GLOSSARY

ALARA is an acronym for As Low As Reasonably Achievable

Ancillary or Ancillary Equipment is the generic name of a device used to carry out short term operations.

BLEU Fuel is blended low enriched uranium (BLEU) fuel that is the same as commercial spent fuel but with a higher cobalt impurity.

Bottom Lid means the removable lid that fastens to the bottom of the HI-TRAC VW transfer cask body to create a gasketed barrier against in-leakage of pool water in the space around the MPC.

BWR is an acronym for Boiling Water Reactor.

CG is an acronym for center of gravity.

Commercial Spent Fuel or CSF refers to nuclear fuel used to produce energy in a commercial nuclear power plant.

Confinement Boundary is the outline formed by the all-welded cylindrical enclosure of the MPC shell, MPC baseplate, MPC lid, MPC port cover plates, and the MPC closure ring which provides redundant sealing.

Confinement System means the Multi-Purpose Canister (MPC) which encloses and confines the spent nuclear fuel during storage.

Controlled Area means that area immediately surrounding an ISFSI for which the owner/user exercises authority over its use and within which operations are performed.

Cooling Time (or post-irradiation cooling time) for a spent fuel assembly is the time between reactor shutdown and the time the spent fuel assembly is loaded into the MPC.

Critical Characteristic means a feature of a component or assembly that is necessary for the proper safety function of the component or assembly. Critical characteristics of a material are those attributes that have been identified, in the associated material specification, as necessary to render the material's intended function.

DAS is the abbreviation for the Decontamination and Assembly Station. It means the location where the Transfer Cask is decontaminated and the MPC is processed (i.e., where all operations culminating in lid and closure ring welding are completed).

DBE means Design Basis Earthquake.

DCSS is an acronym for Dry Cask Storage System.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not replaced with dummy fuel rods, missing structural components such as grid spacers, whose structural integrity has been impaired such that geometric rearrangement of fuel or gross failure of the cladding is expected based on engineering evaluations, or those that cannot be handled by normal means. Fuel assemblies that cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Container (or Canister) or DFC means a specially designed enclosure for damaged fuel or fuel debris which permits flow of gaseous and liquid media while minimizing dispersal of gross particulates.

Damaged Fuel Isolators or DFIs are a specially designed barriers installed at the top and bottom of the storage cell space which permit flow of gaseous and liquid media while preventing the potential migration of fissile material from fuel assemblies with cladding damage. DFIs are used ONLY with damaged fuel assemblies which can be handled by normal means and whose structural integrity is such that geometric rearrangement of fuel is not expected. Damaged fuel stored in DFIs may contain missing or partial fuel rods and or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks.

Design Basis Load (DBL) is a loading which bounds one or more events that are applicable to the storage system during its service life.

Design Heat Load is the computed heat rejection capacity of the HI-STORM system with a certified MPC loaded with CSF stored in uniform storage with the ambient at the normal temperature and the peak cladding temperature (PCT) limit at 400ºC. The Design Heat Load is less than the thermal capacity of the system by a suitable margin that reflects the conservatism in the system thermal analysis.

Design Life is the minimum duration for which the component is engineered to perform its intended function set forth in this SAR, if operated and maintained in accordance with this SAR.

Design Report is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety. The SAR serves as the Design Report for the HI-STORM FW System.

Design Specification is a document prepared in accordance with the quality assurance requirements of 10CFR72 Subpart G to provide a complete set of design criteria and functional requirements for a system, structure, or component, designated as Important to Safety, intended

boundary for storage conditions.

MPC Transfer means transfer of the MPC between the overpack and the transfer cask which begins when the MPC is lifted off the HI-TRAC bottom lid and ends when the MPC is supported from beneath by the overpack (or the reverse).

NDT is an acronym for Nil Ductility Transition Temperature, which is defined as the temperature at which the fracture stress in a material with a small flaw is equal to the yield stress in the same material if it had no flaws.

Neutron Absorber is a generic term to indicate any neutron absorber material qualified for use in the HI-STORM FW System.

Neutron Shielding means a material used to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), Neutron Source Assemblies (NSAs), water displacement guide tube plugs, orifice rod assemblies, Instrument Tube Tie Rods (ITTRs), Guide Tube Anchors (GTAs), vibration suppressor inserts, and components of these devices such as individual rods.

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

Plain Concrete is concrete that is unreinforced.

Post-Core Decay Time (PCDT) is synonymous with cooling time.

PWR is an acronym for pressurized water reactor.

Reactivity is used synonymously with effective neutron multiplication factor or k-effective.

Regionalized Fuel Storage is a term used to describe an optimized fuel loading strategy wherein the storage locations are ascribed to distinct regions each with its own maximum allowable specific heat generation rate.

Removable Shielding Girdle is an ancillary designed to be installed to provide added shielding to the personnel working in the top region of the transfer cask.

Repaired/Reconstituted Fuel Assembly is a spent fuel assembly which contains dummy fuel rods that displaces an amount of water greater than or equal to the original fuel rods and/or which contains structural repairs so it can be handled by normal means. If irradiated dummy

stainless steel rods are present in the fuel assembly, the dummy/replacement rods will be considered in the site specific dose calculations.

SAR is an acronym for Safety Analysis Report.

Service Life means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of this FSAR. Service Life may be much longer than the Design Life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

Short-term Operations means those normal operational evolutions necessary to support fuel loading or fuel unloading operations. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and onsite handling of a loaded HI-TRAC VW transfer cask or HI-STORM FW overpack.

Single Failure Proof in order for a lifting device or special lifting device to be considered single failure proof, the design must follow the guidance in NUREG-0612, which requires that a single failure proof device have twice the normal safety margin. This designation can be achieved by either providing redundant devices (load paths) or providing twice the design factor as required by the applicable code.

SNF is an acronym for spent nuclear fuel.

SSC is an acronym for Structures, Systems and Components.

STP is Standard Temperature and Pressure conditions.

TAL is an acronym for the Threaded Anchor Location. TALs are used in the HI-STORM FW and HI-TRAC VW casks as well as the MPCs.

Thermo-siphon is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC fuel basket.

Traveler means the set of sequential instructions used in a controlled manufacturing program to ensure that all required tests and examinations required upon the completion of each significant manufacturing activity are performed and documented for archival reference.

Undamaged Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means; or b) a BWR fuel assembly with an intact channel, a maximum planar average initial 3.3 wt% U-235, without known or suspected GROSSLY BREACHED SPENT FUEL RODS, and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an

FIGURE 1.2.9 DELETED

2.1.6 Radiological Parameters for Design Basis SNF

The principal radiological design criteria for the HI-STORM FW System are the 10CFR72 §104 and §106 operator-controlled boundary dose rate limits, and the requirement to maintain operational dose rates as low as reasonably achievable (ALARA). The radiation dose is directly affected by the gamma and neutron source terms of the assembly, which is a function of the assembly type, and the burnup, enrichment and cooling time of the assemblies. Dose rates are further directly affected by the size and arrangement of the ISFSI, and the specifics of the loading operations. All these parameters are site-dependent, and the compliance with the regulatory dose rate requirements are performed in site-specific calculations. The evaluations here are therefore performed with reference fuel assemblies, and with parameters that result in 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 2.1.1 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 criteriona 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 2.1.1.

2.1.6.1 Radiological Parameters for Spent Fuel and Non-fuel Hardware in MPC-32ML, MPC-37 and MPC-89

MPC-32ML is authorized to store 16x16D spent fuel with burnup - cooling time combinations as given in Table 2.1.9. Spent fuel with burnup – cooling time combinations authorized for storage according to the alternative storage patterns shown in Figures 1.2.3 through 1.2.5 (MPC-37) and 1.2.6 through 1.2.7 (MPC-89) are given in Table 2.1.10.

The burnup and cooling time for every fuel assembly loaded into the MPC-32ML, MPC-37 and MPC-89 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 or Table 2.1.10

Minimum cooling time must also meet limits specified in Tables 2.1.1a and 2.1.1b. If the calculated *Ct* is less than the cooling time limits in Tables 2.1.1a or 2.1.1b, the minimum cooling time in table is used.

