HOLTEC INTERNATIONAL

Krishna P. Singh Technology Campus, 1 Holtec Blvd., Camden, NJ 08104
Telephone (856) 797-0900
Fax (856) 797-0909

April 12, 2019

Yen Chen, Project Manager – Licensing Branch
Division of Spent Fuel Management
Office of Nuclear Material Safety and Safeguards

ATTN: Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Docket No. 72-1032, Certificate of Compliance (CoC) No. 1032
CAC No. 001028, EPID No. L-2017-LLA-0030

Reference:
1. "Holtec International HI-STORM Flood/Wind Multipurpose Canister Storage System Amendment Request 1032-4" (Letter No. 5018043 from Kimberly Manzione (Holtec) to Mark Lombard (NRC) dated March 11, 2016)
2. "Amendment No. 4 to Certificate of Compliance No. 1032 for the HI-STORM Flood/Wind Multipurpose Canister Storage System – Second Request for Additional Information" (Letter from Yen-Ju Chen (NRC) to K. Manzione (Holtec), dated January 29, 2019)

Subject: HI-STORM FW Amendment 4 Response to Second Request for Additional Information

Dear Ms. Chen:

By letter dated January 29, 2019 [2], NRC staff documented their second request for additional information (RAI) that is required to complete their detailed technical review of HI-STORM FW CoC 1032 Amendment Request 4 [1] submitted March 11, 2016.

Holtec's response to NRC staff RAI and supporting information are in the attachments to this letter. Attachment 1 contains Holtec's response to RAI 5-5. Attachments 2 and 3 contain the non-proprietary and proprietary versions, respectively, of proposed changes to the HI-STORM FW FSAR (Proposed Rev. 5F). HI-STORM FW FSAR proposed changes specific to the second request for additional information are highlighted. Attachments 4 and 5 contain supporting calculation packages and data files. Attachments 3 through 5 contain Holtec proprietary

NMSS26

HOLTEC INTERNATIONAL

Krishna P. Singh Technology Campus, 1 Holtec Blvd., Camden, NJ 08104
Telephone (856) 797-0900
Fax (856) 797-0909

information. Attachment 6 contains an affidavit in accordance with 10 CFR 2.390 requesting that information in Attachments 3 through 5 be withheld from public disclosure.

Please contact me at (856)797-0900 extension 3844 if you have any questions or require any additional information.

Sincerely,

Royston Ngwayah
Licensing Engineer,
Holtec International

cc: (letter only without attachments)
Mike Layton (NRC)
John McKirgan (NRC)

Attachments:
Attachment 1: HI-STORM FW CoC Amendment 4 Second Round RAI Response (non-proprietary)
Attachment 2: HI-STORM FW FSAR (Report HI-2114830 Proposed Rev. 5.F), changed pages (non-proprietary)
Attachment 3: HI-STORM FW FSAR (Report HI-2114830 Proposed Rev. 5.F), changed pages (Holtec proprietary)
Attachment 4: Lower Bound Fuel Enrichment Based on Industry Data, HI-2188480 (Rev. 0) and computer data files (Holtec Proprietary)
Attachment 5: HI-STORM FW and HI-TRAC VW Shielding Analysis, HI-2094431 (Rev. 15) and computer data files (Holtec Proprietary)
Attachment 6: Affidavit pursuant to 10 CFR 2.390

AFFIDAVIT PURSUANT TO 10 CFR 2.390

I, Kimberly Manzione, being duly sworn, depose and state as follows:

(1) I have reviewed the information described in paragraph (2) which is sought to be withheld, and am authorized to apply for its withholding.

(2) The information sought to be withheld are Attachments 3 through 5 to Holtec Letter 5018066, which contain Holtec Proprietary information.

(3) In making this application for withholding of proprietary information of which it is the owner, Holtec International relies upon the exemption from disclosure set forth in the Freedom of Information Act ("FOIA"), 5 USC Sec. 552(b)(4) and the Trade Secrets Act, 18 USC Sec. 1905, and NRC regulations 10CFR Part 9.17(a)(4), 2.390(a)(4), and 2.390(b)(1) for "trade secrets and commercial or financial information obtained from a person and privileged or confidential" (Exemption 4). The material for which exemption from disclosure is here sought is all "confidential commercial information", and some portions also qualify under the narrower definition of "trade secret", within the meanings assigned to those terms for purposes of FOIA Exemption 4 in, respectively, Critical Mass Energy Project v. Nuclear Regulatory Commission, 975F2d871 (DC Cir. 1992), and Public Citizen Health Research Group v. FDA, 704F2d1280 (DC Cir. 1983).

AFFIDAVIT PURSUANT TO 10 CFR 2.390

(4) Some examples of categories of information which fit into the definition of proprietary information are:

a. Information that discloses a process, method, or apparatus, including supporting data and analyses, where prevention of its use by Holtec's competitors without license from Holtec International constitutes a competitive economic advantage over other companies;

b. Information which, if used by a competitor, would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing of a similar product.

c. Information which reveals cost or price information, production, capacities, budget levels, or commercial strategies of Holtec International, its customers, or its suppliers;

d. Information which reveals aspects of past, present, or future Holtec International customer-funded development plans and programs of potential commercial value to Holtec International;

e. Information which discloses patentable subject matter for which it may be desirable to obtain patent protection.

The information sought to be withheld is considered to be proprietary for the reasons set forth in paragraphs 4.a and 4.b above.

(5) The information sought to be withheld is being submitted to the NRC in confidence. The information (including that compiled from many sources) is of a sort customarily held in confidence by Holtec International, and is in fact so held. The information sought to be withheld has, to the best of my knowledge and belief, consistently been held in confidence by Holtec International. No public disclosure has been made, and it is not available in public sources. All disclosures to third parties, including any required transmittals to the NRC, have been made, or must be made, pursuant to

regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence. Its initial designation as proprietary information, and the subsequent steps taken to prevent its unauthorized disclosure, are as set forth in paragraphs (6) and (7) following.

(6) Initial approval of proprietary treatment of a document is made by the manager of the originating component, the person most likely to be acquainted with the value and sensitivity of the information in relation to industry knowledge. Access to such documents within Holtec International is limited on a "need to know" basis.

(7) The procedure for approval of external release of such a document typically requires review by the staff manager, project manager, principal scientist or other equivalent authority, by the manager of the cognizant marketing function (or his designee), and by the Legal Operation, for technical content, competitive effect, and determination of the accuracy of the proprietary designation. Disclosures outside Holtec International are limited to regulatory bodies, customers, and potential customers, and their agents, suppliers, and licensees, and others with a legitimate need for the information, and then only in accordance with appropriate regulatory provisions or proprietary agreements.

