SAFETY ASSESSMENT OF OPG's USED FUEL DRY STORAGE FACILITIES

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Abstract

Safety assessments of OPG's used fuel dry storage facilities and the associated systems and operations are required to support requests for regulatory approvals to construct and operate those facilities. This paper describes the key methodology and assumptions associated with normal operation and with malfunctions and accident conditions used to perform those safety assessments. A summary of the results obtained from the safety assessment of the Darlington Used Fuel Dry Storage Facility is also provided.

1.0 INTRODUCTION

Ontario Power Generation (OPG) has been operating the Pickering Waste Management Facility (PWMF) and Western Waste Management Facility (WWMF) since 1996 and 2003 respectively. Ten-year cooled or older used fuel contained in seal-welded Dry Storage Containers (DSCs) is stored in these facilities. The construction licence for the Darlington Used Fuel Dry Storage Facility (DUFDSF) was obtained in August 2004.

Safety assessment of each used fuel dry storage facility and the associated systems and operations is required to support requests for regulatory approvals to construct and operate the facility. The objective of the safety assessment is to assess the dose consequences to the public and the workers under normal operation and postulated credible accident scenarios and to confirm that dose rates are below the regulatory dose limits for the workers and members of the public.

For illustrative purposes, the results of the DUFDSF safety assessment are presented here.

2.0 ASSESSMENT BASIS

The following sections describe the assumptions used in performing the safety assessment of the facility during normal operation and during malfunctions and accident conditions.

Each safety assessment is generally supported by a number of analyses and assessments such as shielding analysis of the container and the facility, criticality assessment, thermal analysis/assessment, tornado assessment and earthquake assessment which are based on the assumptions described below.

2.1 REFERENCE FUEL BUNDLE

As the basis for the safety assessment of a particular facility, a reference used fuel bundle is defined based on the operating history and data on fuel discharged from the reactors of the specific nuclear generating station.

The primary factors that determine the characteristic of the used fuel are the physical attributes, power and burnup histories, and decay time. These factors are in turn influenced by fuelling strategies and reactor conditions. Therefore for the purpose of performing the safety assessment of a given facility and processes, a reference fuel bundle is defined. The reference used fuel bundle is defined based on the burn-up histograms of used fuel produced for 5-years such that it would conservatively represent the used fuel bundles discharged from the reactors at a given nuclear station. Thus the safety assessment based on the defined reference fuel bundle generates conservative results. The reference fuel bundle is also assumed to have been out of the reactor core for 10 years. Only fuel which has been in the fuel bays for 10 years or longer is transferred to dry storage.

For example in the case of the Darlington Used Fuel Dry Storage Facility (DUFDSF) the reference used fuel bundle was conservatively chosen to have the dimensions of the long bundles used at the Darlington reactors although presently only 30% of the bundles discharged from these reactors are long bundles. The burnup for the reference fuel bundle was chosen to be 240 MWh/kgU which is the 95th percentile of the discharged bundles burnup.

2.2 RADIONUCLIDE INVENTORY

Once the reference fuel bundle is defined, the associated radionuclide inventory is calculated using the computer code ORIGEN-S (ORNL95). The code also allows us to obtain the gamma spectrum for a reference fuel bundle which is used for the shielding analysis. The energy produced by radioactive decay is released from the bundle in the form of heat and radiation which is used for the thermal analysis.

2.3 DECAY HEAT

The energy produced by radioactive decay is released from the fuel bundle in the form of heat and radiation. The decay heat for different cooling periods is calculated using the ORIGEN-S computer code based on the reference fuel bundle.