For MPC-37 and MPC-89, the coefficients for above equation for the assembly in an individual cell depend on the heat load limit in that cell, Table 2.1.10 lists the coefficients for several heat load limit ranges. Note that the heat load limits are only used for the lookup of the coefficients in that table, and do not imply any equivalency. Specifically, meeting heat load limits is not a substitute for meeting burnup and cooling time limits, and vice versa.

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.

<u>Method A: Burnup Verification Through Quantitative Burnup Measurement</u>

For each assembly in the MPC-37 where burnup credit is required, the minimum burnup is determined from the burnup requirement applicable to the loading configuration chosen for the cask (see Table 2.1.7). A measurement is then performed that confirms that the fuel assembly burnup exceeds this minimum burnup. The measurement technique may be calibrated to the reactor records for a representative set of assemblies. The assembly burnup value to be compared with the minimum required burnup should be the measured burnup value as adjusted by reducing the value by a combination of the uncertainties in the calibration method and the measurement itself.

<u>Method B: Burnup Verification Through an Administrative Procedure and Qualitative Measurements</u>

Depending on the location in the basket, assemblies loaded into a specific MPC-37 can either be fresh, or have to meet a single minimum burnup value. The assembly burnup value to be compared with the minimum required burnup should be the reactor record burnup value as adjusted by reducing the value by the uncertainties in the reactor record value. An administrative procedure shall be established that prescribes the following steps, which shall be performed for each cask loading:

Table 2.1.1a MATERIAL TO BE STORED		
PARAMETER	**VALUE**	
	MPC-37	MPC-89
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the applicable array/class.	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, with or without channels, fuel debris meeting the limits in Table 2.1.3 for the applicable array/class.
Cladding Type	ZR (see Glossary for definition)	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment	Depending on soluble boron levels or burnup credit and assembly array/class as specified in Table 2.1.6 and Table 2.1.7.	≤ 5.0 wt. % U-235
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 3 21 years and meeting the equation in section 2.1.6 Maximum Assembly Average Burnup: 68.2 GWd/mtU	Minimum Cooling Time: 1.2823 years and meeting the equation in section 2.1.6 Maximum Assembly Average Burnup: 65 GWd/mtU
Non-fuel hardware post-irradiation cooling time and burnup	Minimum Cooling Time: 3 2 years Maximum Burnup†: - BPRAs, WABAs and vibration suppressors: 60 GWd/mtU - TPDs, NSAs, APSRs, RCCAs, CRAs, CEAs, water displacement guide tube plugs and orifice rod assemblies: 630 GWd/mtU - ITTRs: not applicable	N/A
Decay heat per fuel storage location	Regionalized Loading: See Tables 1.2.3a and 1.2.3d	Regionalized Loading: See Tables 1.2.4a and 1.2.4b.

† Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

Table 2.1.1b

MATERIAL TO BE STORED

PARAMETER	VALUE
	MPC-32ML
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the 16x16D array/class only.
Cladding Type	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment	Depending on soluble boron levels and assembly array/class as specified in Table 2.1.6.
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 3 years and meeting the equation in Section 2.1.6 Maximum Assembly Average Burnup: 68.2 GWd/mtU
Non-fuel hardware post-irradiation cooling time and burnup†	Minimum Cooling Time: 3 years Maximum Burnup: - BPRAs, WABAs and vibration suppressors: 60 GWd/mtU - TPDs, NSAs, APSRs, RCCAs, CRAs, CEAs, water displacement guide tube plugs and orifice rod assemblies: 630 GWd/mtU - ITTRs: not applicable
Decay heat per fuel storage location	Uniform Loading per Table 1.2.3b.
Fuel Assembly Nominal Length (in)	≤ 196.122 (including NFH and DFC)
Fuel Assembly Width (in)	≤ 9.04 (nominal design)
Fuel Assembly Weight (lb)	≤ 1860 (without NFH) ≤ 2120 (with NFH) ≤ 2200 (including DFC and NFH).

† Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

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

Polynomial Coefficients, see Paragraph 2.1.6.1			
A	**B**	**C**	**D**
6.7667E-14	-3.6726E-09	8.1319E-05	2.7951E+00

Notes:

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

TABLE 2.1.10
BURNUP AND COOLING TIME FUEL QUALIFICATION REQUIREMENTS
FOR MPC-37 AND MPC-89

ReferenceCell Decay Heat Load Limit[1][2] (kW)	**Polynomial Coefficients, see Paragraph 2.1.6.1**			
	A	**B**	**C**	**D**
MPC-37				
≤ 0.85	1.68353E-13	-9.65193E-09	2.69692E-04	2.95915E-01
0.85 < decay heat ≤ 3.5	1.19409E-14	-1.53990E-09	9.56825E-05	-3.98326E-01
MPC-89				
≤ 0.32	1.65723E-13	-9.28339E-09	2.57533E-04	3.25897E-01
0.32 < decay heat ≤ 0.5	3.97779E-14	-2.80193E-09	1.36784E-04	3.04895E-01
0.5 < decay heat ≤ 0.75	1.44353E-14	-1.21525E-09	8.14851E-05	3.31914E-01
0.75 < decay heat ≤ 1.1	-7.45921E-15	1.09091E-09	-1.14219E-05	9.76224E-01
1.1 < decay heat ≤ 1.45	3.10800E-15	-7.92541E-11	1.56566E-05	6.47040E-01
1.45 < decay heat ≤ 1.6	-8.08081E-15	1.23810E-09	-3.48196E-05	1.11818E+00

Notes:

The maximum allowable decay heat load per fuel basket cell, i.e. a decay heat value that is equal to or greater than the appropriate uniform and regionalized decay heat load limits, is specified.

The burnup and cooling time for every fuel loaded into the MPC-37 or MPC-89 must satisfy the following equation:

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

where,

Ct = Minimum cooling time (years)

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[1] Decay heat per fuel assembly is presented.
[2] A decay heat value that is equal to or greater than the appropriate decay heat load limit.

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

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

Withheld in Accordance with 10 CFR 2.390

Figure 2.1.7 (Continued): Damaged Fuel Isolator (Typical)
Example configuration. Final configuration may vary with fuel type.

Table 2.2.3

TEMPERATURE LIMITS

HI-STORM FW Component	Normal Condition and Design Temperature Limits (°F)	Short Term Events†† Temperature Limits (°F)	Off-Normal and Accident Condition Temperature Limits † (°F)
MPC shell	600*	800*	800*
MPC basket	752	932	932
MPC basket shims	752	932	932
MPC lid	600*	800*	800*
MPC closure ring and port cover plates	500*	800*	800*
MPC baseplate	400*	800*	800*
HI-TRAC VW inner shell	-	600	700
HI-TRAC VW outer shell	-	500	700
HI-TRAC VW bottom lid	-	500	700
HI-TRAC VW water jacket shell	-	500	700**
HI-TRAC VW top flange	-	500	650
HI-TRAC VW bottom lid seals	-	400	N/A
HI-TRAC VW bottom lid bolts	-	400	800
HI-TRAC VW bottom flange	-	400	700
HI-TRAC VW radial neutron shield	-	311	N/A
HI-TRAC VW radial lead	-	600	600

†† Short term operations include, but are not limited to, MPC drying and onsite transport. The 1058°F temperature limit applies to MPCs containing all moderate burnup fuel. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F.

* Temperature limits in Table 1.A.6 shall take precedence if duplex stainless steels are used for the fabrication of confinement boundary components, as described in Appendix 1.A.

** For fire accidents, the steel structure is required to remain physically stable similar to HI-STORM overpack.

† For accident conditions that involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident), the permissible metal temperature of the steel parts is defined by Table 1A of ASME Section II (Part D) for Section III, Class 3 materials as 700°F. For the fire event, the structure is required to remain physically stable (no specific temperature limits apply)

General Notes: 1.The normal condition temperature limits are used in the design basis structural evaluations for MPC and HI-STORM. The short-term condition temperature limits are used in the design basis structural evaluations for HI-TRAC. All other short-term, off-normal, and accident condition structural evaluations are based on bounding temperatures from thermal evaluations presented in Chapter 4.

