup, Initial Enrichment and Cool Time

The design basis PWR and BWR fuel assemblies are determined by considering all assembly types intended for storage in the Universal Storage System, and identifying those assemblies expected to have the highest source terms based on initial loading of fuel and other operating factors. The design basis depletion characteristics for PWR assemblies in this evaluation step are an average burnup of 40,000 MWd/MTU, an initial enrichment of 3.7 wt % ^{235}U, and a 5-year cooling time. The design basis BWR depletion characteristics are an average burnup of 40,000 MWd/MTU, an initial enrichment of 3.25 wt % ^{235}U, and a 5-year cooling time. Detailed source descriptions of these selected assemblies are developed by using the SCALE SAS2H code [5]. The resulting source descriptions for each assembly type are employed in one-dimensional shielding calculations in order to identify bounding design basis assembly descriptions for both PWR and BWR assemblies on the basis of computed dose rates.

The determination of design basis fuel descriptions on the basis of one-dimensional shielding analyses is a unique approach that captures the combined effects of fuel self-shielding, spectral differences between assembly source terms, the relative contributions from gamma and neutron sources, and the influence of cask shielding materials and geometry. The design basis is selected as the result of computed dose rates rather than from a single gross assembly characteristic such as source rate or initial heavy metal loading.

As discussed in Section 5.2.5, the resulting design basis PWR fuel assembly for the shielding evaluation of the standard transfer cask and vertical concrete cask is the Westinghouse 17×17 standard assembly with an average burnup of 40,000 MWd/MTU, an initial enrichment of 3.7 wt % ^{235}U, and a 5-year cooling time, modified by increasing its hardware source. The shielding design basis BWR fuel is a GE 9×9 assembly with a burnup of 40,000 MWd/MTU, an initial enrichment of 3.25 wt % ^{235}U, and a 5-year cooling time, modified by increasing its hardware source. The source term specification is provided in Section 5.2.

Maximum Licensing Dose Rates

Three-dimensional analyses of the hardware source modified Westinghouse 17×17 and GE 9×9 design basis assemblies are then conducted to establish licensing basis dose rates. The three-dimensional dose rates are calculated for the design basis 40,000 GWd/MTU burned, 5-year cooled source used in the one-dimensional fuel comparisons. Section 5.1.3 contains the resulting maximum dose rate discussion. Detailed discussions and dose rate profiles for the transfer cask and the vertical concrete cask are presented in Section 5.4. Maximum dose rates obtained from the three-dimensional analyses are generally higher than those obtained from the one-dimensional analysis, as explicit disk models versus one-dimensional homogenous smearing of the disks captures local mass details and resulting radiation shield performance. Three-dimensional evaluations are also capable of capturing dose peaks associated with radiation streaming paths, such as the air inlets and outlets of the vertical concrete cask.

Shielding and Source Term Extension to the Range of Assemblies, Burnup and Initial Enrichments

One-dimensional shielding evaluations in conjunction with heat load limits are employed in setting minimum cool times for the range of fuel types, including design basis assemblies, having different burnups and initial enrichments than the initial design basis burnup of 40 GWd/MTU

with 5-year cooling and initial enrichments of 3.7 wt % ^{235}U PWR or 3.25 wt % ^{235}U BWR. Dose rate limits are set by determining one-dimensional dose rates for the 40 GWd/MTU, 5-year-cooled design basis Westinghouse 17×17 and GE 9×9 fuel assembly designs. The calculated dose rates at this depletion point are the design basis values, not to be exceeded by any other fuel type, burnup/initial enrichment/cool time combination. Not exceeding the one-dimensional limits provides assurance that the three-dimensional dose rates documented in Section 5.1.3 are not exceeded. Details on the minimum cool time evaluations are provided in Sections 5.4 and 5.5.

Shielding evaluations are performed for the transfer cask with both wet and dry canister cavities. The wet canister cavity condition occurs during the welding of the shield lid. During the welding of the structural lid, the canister cavity is assumed to be completely dry. Note that in the wet canister condition, the modeled water level is the base of the upper end fitting. Shielding evaluations for the concrete cask assume a dry cavity.