(8) The information classified as proprietary was developed and compiled by Holtec International at a significant cost to Holtec International. This information is classified as proprietary because it contains detailed descriptions of analytical approaches and methodologies not available elsewhere. This information would provide other parties, including competitors, with information from Holtec International's technical database and the results of evaluations performed by Holtec International. A substantial effort has been expended by Holtec International to develop this information. Release of this information would improve a competitor's position because it would enable Holtec's competitor to copy our technology and offer it for sale in competition with our company, causing us financial injury.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

(9) Public disclosure of the information sought to be withheld is likely to cause substantial harm to Holtec International's competitive position and foreclose or reduce the availability of profit-making opportunities. The information is part of Holtec International's comprehensive spent fuel storage technology base, and its commercial value extends beyond the original development cost. The value of the technology base goes beyond the extensive physical database and analytical methodology, and includes development of the expertise to determine and apply the appropriate evaluation process.

The research, development, engineering, and analytical costs comprise a substantial investment of time and money by Holtec International.

The precise value of the expertise to devise an evaluation process and apply the correct analytical methodology is difficult to quantify, but it clearly is substantial.

Holtec International's competitive advantage will be lost if its competitors are able to use the results of the Holtec International experience to normalize or verify their own process or if they are able to claim an equivalent understanding by demonstrating that they can arrive at the same or similar conclusions.

The value of this information to Holtec International would be lost if the information were disclosed to the public. Making such information available to competitors without their having been required to undertake a similar expenditure of resources would unfairly provide competitors with a windfall, and deprive Holtec International of the opportunity to exercise its competitive advantage to seek an adequate return on its large investment in developing these very valuable analytical tools.

AFFIDAVIT PURSUANT TO 10 CFR 2.390

STATE OF NEW JERSEY)
) ss:
COUNTY OF CAMDEN)

Kimberly Manzione, being duly sworn, deposes and says:

That she has read the foregoing affidavit and the matters stated therein are true and correct to the best of her knowledge, information, and belief.

Executed at Camden, New Jersey, this 12th day of April, 2019.

Kimberly Manzione
Licensing Manager
Holtec International

Subscribed and sworn before me this 12 day of April, 2019.

Korin M. Fagan

KORIN M FAGAN
Notary Public, State of New Jersey
My Commission Expires
February 15, 2024

NRC RAI 5-5

Verify whether the combinations of burn up, initial enrichment, and cooling time (BECT) presented in Table 5.4.9 of the Safety Analysis Report (SAR) represent the bounding source terms for the new 16x16D fuel and associated fuel loading patterns in the HISTORM FW system to be authorized under Amendment 4. Alternatively, provide a complete list of the BECT combinations for the 16x16D fuel so that the staff can determine if the radiological source terms result in radiological dose rates that are within the HISTORM FW system design limits.

In the letter dated January 19, 2018 (ML 18022A612), the staff requested, in RAI 5-4, the applicant to clarify if the radiation source term used in the shielding evaluation is bounding. The applicant stated in its response that: "Shielding analysis ... [were] performed considering a wide range of conservative combinations of burnups, enrichments and cooling times. These combinations sufficiently bound all possible fuel loadings." The applicant revised the SAR to include combinations of BECT in Table 5.4.9. The applicant stated that these BECT combinations used in calculations result in source terms that are reasonably bounding for all realistically expected assemblies. The staff notes that these BECT combinations do not mirror the BECT limits in Table 2.1-1.III of the Technical Specifications (TS) Appendix B. The applicant provided the decay heat limits for each fuel cell in the fuel basket that is to be loaded with the new 16x16D fuel in TS Appendix B Table 2.3-5 and states, in the SAR, that the decay heat is used to bound the design basis source terms to demonstrate compliance with 10 CFR 72.236(d). However, there is no information provided on the relationship between the decay heat and the radiological source terms. There are BECT combinations for which an assembly could produce the same decay heat but potentially have higher than design basis radiation source terms (neutron, gamma, or both).

It is important to note that the recommendations published in NUREG/CR-6716 are based on a balanced evaluation of parameters important to safety while alleviating limitations in the TS to provide the CoC holders flexibility to make design changes under the provisions in 10 CFR 72.48. The staff requests this information to determine if the HI-STORM FW Amendment No.4 system, with the requested new contents, meets the regulatory requirements of 10 CFR 72.236(d).

Holtec's Response to RAI 5-5:

Holtec is in the process of implementing the Fuel Qualification Tables (FQTs) approach as discussed with the NRC staff during recent conference calls and a meeting on March 19, 2019. The fuel permitted by the FQT is used to determine the radiation source terms for dose and dose rate evaluations. The FSAR Chapters 2 and 5 are revised to include this approach and the proposed changed pages are submitted to the NRC staff to complete our response to this RAI. In summary, the approach for determining the neutron and gamma source terms for the new decay heat load patterns in amendment 4 and supporting calculations are as follows:

The design basis of the fuel permitted to be loaded into the system, in terms of burnups and cooling times, is not a single burnup and cooling time combination, but an equation that allows the calculation of the minimum cooling time as a function of the assembly burnup. While the

equation has some technical background (its loosely related to heat loads), it is, for the purpose of the FSAR, an arbitrarily selected equation, and is validated through dose rate calculations showing the maximum (bounding) dose rates that would correspond to the design basis. For this validation, a sufficient number of burnup and cooling times are selected based on the equation, dose rate calculations are then performed for each combination, and the maximum dose rates are established. Note that the combination of burnup and cooling time that results in the highest dose rate could be different for different dose locations. In general, the number and the values of the combinations are selected so there is reasonable assurance that the condition at or close to the maximum is identified.

For the enrichments used in the calculations in the FSAR, conservatively low values (that covers about 99% of fuel inventory [1]) are used, based on a database of actual fuel assemblies at US reactor sites. The low enrichments are based on [2] and [3]. To determine the enrichments, the data is separated in burnup bins spanning 5 GWd/mtU (i.e. 15 to 20, 20 to 25, 25 to 30, etc.). Then for each bin, an enrichment value is determined that presents a lower bound value for most of the assemblies in that bin.

SCALE 6.2.1, along with the above methodology, is used to develop the source terms which are used to the update the shielding evaluation.

The HI-STORM FW Amendment 4 FSAR is revised to reflect this new approach.

Furthermore, Appendix S is added to [4]. In this new appendix, the effect of lower enrichment or shorter cooling time on dose rate of HI-STORM FW with MPC-32ML loaded with high burnup 16x16D fuel is evaluated. It is shown that the dose rate increase at most dose locations is up to 10%. At the top of the HI-STORM FW, the dose rate increase is less than 20%, but it is not expected anyone works at that location. The sensitivity study performed in this supplement for HBF shows the effect of lower enrichment or shorter cooling time on adjacent, 1 m and 100 m dose rates for a HI-STORM 100 cask with MPC-32ML is relatively small.