2. The temperature limits provided for HI-TRAC VW are applicable to Version V and V2 unless otherwise specified.

Table 2.2.3

TEMPERATURE LIMITS

HI-STORM FW Component	Normal Condition and Design Temperature Limits (°F)	Short Term Events†† Temperature Limits (°F)	Off-Normal and Accident Condition Temperature Limits † (°F)
gamma shield			
HI-TRAC VW Version V2 NSC steel	-	400	600
HI-TRAC VW Version V2 NSC Holtite-A	-	300	N/A350
Fuel Cladding	752 (Storage)	752 or 1058 (Short Term Operations)††	1058 (Off-Normal and Accident Conditions)
Overpack concrete	300 (see HI-STORM 100 FSAR Appendix 1.D)	300	650 (on local temperature of shielding concrete except for fire572
Overpack Lid Top and Bottom Plate	450	450	700572
Remainder of overpack steel structure	350	350	700
Damaged Fuel Isolator	752	932	932

F

each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices, with the exception of instrument tube tie rods (ITTRs), which may be stored in the assembly along with other types of non-fuel hardware.

As described in Chapter 1(see Tables 1.2.3 and 1.2.4), the loading of fuel in all HI-STORM FW MPCs will follow specific heat load limitations.

In order to offer the user more flexibility in fuel storage, the HI-STORM FW System offers several heat load patterns, each with two or more regions with different heat load limits. This is taken into consideration when calculating dose rates in this chapter. The regionalized storage patterns are guided by the considerations of minimizing occupational and site boundary dose to comply with ALARA principles.

Two different lids have been developed for the HI-STORM FW concrete overpack. The lid included in the initial application, referred to as "standard lid", and a revised design with overall improved shielding performance, referred to as "XL lid". Since by now essentially all installations utilize the "XL lid", all dose rates provided for MPC-37 and MPC-89 in this chapter are for that lid design, with the only exception being some tables in Section 5.4 which contain selected results for the "standard lid" from previous versions of this chapter for reference. The shielding analysis of HI-STORM FW with MPC-32ML is performed using a "standard lid" design. All references to the "lid" are to be understood to refer to the "XL lid", unless otherwise noted.

The sections that follow will demonstrate that the design of the HI-STORM FW dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

Acceptance Criteria

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The "controlled area" is defined in 10CFR72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.

2. The system designer must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed dry cask storage system are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.

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 or maximum dose rates, next to the casks and at various distances, to aid the user in applying ALARA considerations and planning of the ISFSI.
- 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 extent, a burnup and cooling time calculation that maximizes the dose rate under accident conditions needs to be selected.

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. This is further complicated by the regionalized loading patterns qualified from a thermal perspective and shown in Figure 1.2.3 through Figure 1.2.5 for MPC-37 and Figures 1.2.6 and 1.2.7 for MPC-89. These contain cells with substantially different heat load limits, and hence substantially different ranges of burnup, enrichment and cooling time combinations. The approach to cover all those variations in a conservative way is outlined below.

To prescribe radiological limits for the fuel to be loaded, loading curves are defined in Tables 2.1.9 and 2.1.10, where a loading curve specifies the minimum cooling time as a function of fuel burnup. Different loading curves are defined for the different heat load limits, so that the thermal and radiological requirements for the fuel in each cell are approximately aligned. However, it should be noted that thermal and radiological limits for each assembly are applied completely independent from each other. The uniform and regionalized loading curves for the fuel to be loaded in the MPC-37, MPC-32ML or MPC-89 canisters are discussed in Subsection 5.2.7.

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

To determine dose rates consistent with both the uniform and regionalized thermal loading, it is necessary to consider the ranges of burnup and cooling times from all loading curves. For that, 8 burnup values between 5 and 70 GWd/mtU are selected, and corresponding minimum required cooling times are established and used in the dose analyzes. The heat load patterns in Figures 1.2.3 through 1.2.7 contain from 5 to 20 regions each, i.e. from 5 to 20 principal locations with different heat load limit. Applying 8 burnup and cooling time combinations to each location would result up to 8^{20} = 1.15E+18 different burnup and cooling time loading arrangements per pattern. Analyzing and comparing those many arrangements would be excessive. Therefore, for the radiological evaluations, some regions and loading patterns (MPC-37) are combined using the highest heat load limit (source term) of each group. For MPC-37, the heat loads for each cell are based on the "Long" fuel heat loads in Figure 1.2.5a. The established bounding heat load limits are provided in Tables 5.0.3 and 5.0.4.

This then results in effectively only 2 or 5 regions to be independently varied for the considered bounding MPC-37 and MPC-89 patterns, and hence 8^2 = 64 or 8^5 = 32,768 different burnup and cooling time arrangements per pattern is to be analyzed, which is manageable. The selected burnup, enrichment and cooling time combinations for the uniform and regionalized loading patterns are listed in Tables 5.0.3, 5.0.4a, 5.0.4b and 5.0.5. The dose rates in the various important locations are calculated for each of these combination arrangements and the maximum is determined for each dose rate location. It should be noted that this maximum can be from a different loading arrangement in different locations.

Based on this approach, the source terms used in the analyses of MPC-37, MPC-32ML or MPC-89 are reasonably bounding for all realistically expected assemblies. All dose rates in this chapter are developed using this approach, unless noted otherwise. Also, as discussed in Section 5.2, the design basis BPRA activity is considered for MPC-37 in this chapter, unless noted otherwise.

All dose rates in Section 5.1 are developed using the approach discussed above. Some dose rates in Section 5.4 were retained from previous versions of the FSAR and that are based on a representative (while still conservative) uniform loading pattern, as discussed in that Section.

Table 5.0.3

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-37 LOADING PATTERNS BASED ON FIGURES 1.2.3 THROUGH 1.2.5 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)	Reference Decay Heat (kW)
High Heat Load Basket Regions	5000	1.1	1.0	3.5
	10000	1.1	1.0	
	20000	1.6	1.0	
	30000	2.4	1.4	
	40000	3.0	1.6	
	50000	3.6	2.0	
	60000	3.9	2.2	
	70000	4.5	2.8	
Low Heat Load Basket Regions	5000	1.1	1.4	0.85
	10000	1.1	2.0	
	20000	1.6	3.0	
	30000	2.4	4.0	
	40000	3.0	6.0	
	50000	3.6	10.0	
	60000	3.9	18.0	
	70000	4.5	29.0	

NOTE:

To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.3 through 1.2.5.

Table 5.0.4a

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-89 LOADING PATTERNS BASED ON FIGURE 1.2.6 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)	Reference Decay Heat (kW)
High Heat Load Basket Regions	5000	0.7	1.0	1.45
	10000	0.9	1.0	
	20000	1.6	1.0	
	30000	2.4	1.0	
	40000	3.0	1.2	
	50000	3.3	1.6	
	60000	3.7	1.8	
	70000	4.0	2.4	
Low Heat Load Basket Regions	5000	0.7	1.4	0.32
	10000	0.9	2.0	
	20000	1.6	3.0	
	30000	2.4	4.0	
	40000	3.0	6.0	
	50000	3.3	10.0	
	60000	3.7	18.0	
	70000	4.0	29.0	

NOTE:

To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.6.

Table 5.0.4b (continued)

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-89 LOADING PATTERNS BASED ON FIGURE 1.2.7 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)	Reference Decay Heat (kW)
Low Heat Load Basket Regions (Region 4)	5000	0.7	1.0	0.5
	10000	0.9	1.4	
	20000	1.6	2.2	
	30000	2.4	2.8	
	40000	3.0	3.5	
	50000	3.3	5.0	
	60000	3.7	7.0	
	70000	4.0	9.0	
Low Heat Load Basket Regions (Region 5)	5000	0.7	1.4	0.32
	10000	0.9	2.0	
	20000	1.6	3.0	
	30000	2.4	4.0	
	40000	3.0	6.0	
	50000	3.3	10.0	
	60000	3.7	18.0	
	70000	4.0	29.0	

NOTE:

To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.6.

Table 5.0.5

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

Burnup (MWD/MTU)	Initial U-235 Enrichment (wt%)	Cooling Time (years)	
		Calculated Using Combination Curve in Table 2.1.9	Used in Shielding Analysis
15000	1.1	3.42	3
20000	1.1	3.49	3
25000	1.6	3.59	3.5
30000	2	3.76	3.6
35000	2.4	4.04	4
40000	2.6	4.50	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

5.1.1 Normal and Off-Normal Operations

Chapter 12 discusses the potential off-normal conditions and their effect on the HI-STORM FW system. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

The 10CFR72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.