Dose rate profiles for the transfer cask and the vertical concrete cask are presented in Section 5.4.

Site-specific fuels, which may have configurations or parameters that are not considered in the design basis fuels, are described in Section 5.6. As described in Section 5.6, the site-specific fuels must either be shown to be bounded by the evaluation of the design basis fuel, or be separately evaluated to establish limits which are maintained by administrative controls.

5.1.1 Fuel Assembly Classification

5.1.1.1 PWR Fuel Assembly Classes

As discussed in Chapters 1.0 and 6.0 of this report, the PWR fuel assemblies to be stored in the vertical concrete cask are divided into three classes on the basis of similarity of their lengths. Of the PWR assemblies to be stored, the following four are selected for further analysis on the basis of their computed radiation source terms:

PWR Assembly Type	Class
Westinghouse 15×15 Std	Class 1
Westinghouse 17×17 Std	Class 1
Babcock & Wilcox 15×15 Mark B	Class 2
Combustion Engineering 16×16 System 80	Class 3

These assembly types represent candidate design basis assemblies. The design basis assembly is chosen by performing one-dimensional shielding calculations for each assembly type. The results of the one-dimensional analysis are used to identify the single limiting assembly type which is then used in subsequent detailed three-dimensional shielding calculations in order to determine bounding dose rates for the PWR case. Using this approach, the limiting assembly type is determined on the basis of actual computed dose rates, including factors such as fuel self-shielding and spectral effects, which would otherwise be ignored if the design basis were selected on the basis of source rates alone.

The candidate PWR fuel assemblies are analyzed on the basis of an assumed initial enrichment of 3.7 wt % ^{235}U, a burnup of 40,000 MWd/MTU, and a cooling time of 5 years. The initial enrichment assumed in the shielding analysis is significantly less than the criticality analysis design basis value of 4.2 wt % ^{235}U, so that the calculated neutron source rate bounds that of higher enrichment fuel, which may reach the design basis burnup of 40,000 MWd/MTU. This assumption produces a neutron source that is 30% higher than that calculated assuming a 4.2 wt % ^{235}U initial enrichment.

In addition, the source terms for each assembly type include bounding fuel and nonfuel hardware source terms associated with certain control components, including burnable poison clusters and power shaping elements specific to each fuel type. The source specifications for the design basis fuel are discussed in Section 5.2.

5.1.1.2 BWR Fuel Assembly Classes

On the basis of similarity of length, the BWR fuel assemblies to be stored in the vertical concrete cask are divided into two classes (Class 4 corresponds to BWR/2–3 assemblies and Class 5 corresponds to BWR/4–6 assemblies). In a manner similar to that employed in the PWR case, the following BWR assemblies are chosen as candidate design basis assemblies for the shielding analysis on the basis of their computed radiation source terms:

BWR Assembly Type	Class
GE 7×7 BWR/2–3 version GE-2b	Class 4
GE 8×8-2 BWR/2–3 version GE-5	Class 4
GE 8×8-4 BWR/2–3 version GE-8	Class 4
GE 7×7 BWR/4–6 version GE-2	Class 5
GE 8×8-2 BWR/4–6 version GE-5	Class 5
GE 8×8-4 BWR/4–6 version GE-10	Class 5
GE 9×9-2 BWR/4–6 version GE-11	Class 5

One-dimensional shielding calculations are performed for each assembly in order to identify a single assembly type as the design basis assembly for subsequent detailed three-dimensional shielding analysis. The candidate BWR fuel assemblies are analyzed on the basis of an initial enrichment of 3.25 wt % ^{235}U, a burnup of 40,000 MWd/MTU, and a cooling time of 5 years. The initial enrichment assumed in the shielding analysis is significantly less than the criticality analysis design basis value of 4.0 wt % ^{235}U, so that the calculated neutron source rate bounds that of higher enrichment fuel, which may reach the design basis burnup of 40,000 MWd/MTU. This assumption produces a neutron source that is 20% higher than that calculated assuming a 4.0 wt % ^{235}U initial enrichment.