2.4 FREE INVENTORIES AVAILABLE FOR RELEASE

In the event that a ten year or older used fuel bundle should become damaged during the dry storage operations, the only significant radionuclides species available for release are krypton-85 and tritium. For a fuel element damaged under abnormal operating conditions, it is postulated that the free inventory of tritium and krypton-85 in the gap between the fuel matrix and the zircaloy sheath, plus 10% of the inventory in the grain boundary would be released. The gap fraction is assumed to be 0.0365 for tritium and for krypton-85. The grain-boundary fraction is assumed to be 0.1209 for tritium and for krypton-85 (OPG01a). Conservatively, for the assessment of airborne emissions, the calculation of tritium and krypton-85 inventories takes into account the activation in the reactor core of small quantities of impurities (OPG00a) present in the used fuel.

2.5 ATMOSPHERIC DILUTION FACTORS

The Atmospheric Dilution Factors (ADFs) are direct multipliers in the dose calculations and are important in the prediction of off-site dose consequences to the public arising from airborne releases of radioactivity at the dry storage facility.

The ADFs are a highly simplified representation of many complex phenomena. The ADFs used for safety assessments have been determined using an up-to-date methodology (OPG02g). The most recent meteorological data measured on-site are used to derive the ADFs. Conservatively, the ADFs calculated for the releases lasting up to 1 hour were used when calculating the off-site dose consequences to the public in the event of a credible postulated malfunction or accident.

The calculated ADFs are grouped into three distinct periods:

a) Short Term: for the first hour period, defined as the 90th percentile value of the cumulative frequency distribution of the calculated ADFs. In the case of the DUFDSF this ADF was calculated to be 1.29×10^{-4} s/m^3

b) Prolonged Term: for the next 24 hour period, defined as the 90th percentile value of cumulative frequency distribution of the calculated ADFs for the worst wind sector, based on 24 consecutive hours of meteorological measurements. In the case of the DUFDSF this ADF was calculated to be 1.27×10^{-5} s/m^3.

c) Long Term: for more than 24 hour period, takes into account joint frequency of wind speed, stability class and wind direction over the time period of interest; they are calculated using the model suggested in Canadian Standards Association standard CSA N288.2 (CSA91). In the case of the DUFDSF this ADF was calculated to be 3.15×10^{-6} s/m^3.

2.6 BREATHING RATES

Breathing rates used are 22.2 m^3/day (2.57×10^{-4} m^3/s) for an adult and 5.16 m^3/day (5.97×10^{-5} m^3/s) for an infant (HC99).

The breathing rate used for a Nuclear Energy Worker, based on an adult performing light exercise, is 1.5 m^3/h (4.27×10^{-4} m^3/s) (ICRP95).

2.7 DOSE CONVERSION FACTORS

The dose conversion factors for tritium, used for this assessment, are 2.0×10^{-11} Sv/Bq for an adult and 5.8×10^{-11} Sv/Bq for an infant (AECL83). The immersion skin absorption factor for tritium oxide in air is taken as 2 (CSA87). The dose conversion factors for krypton-85, used for this assessment, are 2.55×10^{-16} Sv.s^{-1}.Bq^{-1}.m^3 (8.06×10^{-9} Sv.a^{-1}.Bq^{-1}.m^3) for an adult and 1.5 times this value for an infant (HC99).

2.8 DRY STORAGE CONTAINER

The Dry Storage Container (DSC) is a double-shell rectangular container, with outside dimensions of 2.121 x 2.419 m by 3.557 m in height (including the lid), and an inside cavity of 1.046 x 1.322 m by 2.496 m in height. The thickness of each carbon-steel shell is 13 mm. The DSC walls are made of 52 cm (nominal thickness) concrete. The reinforced high-density concrete provides radiation shielding while maintaining adequate used fuel decay heat dissipation. The concrete has a density in the range of 3.5 to 3.7 Mg/m^3 and a full strength of 40 MPa. The maximum total mass (including the lid) is approximately 60 Mg when empty and approximately 70 Mg when loaded with four used fuel storage modules (384 used fuel bundles).

The concrete used as shield in the DSC design is reinforced heavy concrete. The minimum specified concrete density is 3.5 g/cc, but the presence of the rebars results in a homogenized concrete density of 3.57 g/cc. Although in practice the density of the concrete mixed used to manufacture each DSC is higher than the minimum specified density, for safety assessment purposes it is conservatively assumed to be 3.57 g/cc.