References

[1] Lower Bound Fuel Enrichment Based on Industry Data, HI-2188480 Revision 0, Holtec International.
[2] U.S. Energy Information Administration, Form GC-859, "Nuclear Fuel Data Survey" (2013).
[3] U.S. Energy Information Administration, Form GC-859, "Nuclear Fuel Data Survey" (2002).
[4] HI-STORM FW and HI-TRAC VW Shielding Analysis, HI-2094431, Revision 15, Holtec International.

reasonably conservative dose rates. The reference assemblies given in Table 1.0.4 are the predominant assemblies used in the industry.

The design basis dose rates can be met by a variety of burnup levels and cooling times. Table 2.1.1a provides the acceptable ranges of burnup, enrichment and cooling time for all of the authorized fuel assembly array/classes. Table 2.1.5 and Figures 2.1.3 and 2.1.4 provide the axial distribution for the radiological source terms for PWR and BWR fuel assemblies based on the axial burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM FW System.

Non-fuel hardware, as defined in the Glossary, has been evaluated and is also authorized for storage in the PWR MPCs as specified in Table 2.1.1a.

2.1.6.1 Radiological Parameters for Spent Fuel and Non-fuel Hardware in MPC-32ML

MPC-32ML is authorized to store 16x16D spent fuel with burnup - cooling time combinations as given in Table 2.1.9.

The burnup and cooling time for every fuel assembly loaded into the MPC-32ML must satisfy the following equation:

$$Ct = A \cdot Bu^3 + B \cdot Bu^2 + C \cdot Bu + D$$

where,

Ct = Minimum cooling time (years),
Bu = Assembly-average burnup (MWd/mtU),
A, B, C, D = Polynomial coefficients listed in Table 2.1.9

Non-fuel hardware, as defined in the Glossary, has been evaluated and is also authorized for storage in the PWR MPCs as specified in Table 2.1.1b.

2.1.7 Criticality Parameters for Design Basis SNF

Criticality control during loading of the MPC-37 is achieved through either meeting the soluble boron limits in Table 2.1.6 OR verifying that the assemblies meet the minimum burnup requirements in Table 2.1.7. Criticality control during loading of the MPC-32ML is achieved through meeting the soluble boron limits in Table 2.1.6.

For those spent fuel assemblies that need to meet the burnup requirements specified in Table 2.1.7, a burnup verification shall be performed in accordance with either Method A OR Method B described below.

TABLE 2.1.9
BURNUP AND COOLING TIME FUEL QUALIFICATION REQUIREMENTS
FOR MPC-32ML (NOTE 1)

A	**B**	**C**	**D**
6.7667E-14	-3.6726E-09	8.1319E-05	2.7951E+00

Notes:

1. The burnup and cooling time for every fuel loaded into the MPC-32ML must satisfy the following equation:

$$Ct = A \cdot Bu^3 + B \cdot Bu^2 + C \cdot Bu + D$$

where,

Ct = Minimum cooling time (years)

Bu = Assembly-average burnup (MWd/mtU),

A, B, C, D = Polynomial coefficients listed ~~in~~ above

5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM FW system are:

- Gamma radiation originating from the following sources:

    1. Decay of radioactive fission products
    2. Secondary photons from neutron capture in fissile and non-fissile nuclides
    3. Hardware activation products generated during core operations

- Neutron radiation originating from the following sources

    1. Spontaneous fission
    2. α,n reactions in fuel materials
    3. Secondary neutrons produced by fission from subcritical multiplication
    4. γ,n reactions (this source is negligible)

During loading, unloading, and transfer operations, shielding from gamma radiation is provided by the stainless steel structure and the basket of the MPC and the steel, lead, and water in the HI-TRAC transfer cask. For storage, the gamma shielding is provided by the MPC, and the steel and concrete ("Metcon" structure) of the overpack. Shielding from neutron radiation is provided by the concrete of the overpack during storage and by the water of the HI-TRAC transfer cask during loading, unloading, and transfer operations. It is worth noting that the models, used to evaluate the dose calculations in this chapter, are constructed with minimum concrete densities and minimum lead thicknesses.

The shielding analyses were performed with MCNP5 [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the MPC-37 and MPC-89 design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 5 system [5.1.2, 5.1.3]. The source terms for the MPC-32ML design basis fuel were calculated with the TRITON and ORIGAMI sequences from the SCALE 6.2.1 system [5.1.4]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are Westinghouse (W) 17x17 and the General Electric (GE) 10x10, for PWR (in MPC-37) and BWR fuel types, respectively. 16x16D is the design basis fuel assembly for PWR in MPC-32ML. Required site specific shielding evaluations will verify whether those assemblies and assembly parameters are appropriate for the site-specific analyses. Subsection 2.1 specifies the acceptable fuel characteristics, including the acceptable maximum burnup levels and minimum cooling times for storage of fuel in the HI-STORM FW MPCs.

The following presents a discussion that explains the rationale behind the burnup and cooling time combinations that are evaluated in this chapter for normal and accident conditions.

10CFR72 contains two sections that set down main dose rate requirements: §104 for normal and off-normal conditions, and §106 for accident conditions. The relationship of these requirements to the analyses in this Chapter 5, and the burnup and cooling times selected for the various analyses, are as follows:

- 10CFR72.104 specifies the dose limits from an ISFSI (and other operations) at a site boundary under normal and off-normal conditions. Compliance with §104 can therefore only be demonstrated on a site-specific basis, since it depends not only on the design of the cask system and the loaded fuel, but also on the ISFSI layout, the distance to the site boundary, and possibly other factors such as use of higher density concrete or the terrain around the ISFSI. The purpose of this chapter is therefore to present a general overview over the expected dose rates, next to the casks and at various distances, to aid the user in applying ALARA considerations and planning of the ISFSI. To that extent, it is sufficient to present reasonably conservative dose rate values, based on a reasonable conservative choice of burnups and cooling times of the assemblies.
- For the accident dose limit in 10CFR72.106 it is desirable to show compliance in this Chapter 5 on a generic basis, so that calculations on a site-by-site basis are not required.† To that extend, a burnup and cooling time calculation that maximizes the dose rate under accident conditions needs to be selected.

The HI-STORM FW System offers three-region loading configurations for MPC-37 and MPC-89 as shown in Table 1.2.3a and Table 1.2.4 in Chapter 1. The uniform loading configuration for MPC-32ML is shown in Table 1.2.3b.

- For the MPC-37, there are two heat load patterns, each with a three-region loading configuration – Loading Pattern A and Loading Pattern B. An important difference between Pattern A and Pattern B loading is the loading is the maximum allowed heat load of the cells on the periphery of the MPC-37. Pattern A contains the cells with the lowest decay heat on the periphery, while Pattern B contains the cells with the highest decay heat on the periphery. In Pattern A, fuel assemblies with higher heat loads are loaded in the inner region allowing the user to take advantage of self-shielding from the fuel assemblies with lower heat loads in the outer regions. However, for Pattern B, the fuel assemblies with the higher heat loads could be loaded in the outer region (Region 3). Based on this difference it is expected that Pattern B will have higher dose rates than Pattern A. Therefore, for dose calculations Pattern B is selected, as it is the more limiting of the two loading patterns. Furthermore, uniform loading of MPC-37 cells is assumed for dose calculations. The burnup and cooling time combination is selected as representative of the cells on the periphery. This is a conservative approach, as it assumes that all thirty seven cells have a decay heat per cell equal to or slightly exceeding the decay heat of the periphery cells.