5.1.2 Codes Employed

The SCALE 4.3PC [4] code system is used in the analysis of the vertical concrete cask and the transfer cask, with the MCBEND [23] code used to calculate dose rates at the concrete cask air inlets and outlets. Source terms are generated by using the SAS2H [5] sequence as described in Section 5.2. One-dimensional radial and axial SAS1 [6] analyses are performed in order to identify design basis PWR and BWR fuel types. With these design basis source descriptions, detailed three-dimensional analyses are performed by using the SAS4 [3] Monte Carlo shielding analysis sequence and the MCBEND Monte Carlo code. Modifications to SAS4 permit computation of dose rate profiles along surface detectors. These changes are further described in Section 5.4.1.

The 27-group neutron, 18 group gamma, coupled cross-section library (27N-18COUPLE) [7] derived from ENDF/B-IV data is used in the concrete cask and standard transfer cask shielding evaluations. The MCBEND shielding evaluations use the 28-group and 22-group gamma energy structures embedded in the code. Source terms include fuel neutron, fuel gamma, and gamma contributions from activated hardware. The effects of subcritical neutron multiplication and secondary gamma production due to neutron capture are included in the analysis. Dose rate evaluations include the effect of axial fuel burnup variation on fuel neutron and gamma source terms as described in Section 5.2.6.

5.1.3 Results of Analysis

This section summarizes the results of the three-dimensional shielding analysis. Reported values are rounded up to the indicated level of precision. Due to the statistical nature of Monte Carlo

analysis, all dose rate results are shown with the relative standard deviation in the result, expressed as a percentage.

5.1.3.1 Dose Rates for Vertical Concrete Cask

Cask Containing PWR Fuel

A summary of the maximum calculated dose rates for the concrete cask under normal and accident conditions is shown in Table 5.1-1 for the design basis PWR fuel. These dose rates are based on three-dimensional Monte Carlo analysis. Uncertainty in Monte Carlo results is indicated in parentheses. Under normal conditions with design basis fuel and the Transportable Storage Cask centered in the Vertical Concrete Cask, the concrete cask maximum side wall surface dose rate is 49 (<1%) mrem/hr at the fuel midplane and 56 (6%) mrem/hr on the top surface at locations on the cask top directly above the outlet vents. Since the concrete cask is vertical during normal storage operation, the cask bottom is inaccessible. The maximum surface dose rate at the lower air inlet openings is 136 (1%) mrem/hr with supplemental shielding and 694 (<1%) mrem/hr without supplemental shielding. The maximum surface dose rate at the air outlet openings is 63 (1%) mrem/hr. The average maximum inlet plus outlet dose rate is 99.5 mrem/hr with supplemental shielding.

The overall cask side average surface dose rate is 38 (<1%) mrem/hr for the PWR design basis fuel. On the cask top, the PWR average surface dose rate is 27 (2%) mrem/hr.

The postulated accident condition involves a projectile impact resulting in localized loss of 6 inches of concrete. The accident is analyzed assuming that the outermost 3 inches of concrete is lost from the entire outer surface of the cask. In this case, the surface average dose rate increases to 89 (<1%) mrem/hr with design basis PWR fuel. The maximum dose rate, assuming a 3-inch concrete loss over the entire radial surface of the cask, is 143 (3%) mrem/hr. At the postulated missile impact area, the estimated localized dose rate is less than 250 mrem/hr. There are no design basis accidents that result in a tip-over of the concrete cask.

Cask Containing BWR Fuel

Table 5.1-2 provides the maximum calculated dose rates for the concrete cask under normal and accident conditions for the design basis BWR fuel. As in the PWR case, these dose rates are based on three-dimensional Monte Carlo analysis. Uncertainty in Monte Carlo results is

indicated in parentheses. Under normal conditions with design basis BWR fuel, the concrete cask maximum side surface dose rate is 31 (1%) mrem/hr at the fuel midplane and 43 (5%) mrem/hr on the top surface at locations directly above the air outlet structures. The dose rate at the air inlet opening is 129 (1%) mrem/hr with supplemental shielding and 645 (<1%) mrem/hr without supplemental shielding. The maximum surface dose rate at the air outlet openings is 55 (1%) mrem/hr.