Each DSC has the capacity to store 384 used fuel bundles. For the safety assessment, it is assumed that all the bundles have the same properties and characteristics as the reference used fuel bundle.

2.9 USED FUEL DRY STORAGE FACILITY

In general, OPG's used fuel dry storage facilities consist of a processing building and a specified number of storage buildings. To assess the dose consequences, the facility is assumed to be completely full with DSCs, with each DSC containing the maximum load of 384 reference fuel bundles.

3.0 ACCEPTANCE CRITERIA

The radiation safety requirements under normal operation for a used fuel dry storage facility are the following:

- The dose rate target for the public at the station site boundary shall be less than the OPG target of 10 µSv/y, which is 1% of the regulatory limit of 1 mSv/y for members of the public, based on full occupancy.
- The dose rate limit at the used fuel dry storage facility fence shall be 0.5 µSv/h, based on the regulatory limit of 1 mSv/y for members of the public and 2000 hours work per year by non Nuclear Energy Workers (non-NEWs)
- The regulatory dose rate limit for NEWs is 50 mSv/y in any single year, and 100 mSv over five years.

The radiation requirements considered under an abnormal event or accident are the following:

- The dose target for the public at or beyond the OPG property boundary due to an abnormal event or accident shall be 1 mSv.

- The dose target for a worker due to an abnormal event or accident shall be 50 mSv.

4.0 NORMAL OPERATION ASSESSMENT METHODOLOGY

The computer code ORIGEN-S (ORNL95) together with a burnup dependent library for natural uranium oxide fuelled CANDU reactor 37-element bundle design (GAULD95a) was used to determine the radioactivity content of the irradiated fuel. After calculating the buildup of actinides and fission products during a specified irradiation period, the code then calculates their inventory as a function of the time after discharge from the reactor (i.e. cooling time). The photon and neutron spectrum as a function of the cooling time is also calculated by the code. The breakdown for the gamma and neutron energy groups corresponds to that of BUGLE-80 Multigroup Cross Section Library (ORNL92). Some verification and validation of the ORIGEN-S code and the Nuclear Data Libraries has been performed by AECL under the CANDU Owners Group (COG) Working Party 25 (GAULD95b).

The gamma shielding calculations for a single DSC and for inside and outside the dry storage facility were carried out using the point-kernel integration code QAD-CGGP-A 96.1 (ORNL97). This code simulates the transport of gamma rays originating in a volume-distributed source, travelling through shielding materials and arriving at selected detector points. Both the source and shielding materials can be described in three dimensions. Dose rate or energy deposition buildup factors can be selected based on correlation within a given problem. The gamma attenuation coefficients and buildup factors used by QAD-CGGP-A were obtained from ANS-6.4.3 Standard (ORNL92b) compared to the previous version QAD-CG which used many unpublished factors. The code QAD-CGGP-A uses an improved attenuation coefficient interpolation scheme over a wider range of shielding materials. Double precision combinatorial geometry routines have also been implemented.

The computer code Microskyshine (GROVE87) was used to calculate the contribution to the total dose rates outside the used fuel dry storage facility from gamma skyshine.

5.0 MALFUNCTIONS AND ACCIDENTS ASSESSMENT METHODOLOGY

The safety assessment is divided into the following main stages of operations outside the station:

(a) On-site transfer

(b) Operations inside the processing building

(c) Storage

5.1 IDENTIFICATION OF INITIATING EVENTS

For each stage of the used fuel dry storage operations, release of radiation can occur due to the failure of the systems and components used during the used fuel dry storage operations.

There are two general categories of initiating events resulting in abnormal conditions or accidents: internal events and external events.

Internal events are abnormal conditions generated within the area of operation as a result of equipment failure or human error.

External events are natural and man-made phenomena originating outside the area of operation that have the potential for leading to multiple internal events.