† As it is discussed in Subsection 5.1.2, a site-specific shielding evaluation may be required for accident-condition of MPC-32ML.

- For the MPC-89, there is only one heat load pattern with a three-region loading configuration. Based on the configuration for the MPC-89, fuel assemblies with higher heat loads would be loaded in the inner region allowing the user to take advantage of self-shielding from fuel assemblies with lower heat loads in the outer regions (see Table 1.2.4). However, for simplification, the shielding analyses are performed for a single region, i.e. assuming all assemblies in the basket have the same burnup and cooling time. In the case of the MPC-89 the burnup and cooling time combination is selected as a representative average for the entire basket.

- For the MPC-32ML, there is only one heat load pattern with a uniform loading configuration. The design basis burnup, enrichment, cooling time combinations are listed in Table 5.0.3 and Table 5.0.2 for normal and accident conditions, respectively. These combinations bound the burnup-cooling time allowable combinations in Table 2.1.9. More information is provided in Subsection 5.2.7.~~has the same burnup and enrichment as the design basis fuel combination for MPC-37, but with slightly longer cooling time. The fuel combination heat load is more than the decay heat limit per cell. Furthermore, since different burnup, enrichment and cooling time combinations may produce same decay heat, but different source terms, additional regionalized shielding analysis is performed by dividing the basket cells into Regions 1 to 3, where Region 1 is the innermost cells, and Region 3 is the outermost cells. The detail of this additional analysis is provided in Subsection 5.4.6.~~

While Loading Pattern B for the MPC-37 allows assemblies with higher heat loads and therefore higher source terms in the outer region (Region 3) of the MPC, the guiding principle in selecting fuel loading should still be to preferentially place assemblies with higher source terms in the inner regions of the basket as far as reasonably possible.

It is recognized that for a given heat load, an infinite number of burnup and cooling time combination could be selected, which would result in slightly different dose rate distributions around the cask. For a high burnup with a corresponding longer cooling time, dose locations with a high neutron contribution would show higher dose values, due to the non-linear relationship between burnup and neutron source term. At other locations dose rates are more dominated by contribution from the gamma sources. In these cases, short cooling time and lower burnup combinations with heat load comparable to the higher burnup and corresponding longer cooling time combinations would result in higher dose rates. However, in those cases, there would always be a compensatory effect, since for each dose location, higher neutron dose rates would be partly offset by lower gamma dose rates and vice versa.

Based on these considerations, average burnup and cooling time values are selected for all calculations for normal conditions, i.e values that are away from the extreme values. The selected values for MPC-37 and MPC-89 are shown in Table 5.0.1, and are based on a total heat loads presented in Table 1.2.3. A more bounding approach ~~is used~~ is used for MPC-32ML, independent from the heat loads, as discussed in more detail in this section, and Subsections 5.2.7 and 5.4.6.

For the accident conditions however, it is recognized that the bounding accident condition is the loss of water in the HI-TRAC VW, a condition that is neutron dominated due to the removal of the principal neutron absorber in the HI-TRAC VW (water). For this case, the upper bound burnup is selected, in order to maximize the neutron source strength of all assemblies in the basket, and a corresponding higher cooling time is selected in order to meet the overall heat load limit in the cask. The resulting burnup and cooling times values for accidents are therefore different from those for normal conditions and are listed in Table 5.0.2. In all cases, low initial enrichments are selected, which further increases the neutron source terms from the assemblies

With the burnup and cooling times selected based on above considerations, dose rates calculated for normal conditions will be reasonably conservative, while for accident conditions those will represent reasonable upper bound limits.

Table 5.0.1

DESIGN BASIS FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR MPC-37 AND MPC-89 NORMAL CONDITIONS

Design Basis Burnup and Cooling Times Zircaloy Clad Fuel		
MPC-37	**MPC-89**	~~**MPC-32ML**~~
45,000 MWD/MTU 4.5 Year Cooling 3.6 wt% U-235 Enrichment	45,000 MWD/MTU 5 Year Cooling 3.2 wt% U-235 Enrichment	~~45,000 MWD/MTU~~ ~~4.6 Year Cooling~~ ~~3.6 wt% U-235 Enrichment~~

Table 5.0.2

DESIGN BASIS FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR ACCIDENT CONDITIONS

Design Basis Burnup and Cooling Times Zircaloy Clad Fuel		
MPC-37	**MPC-89**	**MPC-32ML**
65,000 MWD/MTU 8 Year Cooling 4.8 wt% U-235 Enrichment	65,000 MWD/MTU 10 Year Cooling 4.8 wt% U-235 Enrichment	62,500 MWD/MTU 8 Year Cooling 4.6 wt% U-235 Enrichment

Table 5.0.3

DESIGN BASIS BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR MPC-32ML LOADING PATTERNS FOR NORMAL CONDITIONS

Burnup (MWD/MTU)	**Initial U-235 Enrichment (wt%)**	**Cooling Time (years)**	
		~~Time (years)~~Calculated Using Combination Curve in Table 2.1.9	**Used in Shielding Analysis**
15000~~15000~~	1.1~~1.5~~	3.42~~3~~	3
20000~~25000~~	1.1~~1.9~~	3.49~~3~~	3
25000~~35000~~	1.6~~2.7~~	3.59~~3.6~~	3.5
30000~~45000~~	2~~3.0~~	3.76~~4.5~~	3.6
35000~~55000~~	2.4~~3.4~~	4.04~~6.8~~	4
40000~~70000~~	2.6~~4.7~~	4.50~~14~~	4.5
45000	3	5.18	5
50000	3.3	6.14	6
55000	3.6	7.42	7
60000	3.6	9.07	9
65000	3.9	11.15	11
70000	4.2	13.70	13

Table 5.1.10

MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-32MLWITH 16X16D FUEL
~~BURNUP AND COOLING TIME~~
~~45,000 MWD/MTU AND 4.6 YEAR COOLING~~LOADING PATTERNS (SEE TABLE 5.0.3)

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	201	1	78	1	280	280
2	172	<1	<1	<1	173	173
3 (surface)	16	1	16	2	35	45
3 (overpack edge)	16	<1	37	<1	53	77
4 (center)	< 1	1	<1	<1	2	2
4 (mid)	4	<1	1	<1	5	6
4 (outer)	10	<1	20	<1	30	43

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.1.11

MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-32MLWITH 16X16D FUEL
LOADING PATTERNS (SEE TABLE 5.0.3)~~BURNUP AND COOLING TIME~~
~~45,000 MWD/MTU AND 4.6-YEAR COOLING~~

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	52	<1	15	<1	66	67
2	91	<1	1	<1	92	93
3	8	<1	7	<1	16	20
4 (center)	1	<1	1	<1	2	2

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

5.2 SOURCE SPECIFICATION

For MPC-37 and MPC-89 fuel assemblies, the~~The~~ neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 5 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] and [5.2.15] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.