Under accident conditions involving a projectile impact and an assumed 3 inches of concrete removed from the entire radial surface of the cask, the surface dose rate maximum increases to 85 (4%) mrem/hr with design basis BWR fuel. The radial surface average dose rate increases to 54 (<1%) mrem/hr and the cask surface dose rate for the localized loss of 6 inches of concrete is estimated to be less than 250 mrem/hr.

The overall cask side average surface dose rates are 23 (<1%) mrem/hr for the BWR design basis fuel. On the cask top, the BWR average surface dose rate is 20 (1%) mrem/hr.

5.1.3.2 Dose Rates for Transfer Cask

Transfer Cask Containing PWR Fuel

Maximum dose rates for the standard or advanced transfer cask with a wet and dry canister cavity are shown in Table 5.1-3 for design basis PWR fuel. Under wet canister conditions, the maximum surface dose rates with design basis PWR fuel are 259 (<1%) mrem/hr on the cask side and 579 (<1%) mrem/hr on the cask bottom. The cask side average surface dose rate under wet conditions is 137 (<1%) mrem/hr, and the bottom average surface dose rate is 258 (<1%) mrem/hr. Under dry conditions, the maximum surface dose rates are 410 (<1%) mrem/hr on the cask side and 819 (<1%) mrem/hr on the cask bottom. Cask average surface dose rates are 306 (<1%) mrem/hr on the side and 374 (<1%) mrem/hr on the bottom. In normal operation, the bottom of the transfer cask is inaccessible during welding of the canister lids.

During the lid welding operation, localized maximum surface dose rates occur at the canister periphery. Under wet canister conditions with a 5-inch temporary shield in place atop the shield lid, the maximum contact dose rate is 2,092 (4%) mrem/hr. This dose rate is highly localized to the narrow gap between the temporary shielding and the cask inner wall. At 1 meter above the top of the cask, the maximum dose rate is 320 (6%) mrem/hr. The surface average dose rate at the cask top surface is 579 (3%) mrem/hr under these conditions.

Under dry conditions with the shield lid and structural lid in place, and with no additional temporary shielding, the maximum surface dose rate is 715 (<1%) mrem/hr. The cask top average surface dose rate is 369 (2%) mrem/hr under these conditions.

Transfer Cask Containing BWR Fuel

Maximum dose rates for the standard or advanced transfer cask with a wet and dry canister cavity are shown in Table 5.1-4 for design basis BWR fuel. Under wet canister conditions, the maximum surface dose rates with design basis BWR fuel are 189 (<1%) mrem/hr on the cask side and 539 (<1%) mrem/hr on the cask bottom. The cask side average surface dose rate under wet conditions is 79 (<1%) mrem/hr, and the bottom average surface dose rate is 254 (<1%) mrem/hr. Under dry conditions, the maximum surface dose rates are 325 (<1%) mrem/hr on the cask side and 786 (<1%) mrem/hr on the cask bottom. Cask average surface dose rates are 228 (<1%) mrem/hr on the side and 379 (<1%) mrem/hr on the bottom. In normal operation, the bottom of the transfer cask is inaccessible during welding of the canister lids.

During the lid welding operation, localized maximum dose rates occur at the canister periphery. Under wet canister conditions with a 5 inch temporary shield in place atop the shield lid, the maximum surface dose rate is 1803 (4%) mrem/hr. This dose rate is highly localized to the narrow gap between the temporary shielding and the cask inner wall. At 1 meter above the top of the cask, the maximum dose rate is 314 (7%) mrem/hr. The surface average dose rate at the cask top surface is 466 (3%) mrem/hr under these conditions.