The list of initiating events considered for the safety assessment is consistent with the list of initiating events typically used for the nuclear power plant safety assessments.

5.2 EVENT FREQUENCY

Where possible, a frequency of occurrence is estimated for each postulated initiating event. An event is considered to be not credible if the frequency of occurrence is less than 10^{-7} events per year.

Estimation of Internal Event Frequency

The initiating events frequency estimates have been derived from experience at Ontario Power Generation (OPG) stations (OPG02h). The primary source of the data has been the computerized Significant Event Report (SER) system, supplemented by additional information, such as station quarterly reports. Where no occurrences of a particular event were found, alternative sources of information were sought to obtain a realistic frequency estimate.

5.3 SCREENING OF EVENTS

Each event was screened to identify any radiological impact to the public, the workers and/or the environment. Design provisions and procedural measures that could prevent the event or mitigate its consequences were also considered.

5.4 DOSE CONSEQUENCES

5.4.1 Public

The potential doses to an adult and an infant, from airborne tritium and krypton-85 emissions, are estimated for the inhalation and immersion pathways. For acute tritium exposure, only the inhalation and skin absorption pathway need to be considered. For krypton-85 exposure, only the immersion (in air) pathway is significant.
Long-term ADFs for chronic releases (> 24 hours) are used for assessment of emissions from potential fuel element failures during DSC handling under normal operating conditions, and from postulated containment failures during DSC storage under abnormal operating conditions. Chronic releases are assumed to occur at a constant rate over the year.

Short-term ADFs for acute releases are used for assessment of emissions from failure of the transporter under abnormal operating conditions, and from potential fuel element and DSC containment failure resulting from accident conditions. Acute releases are assumed to occur immediately, and are reported as an estimated dose per event.

The equations and data given below were used to calculate public dose consequences.

The dose from exposure to tritium is given by the following equation

$$D(^3H) = N_e * R_e(^3H) * P_{01} * BR * sk_a * DCF_i(^3H)$$

where

$D(^3H)$ = dose from exposure to tritium (Sv)

N_e = number of failed fuel elements

$R_e(^3H)$ = tritium release per failed used fuel element (Bq/element)

P_{01} = atmospheric dilution factor (s/m^3) = 1.29 x10^{-4}

BR = breathing rate (m^3/s) = 2.57 x10^{-4} (for adult) and 5.97 x 10^{-5} (for infant)

sk_a = skin absorption factor of tritiated water in air = 2

$DCF_i(^3H)$ = dose conversion factor for inhalation of HTO (Sv/Bq) =

= 2 x10^{-11} (for adult) and 5.8 x10^{-11} (for infant)

and

$$R_e(^3H) = (f_{gap} + 0.1 f_{gb}) * I_e(^3H)$$

f_{gap} = fraction of radionuclide inventory in the gap = 0.0367

f_{gb} = fraction of radionuclide inventory in the grain boundary = 0.1209

$I_e(^3H)$ = tritium inventory per used fuel element (Bq/element) = 4.92 x 10^9 Bq

The dose from krypton exposure is calculated using the following equation:

$$D(^{85}Kr) = N_e * R_e(^{85}Kr) * P_{01} * DCF_i(^{85}Kr) / N_s$$

where

$D(^{85}Kr)$ = Dose from exposure to ^{85}Kr (Sv)

N_e = Number of failed fuel elements

$R_e(^{85}Kr)$ = ^{85}Kr released per failed fuel element (Bq/element)

$DCF_i(^{85}Kr)$ = Dose conversion factor for ^{85}Kr immersion in air (Sv/yr)/(Bq/m^3) =

= 8.06 x 10^{-9} (for adult) and 1.21 x 10^{-8} (for infant)

P_{01} = Atmospheric dilution factor = 1.29 x10^{-4}

N_s = Number of seconds in a year = 3.15 x 10^7

and

$R_e(^{85}Kr) = (f_{gap} + 0.1 f_{gb}) * I_e(^{85}Kr)$

f_{gap} = fraction of radionuclide inventory in the gap = 0.0367

f_{gb} = fraction of radionuclide inventory in the grain boundary = 0.1209

$I_e(^{85}Kr)$ = krypton inventory per used fuel element (Bq/element) = 2.73 x 10^{10} Bq

The total dose to the public is the sum of the doses due to tritium and krypton exposure given above.