For MPC-32ML, the neutron and gamma source terms and decay heat values were calculated with the TRITON and ORIGAMI modules of the SCALE 6.2.1 system [5.1.4]. SCALE 6.2.1 is a newer version of the SCALE code, providing substantial improvements over earlier versions such as SCALE 5.1.

Sample input files for SAS2H,~~ and~~ ORIGEN-S, TRITON and ORIGAMI are provided in Appendix 5.A. The gamma source term is actually comprised of three distinct sources. The first is a gamma source term from the active fuel region due to decay of fission products. The second source term is from ^{60}Co activity of the stainless steel structural material in the fuel element above and below the active fuel region. The third source is from (n,γ) reactions described below.

A description of the design basis fuel in MPC-37 and MPC-89 for the source term calculations is provided in Table 5.2.1, and in Table 5.2.17 for design basis fuel in MPC-32ML. Subsection 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

In performing the SAS2H~~ and~~, ORIGEN-S, TRITON and ORIGAMI calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Tables 5.2.1 and 5.2.17 resulted in conservative source term calculations.

5.2.1 Gamma Source

Tables 5.2.2 through 5.2.5, and Tables 5.2.18 and 5.2.19 provide the gamma source in MeV/s and photons/s as calculated with SAS2H~~ and~~/~~-~~ORIGEN-S (for source term calculations using SCALE 5.1) or TRITON/ORIGAMI (for source term calculations using SCALE 6.2.1) for the design basis zircaloy clad fuel at the burnups and cooling times used for normal and accident conditions.

Previous analyses were performed for the HI-STORM 100 system to determine the dose contribution from gammas as a function of energy [5.2.17]. The results of these analyses have revealed that, due to the magnitude of the gamma source at lower energies, photons with energies as low as 0.45 MeV must be included in the shielding analysis, but photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant. This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low. Therefore, all photons with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations.
The primary source of activity in the non-fuel regions of an assembly arises from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Therefore, inconel and stainless steel in the non-fuel regions are both assumed to have the same 0.8 gm/kg impurity level.

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. Since both designs may be loaded into the HI-STORM FW system, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers. Although BWR assembly grid spacers are zircaloy, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR zircaloy clad fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 0.8 gm/kg was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses for an 8x8 fuel assembly were used. These masses are also appropriate for the 10x10 assembly since the masses of the non-fuel hardware from a 10x10 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation.

The masses in Table 5.2.1 and Table 5.2.17 were used to calculate a ^{59}Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S (for source term calculations using SCALE 5.1) or ORIGAMI (for source term calculations using SCALE 6.2.1) to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.3] and is described here.

1. The activity of the ^{60}Co is calculated using ORIGEN-S (for source term calculations using SCALE 5.1) or ORIGAMI (for source term calculations using SCALE 6.2.1). The flux used in the calculation was the in-core fuel region flux at full power.

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.6 and Table 5.2.20. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.7 through 5.2.10 provide the ^{60}Co activity utilized in the shielding calculations for normal and accident conditions for the non-fuel regions of the assemblies in the MPC-37 and the MPC-89. Table 5.2.21 provide those data for the assemblies in the MPC-32ML.

In addition to the two sources already mentioned, a third source arises from (n,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments of 3.2 and 3.6 wt% were chosen for the BWR and PWR design basis fuel assemblies under normal conditions, respectively. For the accident conditions, ~~a~~ fuel enrichments of 4.8 wt% ~~was chosen~~ for the BWR and PWR (MPC-37), and 4.6 wt% for the PWR (MPC-32ML) were chosen to accommodate the higher burnups of the selected source terms (see Table 5.0.2) in accordance with Table 5.2.24 of reference [5.2.17].

The neutron source calculated for the design basis fuel assemblies for the MPCs and the design basis fuel are listed in Tables 5.2.11 through 5.2.14, and Table 5.2.22 in neutrons/s for the selected burnup and cooling times used in the shielding evaluations for normal and accident conditions. The neutron spectrum is generated in ORIGEN-S (for source term calculations using SCALE 5.1) or ORIGAMI (for source term calculations using SCALE 6.2.1).

5.2.3 Non-Fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM FW system as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted as specified in Subsection 2.1.

5.2.3.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore,

Subsection 5.4.4 discusses the effect on dose rate of the insertion of APSRs or CRAs into fuel assemblies.

5.2.4 Choice of Design Basis Assembly

The Westinghouse 17x17 and GE 10x10 assemblies were selected as design basis assemblies since they are widely used throughout the industry. Site specific shielding evaluations should verify that those assemblies and assembly parameters are appropriate for the site-specific analyses. Because of its large width, 16x16D (e.g., 16x16 Focus and 16x16 HTP fuel assemblies) was selected as design basis fuel assembly for MPC-32ML.

5.2.5 Decay Heat Loads and Allowable Burnup and Cooling Times

Subsection 2.1 describes the MPC maximum decay heat limits per assembly. The allowable burnup and cooling time limits are derived based on the allowable decay heat limits.

5.2.6 Fuel Assembly Neutron Sources

Neutron source assemblies (NSAs) are used in reactors for startup. There are different types of neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, polonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

During in-core operations, the stainless steel and inconel portions of the NSAs become activated, producing a significant amount of Co-60. A detailed discussion about NSAs is provided in reference [5.2.17], where it is concluded that activation from NSAs are bounded by activation from BPRAs.

For ease of implementation in the CoC, the restriction concerning the number of NSAs is being applied to all types of NSAs. In addition, conservatively NSAs are required to be stored in the inner region of the MPC basket as specified in Subsection 2.1. Further limitations allow for only one NSA to be stored in the MPC-37 or -(see Table 2.1.1a), or MPC-32ML (see Table 2.1.1b).

5.2.7 MPC-32ML Design Basis Burnup and Cooling Times

For the fuel to be loaded into the MPC-32ML canister, the uniform design basis loading curve (which specify burnup and cooling time combinations for each region of the cask) is provided in Table 2.1.9 using polynomial equation and corresponding polynomial coefficients.

In order to qualify the HI-STORM FW System with MPC-32ML and allowable burnup, cooling time combinations in Table 2.1.9, the considered range of burnup, enrichment and cooling time combinations is selected as follows:

- Burnups from 15 GWD/MTU to 70 GWD/MTU, in increments of 5 GWD/MTU;

- The cooling time is calculated for each burnup using the equation and polynomial coefficients in Table 2.1.9. The determined cooling times are rounded down to the nearest available cooling time in the calculated source terms library. The value of 3 year (minimum allowed cooling time) is used for all cooling times below 3 years;
- The lower bound enrichment that covers 99% of available data is determined for each selected burnup range.