Under dry conditions with the shield lid and structural lid in place, and no additional temporary shielding, the maximum surface dose rate is 396 (<1%) mrem/hr. The cask top average surface dose rate is 222 (3%) mrem/hr under these conditions.

Table 5.1-1 Summary of Maximum Dose Rates: Vertical Concrete Cask with PWR Fuel

Condition	Source	Cask Surface (mrem/hr with relative uncertainty)				1 Meter From Surface (mrem/hr with relative uncertainty)			
		Side		Top		Side		Top	
Normal	Neutron	0.1	1%	0.3	14%	<0.1	<1%	5.3	1%
	Gamma	48.6	<1%	55.1	6%	25.2	<1%	8.0	7%
	Total	49.[2]	<1%	56.	6%	26.	<1%	14.	5%
Design Basis Accident	Neutron	0.3	10%	N/A[1]		0.1	2%	N/A[1]	
	Gamma	141.9	3%	N/A[1]		62.5	<1%	N/A[1]	
	Total	143.[3]	3%	N/A[1]		63.	<1%	N/A[1]	

1. No design basis accident impacts top dose rates.
2. At the fuel midplane. Without supplemental shielding, the air inlet dose rate is 694 (<1%) mrem/hr.
3. At the missile impact area.

Table 5.1-2 Summary of Maximum Dose Rates: Vertical Concrete Cask with BWR Fuel

Condition	Source	Cask Surface (mrem/hr with relative uncertainty)				1 Meter From Surface (mrem/hr with relative uncertainty)			
		Side		Top		Side		Top	
Normal	Neutron	0.2	<1%	0.2	19%	<0.1	<1%	3.2	2%
	Gamma	30.6	1%	42.2	5%	15.3	<1%	5.3	4%
	Total	31.[2]	1%	43.	5%	16.	<1%	9.	2%
Design Basis Accident	Neutron	0.5	8%	N/A[1]		0.2	1%	N/A[1]	
	Gamma	83.8	4%	N/A[1]		38.3	1%	N/A[1]	
	Total	85.[3]	4%	N/A[1]		39.	1%	N/A[1]	

1. No design basis accident impacts top dose rates.
2. At the fuel midplane. Without supplemental shielding, the air inlet dose rate is 645 (<1%) mrem/hr.
3. At the missile impact area.

UPDATED

Table 5.1-3 Summary of Maximum Dose Rates: Standard or Advanced Transfer Cask with PWR Fuel

		Cask Surface (mrem/hr with relative uncertainty)						1 Meter From Surface (mrem/hr with relative uncertainty)					
Condition	**Source**	**Side**		**Top**		**Bottom**		**Side**		**Top**		**Bottom**	
Normal Wet[1]	Neutron	0.1	8%	0.2	3%	0.3	2%	1.3	<1%	<0.1	2%	0.1	2%
	Gamma	258.7	<1%	2091.	4%	578.2	<1%	65.3	<1%	319.8	6%	266.4	<1%
	Total	259.	<1%	2092.	4%	579.	<1%	67.	<1%	320.	6%	267.	<1%
Normal Dry[2]	Neutron	12.6	2%	111.5	<1%	37.8	<1%	29.5	<1%	28.7	1%	10.0	<1%
	Gamma	397.2	<1%	603.4	<1%	781.1	<1%	126.5	<1%	278.3	<1%	365.5	<1%
	Total	410.	<1%	715.	<1%	819.	<1%	156.	<1%	307.	<1%	376.	<1%

[1] 5 inches of carbon steel temporary shielding, shield lid in position.
[2] Shield lid and structural lid in position, no additional temporary shielding.