2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 11 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM FW system and presented in Table 2.3.2.

Figure 5.1.1 identifies the locations of the dose points referenced in the dose rate summary tables for the HI-STORM FW overpack. Dose Point #2 is located on the side of the cask at the axial mid-height. Dose Points #1 and #3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point #4 is the dose location on the overpack lid. The dose values reported at the locations shown on Figure 5.1.1 are averaged over a region that is approximately 1 foot in width.

Figure 5.1.2 identifies the location of the dose points for the HI-TRAC VW transfer cask. Dose Point Locations #1 and #3 are situated below and above the water jacket, respectively. In the case of the HI-TRAC VW Version V2, Dose Point Locations #1 and #3 are situated below and above the neutron shield, respectively. Dose Point #4 is the dose location on the HI-TRAC VW lid and dose rates below the HI-TRAC VW are estimated with Dose Point #5. Dose Point Location #2 is situated on the side of the cask at the axial mid-height.

The total dose rates presented are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Subsection 5.2.4.

Tables 5.1.1, 5.1.2 and 5.1.13 provide dose rates adjacent to and one meter from the HI-TRAC VW during normal conditions for the MPC-37, MPC-89 and MPC-32ML. The dose rates listed in Tables 5.1.1, 5.1.2 and 5.1.13 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water. It should be noted that the minimum lead thickness of HI-TRAC VW with MPC-32ML is more than that of HI-TRAC with MPC-37.

Tables 5.1.10 provides dose rates adjacent to and one meter from the HI-TRAC VW Version V2 during normal conditions for the MPC-89. The dose rates listed in Table 5.1.10 correspond to the normal condition in which the MPC is dry and the Gamma Shield Cylinder and Neutron Shield Cylinder are present.

Tables 5.1.5, 5.1.6 and 5.1.11 provide the design basis dose rates adjacent to the HI-STORM FW overpack during normal conditions for the MPC-37, MPC-89 and MPC-32ML. Tables 5.1.7, 5.1.8 and 5.1.12 provide the design basis dose rates at one meter from the HI-STORM FW overpack containing the MPC-37, MPC-89 and MPC-32ML, respectively.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. Table 5.1.3 presents the annual dose to an individual from a single HI-STORM FW cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time).

Figure 5.1.3 is an annual dose versus distance graph for the HI-STORM FW cask array configurations provided in Table 5.1.3. Figure 5.1.4 is an annual dose versus distance graph for the HI-STORM FW cask array configurations provided in Table 5.4.21. These curves, which are based on an 8760 hour occupancy, are provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Subsection 5.2.3 discusses the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM FW system. Subsection 5.4.4 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

The analyses summarized in this section demonstrate that the HI-STORM FW system is in compliance with the radiation and exposure objectives of 10CFR72.106. Since only representative dose rate values for normal conditions are presented in this chapter, compliance with 10CFR72.104 is not being evaluated. This will be performed as part of the site specific evaluations.

5.1.2 Accident Conditions

The 10CFR72.106 radiation dose limits at the controlled area boundary for design basis accidents are:

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose

Table 5.1.11

MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-32MLWITH 16X16D FUEL
LOADING PATTERNS (SEE TABLE 5.0.5)

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 5.1.12

MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-32MLWITH 16X16D FUEL
LOADING PATTERNS (SEE TABLE 5.0.5)

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.

Table 5.1.13

MAXIMUM DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-32MLWITH 16X16D FUEL
LOADING PATTERNS (SEE TABLE 5.0.5)

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)
ADJACENT TO THE HI-TRAC VW						
1	5	4	41	86	136	136
2	1627	14	<1	25	1666	1666
3	67	3	329	3	402	606
4	74	1	364	156	595	858
5	318	1	1887	527	2734	2734
ONE METER FROM THE HI-TRAC VW						
1	190	4	154	8	356	356
2	723	5	7	10	745	746
3	95	1	63	1	160	198
4	236	<1	229	25	490	634
5	168	<1	1028	133	1329	1329

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.
- 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.

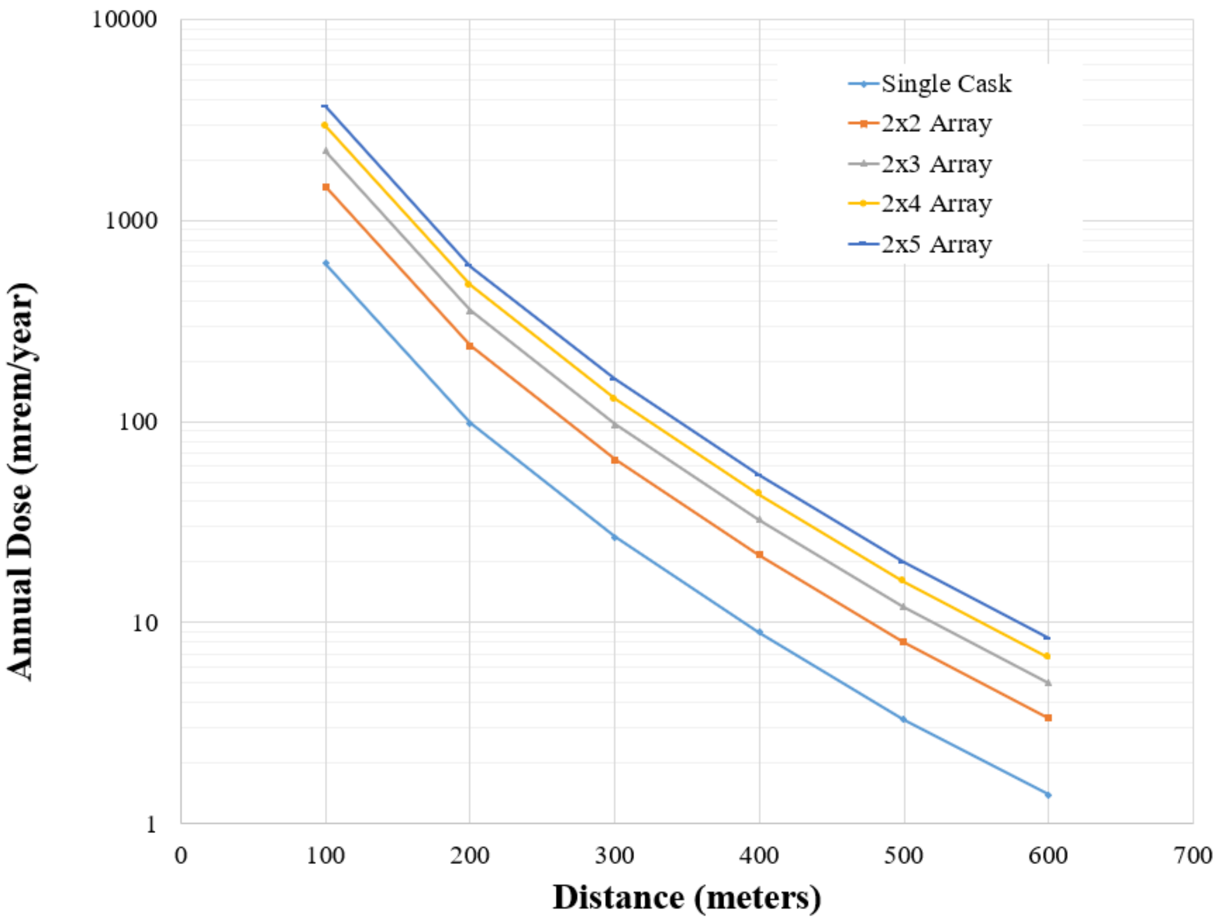

Figure 5.1.4

MAXIMUM ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-32ML FOR
BOUNDING UNIFORM PATTERNS (SEE TABLE 5.0.5)
(8760-HOUR OCCUPANCY ASSUMED)

5.2 SOURCE SPECIFICATION

The design basis neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the TRITON and ORIGAMI sequences of the SCALE 6.2.1 system [5.1.4], which is consistent with other approved Holtec applications [5.2.18]. For some additional calculations presented in Section 5.4, the neutron and gamma source terms available 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.