The final set of the burnup, enrichment and cooling time combinations for normal conditions is provided in Table 5.0.3.

Table 5.2.1

DESCRIPTION OF DESIGN BASIS CLAD FUEL

	PWR (MPC-37)	**BWR (MPC-89)**
Assembly type/class	WE 17×17	GE 10×10
Active fuel length (in.)	144	144
No. of fuel rods	264	92
Rod pitch (in.)	0.496	0.51
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.374	0.404
Cladding thickness (in.)	0.0225	0.026
Pellet diameter (in.)	0.3232	0.345
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.522 (96% of theoretical)
Enrichment (w/o ^{235}U)	3.6	3.2
Specific power (MW/MTU)	43.48	30
Weight of UO_2 (kg)††	532.150	213.531
Weight of U (kg)††	469.144	188.249
No. of Water Rods/ Guide Tubes	25	2
Water Rod/ Guide Tube O.D. (in.)	0.474	0.98
Water Rod/ Guide Tube Thickness (in.)	0.016	0.03

†† Derived from parameters in this table.

Table 5.2.1 (continued) DESCRIPTION OF DESIGN BASIS FUEL		
	PWR (MPC-37)	**BWR (MPC-89)**
Lower End Fitting (kg)	5.9 (steel)	4.8 (steel)
Gas Plenum Springs (kg)	1.150 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.793 (inconel) 0.841 (steel)	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	6.89 (steel) 0.96 (inconel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.6

SCALING FACTORS USED IN CALCULATING THE ^{60}Co SOURCE

Region	**PWR (MPC-37)**	**BWR (MPC-89)**
Handle	N/A	0.05
Upper End Fitting	0.1	0.1
Gas Plenum Spacer	0.1	N/A
Expansion Springs	N/A	0.1
Gas Plenum Springs	0.2	0.2
Incore Grid Spacer	1.0	1.0
Lower End Fitting	0.2	0.15

Table 5.2.17 DESCRIPTION OF 16X16D DESIGN BASIS CLAD FUEL	
	PWR (MPC-32ML)
Assembly type/class	16x16D
Active fuel length (cm)	390
No. of fuel rods	236
Rod pitch (~~in.~~cm)	1.43
Cladding material	Zircaloy-4
Rod diameter (cm)	1.075
Cladding thickness (cm)	0.068
Pellet diameter (cm)	0.911
Pellet material	UO_2
Pellet density (g/cc)	10.45 (95.3% of theoretical)
Enrichment (w/o ^{235}U)	3.6
Specific power (MW/MTU)	36.56
Weight of UO_2 (kg)††	624.651
Weight of U (kg)††	552.~~703~~639
No. of Water Rods/ Guide Tubes	20
Water Rod/ Guide Tube O.D. (cm)	1.41
Water Rod/ Guide Tube Thickness (cm)	0.077

†† Derived from parameters in this table.

Table 5.2.~~19a~~18 CALCULATED 16X16D (MPC-32ML) PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR SELECTED DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS			
Lower Energy	**Upper Energy**	**~~44~~55,000 MWD/MTU ~~4.64~~5.5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.16E+15	3.76E+15
0.7	1.0	9.10E+14	1.07E+15
1.0	1.5	2.09E+14	1.67E+14
1.5	2.0	1.16E+13	6.62E+12
2.0	2.5	6.87E+12	3.05E+12
2.5	3.0	7.39E+11	2.69E+11
Total		3.30E+15	5.01E+15

Table 5.2.19~~b~~ CALCULATED 16X16D (MPC-32ML) PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS			
Lower Energy	**Upper Energy**	**62,500 MWD/MTU 8-Year Cooling~~25MWD/MTU 8-Year Cooling~~**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.07E+15	3.61E+15
0.7	1.0	5.30E+14	6.23E+14
1.0	1.5	1.67E+14	1.34E+14
1.5	2.0	7.19E+12	4.11E+12
2.0	2.5	6.65E+11	2.95E+11
2.5	3.0	1.00E+11	3.64E+10
Total		2.78E+15	4.37E+15

Table 5.2.21

CALCULATED ^{60}Co SOURCE PER ASSEMBLY FOR 16X16D (MPC-32ML) ~~DESIGN BASIS FUEL~~ AT SELECTED DESIGN BASIS BURNUP AND COOLING TIME COMBINATIONS FOR NORMAL AND ACCIDENT CONDITIONS

Location	**~~445~~5,000 MWD/MTU and ~~4.645.5~~-Year Cooling (curies)**	**62,500 MWD/MTU and 8-Year Cooling (curies)~~62,500 MWD/MTU and 8-Year Cooling (curies)~~**
Upper End Fitting	41.46	29.76~~29.76~~
Gas Plenum Springs	11.37	8.16~~7.11~~
Gas Plenum Spacer	19.80	14.21~~16.32~~
Incore Grid Spacers	851.14	610.89~~610.89~~
Lower End Fitting	145.04	104.10~~104.10~~

Table 5.2.22 CALCULATED 16X16D (MPC-32ML) PWR NEUTRON SOURCE PER ASSEMBLY AT SELECTED DESIGN BASIS ~~DESIGN BASIS~~ BURNUP AND COOLING TIME COMBINATIONS FOR NORMAL AND -ACCIDENT CONDITIONS			
Lower Energy (MeV)	**Upper Energy (MeV)**	**~~44~~55,000 MWD/MTU ~~4.64~~5.5-Year Cooling (Neutrons/s)**	**62,500 MWD/MTU 8-Year Cooling (Neutrons/s)~~62,500 MWD/MTU 8-Year Cooling (Neutrons/s)~~**
1.0e-01	4.0e-01	4.75E+07	7.29E+07
4.0e-01	9.0e-01	1.04E+08	1.59E+08
9.0e-01	1.4	1.04E+08	1.59E+08
1.4	1.85	8.27E+07	1.27E+08
1.85	3.0	1.53E+08	2.36E+08
3.0	6.43	1.40E+08	2.15E+08
6.43	20.0	1.34E+07	2.06E+07
Totals		6.44E+08	9.89E+08

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES†

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
PWR (MPC-37)					
Lower End Fitting	0.0	2.738	2.738	SS304	SS304
Space	2.738	3.738	1.0	zircaloy	void
Fuel	3.738	147.738	144.0	fuel & zircaloy	fuel & zircaloy
Gas Plenum Springs	147.738	151.916	4.178	SS304 & inconel	SS304
Gas Plenum Spacer	151.916	156.095	4.179	SS304 & inconel	SS304
Upper End Fitting	156.095	159.765	3.670	SS304 & inconel	SS304
BWR (MPC-89)					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144.0	fuel & zircaloy	fuel & zircaloy
Space	151.385	157.385	6.0	zircaloy	void
Gas Plenum Springs	157.385	166.865	9.48	SS304 & zircaloy	SS304
Expansion Springs	166.865	168.215	1.35	SS304	SS304
Upper End Fitting	168.215	171.555	3.34	SS304	SS304
Handle	171.555	176	4.445	SS304	SS304