Table 5.1-4 Summary of Maximum Dose Rates: Standard or Advanced Transfer Cask with BWR Fuel

		Cask Surface (mrem/hr with relative uncertainty)						1 Meter From Surface (mrem/hr with relative uncertainty)					
Condition	**Source**	**Side**		**Top**		**Bottom**		**Side**		**Top**		**Bottom**	
Normal Wet[1]	Neutron	<0.1	31%	<0.1	17%	<0.1	7%	2.3	<1%	<0.1	12%	<0.1	6%
	Gamma	188.2	<1%	1803.	4%	538.1	<1%	34.8	<1%	313.2	7%	258.5	<1%
	Total	189.	<1%	1803.	4%	539.	<1%	38.	<1%	314.	7%	259.	<1%
Normal Dry[2]	Neutron	152.3	<1%	62.1	1%	34.7	1%	53.5	<1%	16.3	2%	9.3	1%
	Gamma	171.8	1%	333.6	<1%	750.7	<1%	67.3	<1%	156.4	1%	360.3	<1%
	Total	325.	<1%	396.	<1%	786.	<1%	121.	<1%	173.	1%	370.	<1%

[1] 5 inches of carbon steel temporary shielding, shield lid in position.
[2] Shield lid and structural lid in position, no additional temporary shielding.

UPDATED

In Figure 5.4-5, the radial dose rate profile at the top surface of the cask is shown. Two peaks occur in the radial profile. Above the canister/weldment annulus, a peak is formed from approximately equal contributions of end-fitting and plenum gamma and fuel neutron. At radial locations above the upper vents, another peak is observed due primarily to end-fitting gammas.

Three-Dimensional Dose Rates for Concrete Cask Containing BWR Fuel

Figures 5.4-6 through 5.4-10 present the three-dimensional model dose rates for the concrete cask containing BWR fuel. Figure 5.4-6 shows the axial dose rate profile along the cask surface broken down by contributing radiation type. Dose rates along the cask axial surface are dominated by gamma contributions due to the relatively high neutron shielding effectiveness of the concrete. Figure 5.4-7 shows the total dose rate profile at various radial distances from the cask surface.

In the axial profile plots, each datum represents the circumferentially averaged dose rate at the corresponding elevation. Negative elevations indicate axial locations below the fuel axial midplane, and correspond to results obtained from the three-dimensional bottom half model. In the vertical dose profile, peaking is observed at the upper and lower end fitting locations as well as at the locations of the lower intake and upper outlet vents.

At locations away from the air inlets and outlets, the maximum axial dose rates occurs at the fuel midplane, where a peak dose rate of 31 (1%) mrem/hr is computed. At the air outlets, an azimuthal maximum of 55 mrem/hr (1%) is computed. Figure 5.4-8 illustrates the azimuthal variation of total dose rate at the air outlet elevation. Dose rates at the air inlets are considerably higher than at the air outlets. The dose rate at the air inlet opening is 129 (1%) mrem/hr with supplemental shielding and 645 (<1%) mrem/hr without supplemental shielding. The azimuthal variation of dose rate at the air inlet is shown in Figure 5.4-9.

In Figure 5.4-10, the radial dose rate profile at the top surface of the cask is shown. Two peaks occur in the radial profile. Above the canister/weldment annulus, a peak is formed from approximately equal contributions of end-fitting and plenum gamma and fuel neutron. At radial locations above the upper vents, another peak is observed due primarily to end-fitting gammas.

5.4.3.2 Standard Transfer Cask Dose Rates

One-Dimensional Dose Rates

One-dimensional radial dose rates for the standard transfer cask with design basis PWR or BWR fuel are in good agreement with the corresponding three-dimensional models at the radial midplane. As with the concrete cask one-dimensional radial model, the peaks in the radial dose rates due to activated end fittings cannot be captured by one-dimensional analysis. One-dimensional analysis supports the results of the more sophisticated three-dimensional models.

Three-Dimensional Dose Rates for the Standard Transfer Cask Containing PWR Fuel

The three-dimensional model dose rates for the standard transfer cask containing PWR fuel are presented in Figures 5.4-11 through 5.4-19. For the top and bottom axial cases, the SAS4 surface detectors are subdivided in a manner which gives the centermost subdetector a relatively large radius. This detector partitioning more closely balances subdetector areas and avoids poor Monte Carlo statistics on the central subdetector.