Sample input files for TRITON, ORIGAMI, SAS2H, and ORIGEN-S 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 5.2.18 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 TRITON, ORIGAMI, SAS2H, and ORIGEN-S 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 5.2.18 resulted in conservative source term calculations.

5.2.1 Gamma Source

Tables 5.2.2 through 5.2.5, and Tables 5.2.19 and 5.2.20 provide the gamma source in MeV/s and photons/s as calculated with TRITON and ORIGAMI 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 5.2.18 were used to calculate a ^{59}Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGAMI 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 ORIGAMI. 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 5.2.21. 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 5.2.22 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 were chosen for the BWR and PWR design basis fuel assemblies under normal and accident conditions, respectively, as discussed in Subsection 5.2.8.

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 5.2.23 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 ORIGAMI.

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, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different from that of a fuel assembly. Vibration suppressor inserts are considered to be in the same category as BPRAs for the purposes of the analysis in this chapter since these devices have the same configuration (long non-absorbing thimbles which extend into the active fuel region) as a BPRA without the burnable poison.

TPDs are made of stainless steel and contain a small amount of inconel. These devices extend down into the plenum region of the fuel assembly but typically do not extend into the active fuel region. Since these devices are made of stainless steel, there is a significant amount of cobalt-60 produced during irradiation. This is the only significant radiation source from the activation of steel and inconel.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24

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 (see Table 2.1.1a), or MPC-32ML (see Table 2.1.1b).

5.2.7 Design Basis Burnup and Cooling Times

For the fuel to be loaded into the HI-STORM FW system, the uniform and regionalized design basis loading curves (which specify burnup and cooling time combinations for each region of the cask) are provided in Tables 2.1.9 and 2.1.10 using polynomial equation and corresponding polynomial coefficients.

In order to qualify the HI-STORM FW System with allowable burnup, cooling time combinations in Tables 2.1.9 and 2.1.10, the considered range of burnup, enrichment and cooling time combinations is selected as follows:

- 5 GWD/MTU burnup and burnups from 10 GWD/MTU to 70 GWD/MTU, in increments of 10 GWD/MTU for MPC-37 and MPC-89, and burnups from 15 GWD/MTU to 70 GWD/MTU, in increments of 5 GWD/MTU for MPC-32ML;
- The cooling time is calculated for each burnup using the equation and polynomial coefficients in Tables 2.1.9 and 2.1.10. The determined cooling times are rounded down to the nearest available cooling time in the calculated source terms library, which provides a significant conservatism, especially, in the low cooling time area. For MPC-37 and MPC-89, the value of 1 year (minimum allowed cooling time) is used for all cooling times below 1 year. For MPC-32ML, the value of 3 year (minimum allowed cooling time) is used for all cooling times below 3 years;
- The appropriate burnup-specific lower bound enrichment is selected according to Table 5.2.17.

The final sets of the burnup, enrichment and cooling time combinations are provided in Tables 5.0.3 through 5.0.5.

5.2.8 Fuel Enrichment

As discussed in Subsection 5.2.2, enrichments have a significant impact on neutron dose rates, with lower enrichments resulting in higher dose rates at the same burnup. For assemblies with

higher burnups (which result in high neutron source terms) and/or locations that are more neutron dominated, the enrichment would therefore be important in order to present dose rates in a conservative way. However, it would be impractical and excessively conservative to perform all calculations at bounding low enrichment, since low enrichments are generally only found in lower burned assemblies. Therefore, a conservatively low enrichment value is selected based on the burnup. Specifically, based on industry information on more than 130,000 PWR and 185,000 BWR assemblies, the fuel assemblies are distributed over different burnup range bins (0 5, 5-10 … 70-75 GWd/mtU). For instance, for a given burnup group of 5-10 GWd/mtU, the data array includes the enrichments for the fuel assemblies with the burnup from 5,000 MWd/mtU to 9,999 MWd/mtU. Then, in each burnup group, the array of enrichments is sorted from low to high, and the array index that precedes a fraction of 99% of the population is determined. The fuel enrichment under this array position represents the lower bound enrichment that conservatively bounds 99% of the fuel assembly population. The calculated and finally established lower bound enrichment values are summarized in Table 5.2.17.

Given that the considered baskets contain a relatively large number of available cells for fuel loading, selecting the minimum enrichment for all assemblies is considered reasonably conservative. The typical content of the basket would have most assemblies well above the lower bound enrichment assumed in the analyses, so even if a small number of assemblies would be below the assumed minimum, that would have a negligible effect or be essentially inconsequential for the dose rates around the cask. Furthermore, the site-specific shielding analysis shall consider actual or bounding fuel enrichment. Therefore, an explicit lower enrichment limit for the fuel assemblies is not considered necessary.

Table 5.2.4

CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR A SELECTED BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Lower Energy	**Upper Energy**	**40,000 MWD/MTU 3.5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	8.07E+14	1.40E+15
0.7	1.0	3.72E+14	4.38E+14
1.0	1.5	8.11E+13	6.49E+13
1.5	2.0	6.36E+12	3.63E+12
2.0	2.5	6.88E+12	3.06E+12
2.5	3.0	5.69E+11	2.07E+11
Total		1.27E+15	1.91E+15

Table 5.2.5

CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Lower Energy	**Upper Energy**	**70,000 MWD/MTU 2.4-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	1.91E+15	3.32E+15
0.7	1.0	1.13E+15	1.33E+15
1.0	1.5	2.10E+14	1.68E+14
1.5	2.0	1.70E+13	9.70E+12
2.0	2.5	1.80E+13	8.00E+12
2.5	3.0	1.60E+12	5.84E+11
Total		3.29E+15	4.84E+15

Table 5.2.7

CALCULATED MPC-37 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT A SELECTED BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Location	30,000 MWD/MTU and 4-Year Cooling (curies)
Lower End Fitting	73.84
Gas Plenum Springs	14.39
Gas Plenum Spacer	10.22
Expansion Springs	N/A
Incore Grid Spacers	306.62
Upper End Fitting	49.12
Handle	N/A

Table 5.2.8

CALCULATED MPC-37 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Location	70,000 MWD/MTU and 2.8-Year Cooling (curies)
Lower End Fitting	133.42
Gas Plenum Springs	26.01
Gas Plenum Spacer	18.48
Expansion Springs	N/A
Incore Grid Spacers	554.05
Upper End Fitting	88.76
Handle	NA

Table 5.2.9

CALCULATED MPC-89 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT A SELECTED BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Location	40,000 MWD/MTU and 3.5-Year Cooling (curies)
Lower End Fitting	57.08
Gas Plenum Springs	17.44
Gas Plenum Spacer	N/A
Expansion Springs	3.17
Grid Spacer Springs	26.16
Upper End Fitting	15.86
Handle	1.98

Table 5.2.10

CALCULATED MPC-89 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Location	70,000 MWD/MTU and 2.4-Year Cooling (curies)
Lower End Fitting	92.98
Gas Plenum Springs	28.41
Gas Plenum Spacer	N/A
Expansion Springs	5.17
Grid Spacer Springs	42.61
Upper End Fitting	25.83
Handle	3.23

Table 5.2.14 CALCULATED MPC-89 BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS		
Lower Energy (MeV)	**Upper Energy (MeV)**	**70,000 MWD/MTU 2.4-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	5.43E+07
4.0e-01	9.0e-01	1.18E+08
9.0e-01	1.4	1.18E+08
1.4	1.85	9.43E+07
1.85	3.0	1.75E+08
3.0	6.43	1.61E+08
6.43	20.0	1.58E+07
Totals		7.37E+08

Table 5.2.17

LOWER BOUND INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS[1]

Burnup Range[2] (MWD/MTU)	**Initial Enrichment (wt.% ^{235}U)**	
	PWR Fuel	**BWR Fuel**
0,000-5,000	0.7	0.7
5,000-10,000	1.1	0.7
10,000-15,000	1.1	0.9
15,000-20,000	1.1	1.5
20,000-25,000	1.6	1.6
25,000-30,000	2.0	2.0
30,000-35,000	2.4	2.4
35,000-40,000	2.6	2.7
40,000-45,000	3.0	3.0
45,000-50,000	3.3	3.2
50,000-55,000	3.6	3.3
55,000-60,000	3.6	3.7
60,000-65,000	3.9	3.7
65,000-70,000	4.2	3.7
70,000-75,000	4.5	4.0

Notes:

1. Burnup and initial enrichments listed in this table are used in source term calculations for the shielding evaluation of the loading patterns in Figures 1.2.3 through 1.2.7 (MPC-37 and MPC-89) and uniform loading in Table 1.2.3b (MPC-32ML).
2. The burnup ranges do not overlap. Therefore, for MPC-37 and MPC-89, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc. This note does not apply to the maximum burnup of 75,000 MWD/MTU. For MPC-32ML, a lower enrichment value from a preceding burnup range is conservatively used for a transitional burnup, i.e. 20,000-25,000 MWD/MTU means 20,000.1-25,000 MWD/MTU, etc.