† All dimensions start at the bottom of the fuel assembly. The length of the fuel shims must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.1 (Continued)

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE 16X16D FUEL ASSEMBLIES†

Region	Start (cm)	Finish (cm)	Length (cm)	Actual Material	Modeled Material
PWR (MPC-32ML) (Note 1)					
Lower End Fitting	0.0	34.002	34.002	SS304	SS304
Space	34.002	38.612	4.61	zircaloy	void
Fuel	38.612	428.612	390	fuel & zircaloy	fuel & zircaloy
Gas Plenum Springs	428.612	446.189	17.577	SS304 & inconel	SS304
Gas Plenum Spacer	446.189	466.360	20.171	SS304 & inconel	SS304
Upper End Fitting	466.360	489.700	23.34	SS304 & inconel	SS304

Note 1: The width of the fuel assembly is modeled as 22.96 cm.

† All dimensions start at the bottom of the fuel assembly. The length of the fuel shims must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Lower End Fitting (PWR MPC-37)	1.849	SS304	100
Gas Plenum Springs (PWR MPC-37)	0.23626	SS304	100
Gas Plenum Spacer (PWR MPC-37)	0.33559	SS304	100
Upper End Fitting (PWR MPC-37)	1.8359	SS304	100
Lower End Fitting (BWR)	1.5249	SS304	100
Gas Plenum Springs (BWR)	0.27223	SS304	100
Expansion Springs (BWR)	0.69514	SS304	100
Upper End Fitting (BWR)	1.4049	SS304	100
Handle (BWR)	0.26391	SS304	100
Lower End Fitting (MPC-32ML)	0.6022	SS304	100
Gas Plenum Springs (MPC-32ML)	0.159	SS304	100
Gas Plenum Spacer (MPC-32ML)	0.159	SS304	100
Upper End Fitting (MPC-32ML)	1.00325	SS304	100

indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is less than the dose rate from BPRAs (the increase in dose rate on the radial surface due to CRAs and APSRs are virtually negligible). For the surface dose rate at the bottom, the value for the CRA is comparable to or higher than the value from the BPRA. The increase in the bottom dose rates due to the presence of CRAs is on the order of 10-15% (based on bounding configuration 1 in [5.2.17]). The dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied.

While the evaluations described above are based on conservative assumptions, the conclusions can vary slightly depending on the number of CRAs and their operating conditions.

5.4.5 Effect of Uncertainties

The design basis calculations presented in this chapter are based on a range of conservative assumptions, but do not explicitly account for uncertainties in the methodologies, codes and input parameters, that is, it is assumed that the effect of uncertainties is small compared to the numerous conservatisms in the analyses. To show that this assumption is valid, calculations have previously been performed as "best estimate" calculations and with estimated uncertainties added [5.4.9]. In all scenarios considered (e.g., evaluation of conservatisms in modeling assumptions, uncertainties associated with MCNP as well as the depletion analysis (including input parameters), etc.), the total dose rates long with uncertainties are comparable to, or lower than, the corresponding values from the design basis calculations. This provides further confirmation that the design basis calculations are reasonable and conservative.

5.4.6 MPC-32ML ~~with Regionalized~~ Loading Patterns Dose Rates

The dose rates provided in Section 5.1 are the maximum dose rates for HI-STORM FW with MPC-32ML for conservative loading patterns in Table 5.0.3. Table 5.4.9 and Table 5.4.10 provide adjacent and 1-m dose rates for selected burnup-enrichment-cooling time combinations from Table 5.0.3.

The distance dose rates for arrays of HI-STORM FWs with MPC-32ML are provided in Table 5.4.11 for the most bounding loading pattern from Table 5.0.3. It should be noted that the distance dose rates at Table 5.4.11 are more than those provided in Table 5.1.3 because the dose rates in Table 5.4.11 are bounding, but the dose rates in Table 5.1.3 are representative.

~~As discussed in Section 5.2, there is only one heat load pattern with a uniform loading configuration for MPC-32ML. However, different burnup, enrichment and cooling time~~

~~combinations may produce same decay heat, but different source terms. Additional regionalized shielding analysis is provided in this subsection by dividing the MPC-32ML basket cells into Regions 1 to 3, where Region 1 is the innermost cells, and Region 3 is the outermost cells. The fuel burnup, enrichment and cooling time combinations in Table 5.4.9 are used to calculate the adjacent and 1-m dose rates for HI-STORM FW with MPC-32ML. The heat load of each combination is either more than the decay heat limit per cell, or for the minimum cooling time of 3 years. Conservative enrichments are also considered for all combinations. Using very low enrichments (e.g. 1 wt%) with highly burned (e.g. 65 GWd/mtU) fuel would be unrealistically conservative since no such fuel exists. Hence the lower bound enrichment is selected as a function of the burnup, based on a review of actual fuel assemblies in the industry and other Holtec approved cask systems (e.g., HI-STAR 190 SAR [5.4.10] Appendix 7.C). Based on this approach, the source terms used in the analyses are reasonably bounding for all realistically expected assemblies. Each burnup, enrichment, and cooling time combination can be in Region 1, Region 2, and/or Region 3 cells. The maximum adjacent and 1-m dose rates are provided in Tables 5.4.10 and 5.4.11, respectively.~~ Higher concrete density may be used in site specific shielding analysis to further lower the occupational dose rates.

Table 5.4.9

ADJACENT DOSE RATES FOR SELECTED HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS

~~BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR 3-REGION REGIONALIZED LOADING PATTERNS~~

Dose Point Location	**Totals + BPRA (mrem/hr)**				
	25,000 MWD/MTU 3.5-Year Cooling	**30,000 MWD/MTU 3.6-Year Cooling**	**40,000 MWD/MTU 4.5-Year Cooling**	**5~~5~~0,000 MWD/MTU 6.~~8~~-Year Cooling**	**60,000 MWD/MTU 9-Year Cooling**
1	260	280	262	219	162
2	152	163	139	109	80
3 (surface)	41	44	45	41	36
3 (overpack edge)	74	77	75	69	58
4 (center)	1	1	1	1	2
4 (mid)	5	6	6	5	5
4 (outer)	41	43	42	38	32

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.

—Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.~~The "Fuel Gammas" category includes gammas from the spent fuel, 60Co from the spacer grids, and 60Co from the BPRAs in the active fuel region.~~

•

~~Burnup (MWD/MTU)~~	~~Initial U-235 Enrichment (wt%)~~	~~Cooling Time (years)~~
~~15000~~	~~1~~	~~3~~
~~25000~~	~~1.4~~	~~3.2~~
~~35000~~	~~1.8~~	~~4.2~~
~~55000~~	~~3.1~~	~~8~~
~~60000~~	~~3.8~~	~~9.5~~

Table 5.4.10

1-METER DOSE RATES FOR SELECTED HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS

~~BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR~~

Dose Point Location	Totals + BPRA (mrem/hr)				
	25,000 MWD/MTU 3.5-Year Cooling	30,000 MWD/MTU 3.6-Year Cooling	40,000 MWD/MTU 4.5-Year Cooling	50,000 MWD/MTU 6-Year Cooling	60,000 MWD/MTU 9-Year Cooling
1	60	64	57	46	34
2	81	87	74	58	42
3	19	20	19	17	14
4 (center)	2	2	2	2	2

~~3 REGION REGIONALIZED LOADING PATTERNS~~

~~MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW OVERPACK~~
~~FOR NORMAL CONDITIONS~~
~~MPC-32MLWITH 16X16D FUEL~~
~~BURNUP AND COOLING TIME~~
~~REGIONALIZED LOADING PATTERNS~~

~~Dose Point Location~~	~~Fuel Gammas (mrem/hr)~~	~~(n,γ) Gammas (mrem/hr)~~	~~^{60}Co Gammas (mrem/hr)~~	~~Neutrons (mrem/hr)~~	~~Totals (mrem/hr)~~	~~Totals with BPRAs (mrem/hr)~~

~~1~~	~~206~~	~~1~~	~~87~~	~~2~~	~~295~~	~~295~~
~~2~~	~~170~~	~~1~~	~~< 1~~	~~1~~	~~172~~	~~172~~
~~3 (surface)~~	~~16~~	~~1~~	~~18~~	~~2~~	~~37~~	~~47~~
~~3 (overpack edge)~~	~~12~~	~~< 1~~	~~45~~	~~1~~	~~57~~	~~82~~
~~4 (center)~~	~~< 1~~	~~1.0~~	~~0.2~~	~~0.2~~	~~1.5~~	~~1.7~~
~~4 (mid)~~	~~4~~	~~< 1~~	~~1~~	~~< 1~~	~~6~~	~~6~~
~~4 (outer)~~	~~10~~	~~< 1~~	~~22~~	~~< 1~~	~~33~~	~~45~~

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

~~Notes:~~
~~Refer to Figure 5.1.1 for dose locations.~~
~~Values are rounded to nearest integer where appropriate.~~
~~Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.~~

~~Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.~~

~~The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.~~

Table 5.4.11

MAXIMUM DOSE RATES FOR ARRAYS OF HI-STORM FWs WITH MPC-32ML LOADING PATTERNS (SEE TABLE 5.0.3)

Array Configuration	**1 Cask**	**2x2**	**2x3**	**2x4**	**2x5**
HI-STORM FW Overpack					
Annual Dose (mrem/year)	9	22	12	16	20
Distance to Controlled Area Boundary (meters)	400	400	500	500	500

~~DELETED.~~

~~MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK FOR NORMAL CONDITIONS MPC-32MLWITH 16X16D FUEL BURNUP AND COOLING TIME REGIONALIZED LOADING PATTERNS~~

~~**Dose Point Location**~~	~~**Fuel Gammas (mrem/hr)**~~	~~**(n,γ) Gammas (mrem/hr)**~~	~~**^{60}Co Gammas (mrem/hr)**~~	~~**Neutrons (mrem/hr)**~~	~~**Totals (mrem/hr)**~~	~~**Totals with BPRAs (mrem/hr)**~~
~~1~~	~~51~~	~~< 1~~	~~16~~	~~< 1~~	~~68~~	~~68~~
~~2~~	~~90~~	~~< 1~~	~~1~~	~~< 1~~	~~92~~	~~92~~
~~3~~	~~8~~	~~< 1~~	~~8~~	~~< 1~~	~~17~~	~~21~~
~~4 (center)~~	~~0.6~~	~~0.4~~	~~0.7~~	~~0.3~~	~~2.0~~	~~2.5~~

~~Notes:~~

- ~~Refer to Figure 5.1.1 for dose locations.~~
- ~~Values are rounded to nearest integer where appropriate.~~
- ~~The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.~~

5.5 REGULATORY COMPLIANCE

Chapters 1 and 2 and this chapter of this SAR describe in detail the shielding structures, systems, and components (SSCs) important to safety.

The shielding-significant SSCs important to safety have been evaluated in this chapter and their impact on personnel and public health and safety resulting from operation of an independent spent fuel storage installation (ISFSI) utilizing the HI-STORM FW system has been evaluated.

It has been shown that the design of the shielding system of the HI-STORM FW system is in compliance with 10CFR72 and that the applicable design and acceptance criteria including 10CFR20 have been satisfied. Thus, this shielding evaluation provides reasonable assurance that the HI-STORM FW system will allow safe storage of spent fuel in full conformance with 10CFR72.

5.6 REFERENCES

[5.1.1] LA-UR-03-1987, MCNP — A General Monte Carlo N-Particle Transport Code, Version 5, April 24, 2003 (Revised 10/3/05).

[5.1.2] I.C. Gauld, O.W. Hermann, "SAS2: A Coupled One-Dimensional Depletion and Shielding Analysis Module," ORNL/TM-2005/39, Version 5.1, Vol. I, Book 3, Sect. S2, Oak Ridge National Laboratory, November 2006.

[5.1.3] I.C. Gauld, O.W. Hermann, R.M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," ORNL/TM-2005/39, Version 5.1, Vol. II, Book 1, Sect. F7, Oak Ridge National Laboratory, November 2006.

[5.1.4] SCALE Manual B.T. Rearden and M.A. Jessee, Eds., *SCALE Code System*, ORNL/TM-2005/39, Version 6.2.1, Oak Ridge National Laboratory, Oak Ridge, Tennessee (2016). Available from Radiation Safety Information Computational Center as CCC-834.

[5.2.1] NUREG-1536, SRP for Dry Cask Storage Systems, USNRC, Washington, DC, January 1997.

[5.2.2] A.G. Croff, M.A. Bjerke, G.W. Morrison, L.M. Petrie, "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code," ORNL/TM-6051, Oak Ridge National Laboratory, September 1978.

[5.2.3] A. Luksic, "Spent Fuel Assembly Hardware: Characterization and 10CFR 61 Classification for Waste Disposal," PNL-6906-vol. 1, Pacific Northwest Laboratory, June 1989.

[5.2.4] J.W. Roddy et al., "Physical and Decay Characteristics of Commercial LWR Spent Fuel," ORNL/TM-9591/V1&R1, Oak Ridge National Laboratory, January 1996.

[5.2.5] "Characteristics of Spent Fuel, High Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation," DOE/RW-0184, U.S. Department of Energy, December 1987.

[5.2.6] Not Used.

APPENDIX 5.A

SAMPLE INPUT FILES FOR SAS2H, ORIGEN-S, TRITON, ORIGAMI, AND MCNP

Proprietary Appendix Withheld in Accordance with 10 CFR 2.390