The transfer cask side dose rate profiles with a dry cavity are shown in Figure 5.4-11 for the constituent source components and in Figure 5.4-13 at various distances from the cask surface. In this condition, the majority of the dose rate is from fuel neutron and gamma source, but significant peaks are shown from the activated end fittings. In this condition, the peak dose rate on the side of the transfer cask is 410 (<1%) mrem/hr.

The transfer cask side dose rate profiles with a wet canister are shown in Figure 5.4-12 for the constituent source components and in Figure 5.4-14 at various distances from the cask surface. In the wet case, the majority of the dose rate is from fuel gamma sources and activated nonfuel hardware gamma. Note that in the wet condition, it is assumed in the model that the water level in the canister is lowered to the base of the upper end-fitting in order to facilitate the lid welding operations. Thus, the top end fitting is uncovered and causes a peak in dose rate at the top of the transfer cask due to the gamma source from the activated top end fitting. In this condition, the peak dose rate on the side of the transfer cask is 259 (<1%) mrem/hr.

When configured for the shield lid welding operation, the standard transfer cask, with wet canister and temporary shielding in place, has a peak surface dose rate of 2,092 (4%) in the narrow gap between the temporary shield and the cask inner shell. This dose rate is highly

Sample response functions for the PWR and BWR storage casks and the standard transfer cask are listed in Table 5.5-3 and Table 5.5-4 for neutron and gamma sources, respectively. Only seven energy groups are presented for the fuel neutron source since the complete SAS2H neutron source is located in these energy groups.

With the dose rate response method a convenient and simple method for determining storage and transfer cask surface dose rates is available.

5.5.4 Minimum Allowable Cooling Time Determination

The following strategy is used to determine limiting cooling times for each combination of fuel type, initial enrichment, and burnup:

a) Determine decay heat and dose rate values at each cooling time step.

b) Interpolate in the resulting collection of data to find minimum cooling time required to meet each limiting value, decay heat and transfer and storage cask dose rate, individually.

c) Select the maximum of this collection of minimum required cooling times, rounded up to the next whole year, as the minimum required cooling time for this combination of burnup, enrichment and cooling time.

5.5.4.1 PWR and BWR Assembly Minimum Cooling Times

Minimum allowable cooling times are established for each of the fuel type, burnup, and enrichment combinations based on the cask decay heat limit of 23 kW. Listed in Table 5.5-2 is a comparison of one-dimensional limits to the corresponding three-dimensional dose rates, demonstrating that the dose rate limits applied in the one-dimensional analysis comply with the three-dimensional analysis results in Sections 5.1 and 5.4. A sample of the calculated cooling times required to reach each of the limits for Westinghouse 17×17 and GE 9×9 fuel assemblies at 40 GWd/MTU are shown in Tables 5.5-5 and 5.5-6, respectively. The identical calculation sequence is repeated for all the assembly types and burnups indicated in Section 5.5-1. The limiting cooling times are then collapsed to array size specific limiting values as listed in Table 5.5-7 and Table 5.5-8.

Table 5.5-1 Limiting PWR and BWR Fuel Types Based on Uranium Loading

Reactor	Array	Fuel Assembly
PWR	17×17	WE 17×17 Standard
PWR	16×16	CE 16×16 System 80
PWR	15×15	BW 15×15
PWR	14×14	WE 14×14
BWR	9×9	GE 9×9-79 Fuel Rods (GE 9×9-2L)
BWR	8×8	GE 8×8-63 Fuel Rods
BWR	7×7	GE 7×7

Table 5.5-2 Design Basis Assembly Dose Rate Limit (mrem/hr)

Configuration (Radial Dose Rates)	Neutron	Gamma	Hardware Gamma	1-D Total (mrem/hr)	3-D Total (mrem/hr)
PWR Storage	0.6	22.5	11.1	34.2	49
BWR Storage	0.9	16.5	0.1	17.6	31
PWR Transfer (dry)	68.1	127.4	82.4	277.8	~375
BWR Transfer (dry)	108.0	92.6	1.1	201.6	~320