Table 5.2.18 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 (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)[††]	552.639
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 5.2.18 (continued) DESCRIPTION OF 16X16D DESIGN BASIS FUEL	
	PWR (MPC-32ML)
Lower End Fitting (kg)	10.795 (steel/inconel)
Gas Plenum Springs (kg)	1.474 (steel/inconel)
Gas Plenum Spacer (kg)	1.692 (steel/inconel)
Upper End Fitting (kg)	12.344 (steel/inconel)
Incore Grid Spacers (kg)	12.67 (inconel)

Table 5.2.19 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**	**45,000 MWD/MTU 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 5.2.20 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**	
(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

SCALING FACTORS USED IN CALCULATING THE 16X16D (MPC-32ML) ^{60}Co SOURCE

Region	PWR (MPC-32ML)
Upper End Fitting	0.05
Gas Plenum Spacer	0.1
Gas Plenum Springs	0.2
Incore Grid Spacer	1.0
Lower End Fitting	0.2

Table 5.2.22

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

Location	**45,000 MWD/MTU and 5-Year Cooling (curies)**	**62,500 MWD/MTU and 8-Year Cooling (curies)**
Upper End Fitting	41.46	29.76
Gas Plenum Springs	11.37	8.16
Gas Plenum Spacer	19.80	14.21
Incore Grid Spacers	851.14	610.89
Lower End Fitting	145.04	104.10

Table 5.2.23 CALCULATED 16X16D (MPC-32ML) PWR NEUTRON SOURCE PER ASSEMBLY AT SELECTED DESIGN BASIS BURNUP AND COOLING TIME COMBINATIONS FOR NORMAL AND ACCIDENT CONDITIONS			
Lower Energy (MeV)	**Upper Energy (MeV)**	**45,000 MWD/MTU 5-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

Figure 5.3.14 shows a cross sectional view of the HI-TRAC VW Version V2 with the Neutron Shield Cylinder and MPC-89, as it was modeled in MCNP. Figure 5.3.15 shows a cross sectional view of the HI-TRAC VW Version V2 with the MPC-89, in which the MPC and annulus between the MPC and HI-TRAC inner cavity are filled with water, as it was modeled in MCNP.

Calculations were performed for the HI-STORM 100 [5.2.17] to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it is acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR (in MPC-37) and BWR homogenized fuel assembly is equal to 17 times the pitch and 10 times the pitch, respectively. Homogenization results in a noticeable decrease in run time. The width of 16x16D fuel assembly in MCNP model of MPC-32ML is provided as a note under Table 5.3.1.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed below.

1. The fuel shims are not modeled because they are not needed on all fuel assembly types. However, most PWR fuel assemblies will have fuel shims. The fuel shim length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis. This is conservative since it removes steel that would provide a small amount of additional shielding.

2. The MPC basket supports are not modeled. This is conservative since it removes material that would provide a small increase in shielding.

3. The MPC cavity height, MPC height and HI-STORM FW cavity height for HI-STORM FW with MPC-32ML are calculated using the length of fuel without non-fuel hardware and/or DFC, and data provided in Table 3.2.1.

Conservatively, the zircaloy flow channels are not included in the modeling of the BWR assemblies, unless explicitly mentioned.

Also, it should be noted that all dose calculations presented in this Chapter are performed with the HI-TRAC VW (standard) model unless otherwise noted. Site specific analysis of the HI-TRAC VW should consider the specific version of the HI-TRAC VW (for example, HI-TRAC VW (standard), HI-TRAC VW Version P, HI-TRAC VW Version V, HI-TRAC VW Version V2). Additionally, the HI-TRAC VW radial lead thickness, which is a site specific feature that is maximized to the extent possible without exceeding the site crane capacity or site dimensional constraints, is also considered in site specific shielding evaluations.

$$R_{Total} = \frac{\sqrt{S^2_{Total}}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} S_i^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} (R_i \times T_i)^2}}{T_{Total}} \quad \text{(Equation 5.4.5)}$$

where,

i	=	tally component index
n	=	total number of components
T_{Total}	=	total estimated tally
T_i	=	tally i component
S^2_{Total}	=	total estimated variance
S^2_i	=	variance of the i component
R_i	=	relative error of the i component
R_{Total}	=	total estimated relative error

Note that the two-step approach outlined above allows the accurate consideration of the neutron and gamma source spectrum, and the location of the individual assemblies, since the tallies are calculated in MCNP as a function of the starting energy group and the assembly location, and then in the second step multiplied with the source strength in each group in each location. It is therefore equivalent to a one-step calculation where source terms are directly specified in the MCNP input files, except for the following approximations:

The first approximation is that fuel is modeled as fresh UO_2 fuel (rather than spent fuel) in MCNP, with an upper bound enrichment. The second approximation is related to the axial burnup profile. The profile is modeled by assigning a source probability to each of the 10 axial sections of the active region, based on a representative axial burnup profile [5.2.17]. For fuel gammas, the probability is proportional to the burnup, since the gamma source strength changes essentially linearly with burnup. For neutrons, the probability is proportional to the burnup raised to the power of 4.2, since the neutron source strength is proportional to the burnup raised to about that power [5.4.7]. This is a standard approach that has been previously used in the licensing calculations for the HI-STAR 100 cask [5.4.8] and HI-STORM 100 system [5.2.17].

Tables 5.1.5, 5.1.6 and 5.1.11 provide the design basis dose rates adjacent to the HI-STORM overpack during normal conditions for the MPC types in Table 1.0.1. Tables 5.1.7, 5.1.8 and 5.1.12 provide the design basis dose rates at one meter from the overpack. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Subsections 5.1.1 and 5.1.2.

Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-37 condition with an empty water-jacket (condition in which

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 Loading Pattern Dose Rates

The dose rates provided in Tables 5.1.11 and 5.1.12 are the maximum dose rates for HI-STORM FW with MPC-32ML for conservative loading patterns in Table 5.0.5. Table 5.4.19 and Table 5.4.20 provide adjacent and 1-m dose rates for all burnup-enrichment-cooling time combinations from Table 5.0.5.

The distance dose rates for arrays of HI-STORM FWs with MPC-32ML are provided in Table 5.4.21 for the most bounding loading pattern from Table 5.0.5.

The dose rates provided in Table 5.1.13 are the maximum dose rates for HI-TRAC VW with MPC-32ML for conservative loading patterns in Table 5.0.5. Table 5.4.22 provides adjacent and 1-m dose rates for all burnup-enrichment-cooling time combinations from Table 5.0.5.

Higher concrete density may be used in site specific shielding analysis to further lower the occupational dose rates.

5.4.7 Dose Rate Evaluation for Fuel Assemblies with Irradiated Stainless Steel Replacement Rods or BLEU Fuel

Some fuel assemblies may contain irradiated stainless steel rods or BLEU fuel material. From shielding perspective, assemblies containing Blended Low Enriched Uranium (BLEU) fuel material are essentially identical to UO_2 fuel except for the presence of a higher quantity of cobalt impurity.

Table 5.4.19

ADJACENT DOSE RATES FOR HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS (SEE TABLE 5.0.5 FOR LOADING PATTERNS)

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	15,000 MWD/MTU 3-Year Cooling	**20,000 MWD/MTU 3-Year Cooling**	**25,000 MWD/MTU 3.5-Year Cooling**	**30,000 MWD/MTU 3.6-Year Cooling**	**35,000 MWD/MTU 4-Year Cooling**	**40,000 MWD/MTU 4.5-Year Cooling**
1	219	279	260	280	271	262
2	138	173	152	163	151	139
3 (surface)	34	42	41	44	44	45
3 (overpack edge)	66	77	74	77	75	75
4 (center)	1	1	1	1	1	1
4 (mid)	4	5	5	6	6	6
4 (outer)	36	43	41	43	42	42

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.

Table 5.4.19 (continued)

ADJACENT DOSE RATES FOR HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS (SEE TABLE 5.0.5 FOR LOADING PATTERNS)

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	45,000 MWD/MTU 5-Year Cooling	**50,000 MWD/MTU 6-Year Cooling**	**55,000 MWD/MTU 7-Year Cooling**	**60,000 MWD/MTU 9-Year Cooling**	**65,000 MWD/MTU 11-Year Cooling**	**70,000 MWD/MTU 13-Year Cooling**
1	250	219	197	162	136	118
2	129	109	97	80	70	63
3 (surface)	44	41	40	36	33	31
3 (overpack edge)	73	69	65	58	52	47
4 (center)	1	1	2	2	2	2
4 (mid)	6	5	5	5	4	4
4 (outer)	41	38	36	32	28	26

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.

Table 5.4.20

1-METER DOSE RATES FOR HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS (SEE TABLE 5.0.5 FOR LOADING PATTERNS)

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	15,000 MWD/MTU 3-Year Cooling	**20,000 MWD/MTU 3-Year Cooling**	**25,000 MWD/MTU 3.5-Year Cooling**	**30,000 MWD/MTU 3.6-Year Cooling**	**35,000 MWD/MTU 4-Year Cooling**	**40,000 MWD/MTU 4.5-Year Cooling**
1	53	67	60	64	60	57
2	74	93	81	87	81	74
3	17	20	19	20	20	19
4 (center)	2	2	2	2	2	2

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.

Table 5.4.20 (continued)

1-METER DOSE RATES FOR HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS (SEE TABLE 5.0.5 FOR LOADING PATTERNS)

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	45,000 MWD/MTU 5-Year Cooling	**50,000 MWD/MTU 6-Year Cooling**	**55,000 MWD/MTU 7-Year Cooling**	**60,000 MWD/MTU 9-Year Cooling**	**65,000 MWD/MTU 11-Year Cooling**	**70,000 MWD/MTU 13-Year Cooling**
1	53	46	41	34	28	25
2	69	58	51	42	36	33
3	19	17	16	14	13	12
4 (center)	2	2	2	2	2	2

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.

Table 5.4.21

MAXIMUM DOSE RATES FOR ARRAYS OF HI-STORM FWs WITH MPC-32ML LOADING PATTERNS (SEE TABLE 5.0.5)

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

Table 5.4.22
DOSE RATES FOR HI-TRAC VW WITH MPC-32ML WITH SELECTED 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS

Dose Point Location	Totals + BPRA (mrem/hr)					
	15,000 MWD/MTU 3-Year Cooling	**20,000 MWD/MTU 3-Year Cooling**	**25,000 MWD/MTU 3.5-Year Cooling**	**30,000 MWD/MTU 3.6-Year Cooling**	**35,000 MWD/MTU 4-Year Cooling**	**40,000 MWD/MTU 4.5-Year Cooling**
ADJACENT TO THE HI-TRAC VW						
1	61	82	83	92	99	110
2	1306	1666	1518	1654	1597	1553
3	516	598	585	606	602	604
4	641	757	751	795	810	846
5	1850	2389	2354	2539	2591	2719
ONE METER FROM THE HI-TRAC VW						
1	280	355	333	356	348	344
2	587	746	675	732	703	679
3	164	198	187	197	193	191
4	472	574	573	616	624	634
5	957	1223	1197	1278	1289	1329

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.

Table 5.4.22 (continued)
DOSE RATES FOR HI-TRAC VW WITH MPC-32ML WITH SELECTED 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	45,000 MWD/MTU 5-Year Cooling	**50,000 MWD/MTU 6-Year Cooling**	**55,000 MWD/MTU 7-Year Cooling**	**60,000 MWD/MTU 9-Year Cooling**	**65,000 MWD/MTU 11-Year Cooling**	**70,000 MWD/MTU 13-Year Cooling**
ADJACENT TO THE HI-TRAC VW						
1	117	121	127	136	135	135
2	1504	1361	1292	1185	1105	1058
3	591	559	531	481	433	395
4	858	852	854	852	821	800
5	2734	2649	2590	2470	2259	2103
ONE METER FROM THE HI-TRAC VW						
1	333	303	283	251	221	199
2	653	587	553	501	464	441
3	186	172	163	149	136	127
4	628	590	560	502	453	418
5	1316	1246	1188	1082	952	853

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.

8.2.2 Nonstructural Materials

i. Aluminum Alloy

The space between the fuel basket and the inside surface of the Confinement Boundary is occupied by specially shaped precision extruded or machined basket shims made of a high strength and creep resistant aluminum alloy. The basket shims establish a conformal contact interface with the fuel basket and the MPC shell, and thus prevent significant movement of the basket. The basket shims are extruded and/or machined to a precise shape with a high degree of accuracy.

The clearance between the basket shims and the interfacing machined surface of the MPC cavity is set to be sufficiently small such that the thermal expansion of the parts inside the MPC under Design Basis heat load conditions will minimize any macro-gaps at the interface and thus minimize any resistance to the outward flow of heat, while ensuring that there is no restraint of free thermal expansion.

To further enhance thermal performance, the aluminum alloy basket shims are hard anodized. This provides for added corrosion protection and to achieve the emissivity value specified in Section 4.2. Mechanical properties of the shim material are provided in Section 3.3.

The basket shim material utilized in the HI-STORM FW system has also been used in other casks (viz. HI-STAR 180).

ii. Concrete

The plain concrete between the overpack inner and outer steel shells and in the overpack lid is specified to provide the necessary shielding properties and compressive strength. Table 1.2.5 in this FSAR and Appendix 1.D of the HI-STORM 100 FSAR which provide technical and placement requirements on plain concrete are also invoked for HI-STORM FW concrete.

The HI-STORM FW overpack concrete is enclosed in steel inner and outer shells connected to each other by radial ribs, and top and bottom plates and does not require rebar. As the HI-STORM FW overpack concrete is not reinforced, the structural analysis of the overpack only credits the compressive strength of the concrete.

The technical requirements on testing and qualification of the HI-STORM FW plain concrete are identical to those used in the HI-STORM 100 program. Accordingly, the testing and placement guidelines in Appendix 1.D of the HI-STORM 100 FSAR (Docket No. 72-1014), is incorporated in this SAR by reference.

ACI 318 is the reference code for the plain concrete in the HI-STORM FW overpack. ACI 318.1-85(05) is the applicable code utilized to determine the allowable compressive strength of the plain concrete credited in structural analysis.

8.8 GAMMA AND NEUTRON SHIELDING MATERIALS

Gamma and neutron shield materials in the HI-STORM FW System are discussed in Section 1.2. The primary shielding materials used in the HI-STORM FW system, like the HI-STORM 100 system, are plain concrete, steel, lead, and water.

The plain concrete enclosed by cylindrical steel shells, a thick steel baseplate, and a top annular plate provides the main shielding function in the HI-STORM FW overpack. The overpack lid has appropriate concrete shielding to provide neutron and gamma attenuation to minimize skyshine.

The transfer cask in the HI-STORM FW system (HI-TRAC VW) is provided with steel and lead shielding to ensure that the radiation and exposure objectives of 10CFR72.104 and 10CFR72.106 are met. The space between the inner shell and the middle shell is occupied by lead, conforming to ASTM B29, which provides the bulk of the cask's (gamma) radiation shielding capability. The water jacket between the middle shell and the outermost shell (filled with demineralized water or ethylene glycol fortified water, depending on the site environmental constraints) provides most of the neutron shielding capability to the cask. The water in the water jacket serves as the neutron shield on demand: When the cask is in the pool and the MPC is full of water, the water jacket is kept empty (or partially empty as necessary) to minimize the cask's weight, the neutron shielding function being provided by the water in the MPC cavity. However, when the MPC is emptied of water at the Decontamination and Assembly Station (DAS), then the neutron shielding capacity of the cask is replenished by filling the water jacket. The HI-TRAC VW bottom lid is extra thick steel to provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC.

8.8.1 Concrete

Table 1.2.5 of this FSAR and Appendix 1.D of HI-STORM 100 FSAR provide details of the concrete properties and the testing requirements. The *critical characteristics* of concrete are its density and compressive strength.

The density of plain concrete within the HI-STORM FW overpack is subject to a minor decrease due to long-term exposure to elevated temperatures. The reduction in density occurs primarily due to liberation of unbonded water by evaporation.

The density of concrete has been classified into three states in the published literature [8.8.1].

a) fresh density: the density of freshly mixed concrete
b) air-dry density: drying in air under ambient conditions, where moisture is lost until a quasi-equilibrium is reached
c) oven-dry density: concrete dried in an oven at 105ºC (221ºF)

Because the bulk temperature of concrete in HI-STORM FW is spatially variable, the oven-dry density is conservatively used as the reference density for shielding analysis.

Density loss during the initial drying process is considered in the fabrication of the HI-STORM FW overpack by providing wet concrete densities above the minimum required dry (hardened paste) density. Density loss during drying is on the order of 1% and conservatively imposes a larger delta between wet density and the minimum dry density. The data in the literature, viz., Neville [8.8.1] indicates that the density difference between the air-dry condition and oven-dry condition is about one fourth of the density difference experienced during the drying process. Therefore, the loss in density would be expected to be on the order of 0.25%. This density loss is very low and is considered too small to have a significant impact on the shielding performance of the overpack. Thus, the minimum "fresh density" during concrete placement is set equal to the reference density (Table 1.2.5) plus 1.25%.

Section 5.3 considers the minimum density requirements of concrete for effective shielding. The density requirement is confirmed per Table 1.2.5 of this FSAR and Appendix 1.D of the HI-STORM 100 FSAR.

8.8.2 Steel

Section 5.3 provides a discussion on steel as a shielding material and its composition used in the evaluation of its shielding characteristics.

8.8.3 Lead

Section 1.2 provides a discussion on lead used in HI-TRAC VW for gamma shielding. In the HI-TRAC VW transfer cask radial direction, gamma and neutron shielding consists of steel-lead-steel and water, respectively. In the HI-TRAC VW bottom lid, layers of steel-lead-steel provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC.

Mechanical properties of lead are provided in Section 3.3. Section 5.3 provides the minimum density and composition (mass fraction of trace elements) of lead.

8.8.4 Water

Water is used as a neutron shield in the HI-TRAC VW transfer cask. Section 5.3 provides the minimum density requirements of water for transfer cask water jacket and inside MPC. The shielding effectiveness is calculated based on the minimum water density at the highest operating temperature. Calculations show that additives for freeze protection (at low temperature operation) such as ethylene glycol do not have any adverse effect on effectiveness of the neutron shielding function of water in the water jacket.

As discussed in Section 5.1, there is only one accident that has any significant impact on the shielding configuration. This accident is the postulated loss of the neutron shield (water) in the HI-TRAC VW. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by air.

11. Install the inflatable annulus seal around the MPC.

12. To the extent practicable, apply waterproof tape over any empty bolt holes or locations where water may create a decontamination issue.

Note:
Canister filling and draining operations vary by site. Instructions are provided on a site-specific basis.

13. Fill the MPC with water to approximately 12 inches below the top of the MPC shell. Refer to LCO 3.3.1 for boron concentration requirements.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

14. Place HI-TRAC VW in the designated cask loading area.

15. If used, the DFC can be installed in those cells where damaged fuel or fuel debris will be stored. If used, the bottom DFI can be installed in those cells where damaged fuel which can be handled by normal means and meeting the criteria of section 2.1.3.1, will be stored.

16. Verify spent fuel pool for boron concentration requirements in accordance with LCO 3.3.1. Testing must be completed within four hours prior to loading and every 48 hours after in accordance with the LCO. Two independent measurements shall be taken to ensure that the requirement of 10 CFR 72.124(a) is met.

9.2.3 MPC Fuel Loading

Note:
When loading an MPC requiring soluble boron, the boron concentration of the water shall be checked in accordance with LCO 3.3.1 before and during operations with fuel and water in the MPC.

1. Ensure that only fuel assemblies that meet all the applicable conditions for loading, as specified in the Approved Contents Section of Appendix B to the CoC, have been selected for loading into the MPC. Given the complexity of some of the approved loading configurations, caution must be taken in developing and verifying a plan for the fuel to be loaded. The selection of fuel to be loaded may include (but may not be limited to) the following considerations:
   a. Assemblies must meet the dimensional requirements for the applicable array/class.
   b. If fuel is to be loaded using one of the loading patterns where requirements differ between basket cells, the pattern should be identified, and each

assembly should be shown to meet the applicable requirements for the designated basket cell. This may include:

i. Assembly decay heat (including any decay heat contribution of non-fuel-hardware located in the assembly, and any adjustments for fuel lengths) meets the decay heat limit of the cell
ii. Assembly cooling time meets the cooling time limit established for the specific assembly and/or cell location. The specific cooling time limit for the assembly and cell may be calculated based on the assembly burnup and/or the decay heat limit of the cell, using the equation and appropriate coefficients in Appendix B of the CoC. Regardless of the result of the equation, assemblies must meet the minimum cooling time requirements.

2. Ensure assemblies are characterized according to their condition, and that damaged fuel or fuel debris is either loaded into damaged fuel containers (DFCs), or, only for damaged fuel that can be handled by normal means, loaded in basket cells with DFI assemblies at the top and bottom of the cell.
3. Load the pre-selected fuel assemblies into the MPC in accordance with the approved loading plan.
4. Perform a post-loading visual verification of the assembly identification to confirm the serial numbers match the approved loading plan
5. If required, install fuel shims and/or DFI top caps where necessary in the cells.

9.2.4 MPC Closure

1. Install MPC lid and remove the HI-TRAC VW from the spent fuel pool as follows:

Rig the MPC lid for installation in the MPC in accordance with site-approved rigging procedures.

Install the drain line to the underside of the MPC lid.

Align the MPC lid and lift yoke so the drain line will be positioned in the MPC for installation.

Seat the MPC lid in the MPC and visually verify that the lid is properly installed.

Record the time to begin the time-to-boil monitoring, if necessary.

Engage the lift yoke to HI-TRAC VW.

Code, Section III, Subsection NB, Article NB-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable.

If a leak is discovered, the test pressure shall be reduced, the MPC cavity water level lowered, if applicable, the MPC cavity vented, and the weld shall be examined to determine the cause of the leakage and/or cracking. Repairs to the weld shall be performed in accordance with written and approved procedures prepared in accordance with the ASME Code, Section III, Article NB-4450.

The MPC confinement boundary pressure test shall be repeated until all required examinations are found to be acceptable. Test results shall be documented and maintained as part of the loaded MPC quality documentation package.

10.1.3 Materials Testing

The majority of materials used in the HI-TRAC transfer cask and a portion of the material in the HI-STORM overpack are ferritic steels. ASME Code, Section II and Section III require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures. Certain versions of the HI-TRAC include Holtite neutron shielding material.

Materials of the HI-TRAC transfer cask and HI-STORM overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section IIA and/or ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The materials to be tested are identified in Table 3.1.9 and applicable weld materials. Table 3.1.9 provides the test temperatures and test acceptance criteria to be used when performing the material testing specified above.

For Holtite neutron shielding material, each manufactured lot of material shall be tested to verify the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) meet the requirements specified in Table 1.2.5. Appendix 1.B of HI-STORM 100 System FSAR [1.1.3] provides the Holtite-A material properties germane to its function as a neutron shield. A manufactured lot is defined as the total amount of material used to make any number of mixed batches comprised of constituent ingredients from the same lot/batch identification numbers supplied by the constituent manufacturer. Testing shall be performed in accordance with written and approved procedures and/or standards. Material composition, boron concentration, and density (or specific gravity) data for each manufactured lot of neutron shield material shall become part of the quality documentation package. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps from occurring in the material. Samples of each manufactured lot of neutron shield material shall be maintained by Holtec International as part of the quality record documentation package.

The concrete utilized in the construction of the HI-STORM overpack shall be mixed, poured, and tested as set down in Chapter 1.D of the HI-STORM 100 FSAR (Docket 72-1014) [10.1.6] in accordance with written and approved procedures. Testing shall verify the compressive strength