5.4.2 Occupational

A worker is assumed to be present in the vicinity of the accident location wearing no protective clothing or respiratory protection at the time of the accident. The worker's response time to remove himself or herself from the accident location (i.e., under emergency back-out conditions) is assumed to be 2 minutes.

The resulting dose rate is assessed using the semi-infinite cloud model. The cloud volume is assumed to be 500 m^3 and the exposure time 120 seconds.

The equations and data given below were used to calculate occupational dose consequences.

The dose from exposure to tritium is given by the following equation:

$$D(^3H) = N_e * R_e(^3H) * T * BR * sk_a * DCF_i(^3H) / V$$

where

$D(^3H)$ = dose from exposure to tritium (Sv)

N_e = number of failed fuel elements

$R_e(^3H)$ = tritium release per failed used fuel element (Bq/element)

T = exposure time in seconds = 120

BR = NEW adult breathing rate (m^3/s) = 4.17 x10^{-4}

sk_a = skin absorption factor of tritiated water in air = 2

$DCF_i(^3H)$ = dose conversion factor for inhalation of HTO (Sv/Bq) = 2 x10^{-11}

V = contaminated cloud volume (m^3) = 500

and

$R_e(^3H) = (f_{gap} + 0.1 f_{gb}) * I_e(^3H)$

f_{gap} = fraction of radionuclide inventory in the gap = 0.0367

f_{gb} = fraction of radionuclide inventory in the grain boundary = 0.1209

$I_e(^3H)$ = tritium inventory per used fuel element (Bq/element) = 4.92 x 10^9 Bq

The dose from krypton exposure is calculated using the following equation:

$$D(^{85}Kr) = N_e * R_e(^{85}Kr) * T * DCF_i(^{85}Kr)/(N_s \times V)$$

where

$D(^{85}Kr)$ = Dose from exposure to ^{85}Kr (Sv)

N_e = Number of failed fuel elements

$R_e(^{85}Kr)$ = ^{85}Kr released per failed used fuel element (Bq/element)

$DCF_i(^{85}Kr)$ = Dose conversion factor for ^{85}Kr immersion in air (Sv/yr)/(Bq/m^3) =

= 8.06 x 10^{-9} (B-HC99)

T = exposure time in seconds = 120

N_s = Number of seconds in a year = 3.15 x 10^7

V = contaminated cloud volume (m^3) = 500

and

$$R_e(^{85}Kr) = (f_{gap} + 0.1 f_{gb}) * I_e(^{85}Kr)$$

f_{gap} = fraction of radionuclide inventory in the gap = 0.0367

f_{gb} = fraction of radionuclide inventory in the grain boundary = 0.1209

$I_e(^{85}Kr)$ = krypton inventory per used fuel element (Bq/element) = 2.73 x 10^{10} Bq

The total dose to the worker would be the sum of the doses due to tritium and krypton exposure calculated as above.

6.0 RESULTS

The results presented in this section are those obtained from the safety assessment of the DUFDSF which was approved for construction by the CNSC last year.

6.1 ASSESSMENT UNDER NORMAL OPERATING CONDITIONS

Under normal operating conditions the safety assessment takes into consideration the different stages of the dry storage operation: transfer of the DSCs from the fuel bay to the dry storage facility, processing of the DSCs, and storage of the DSCs.

The direct radiation fields at each stage of the dry storage operations have been calculated to ensure that the acceptance criteria are met. Figure 1 shows the dose rates from a single DSC filled with 384 DUFDSF reference fuel bundles. Figure 2 gives the dose rate versus distance from the facility wall. The facility was assumed to have 1500 DSCs filled with 10 year old DUFDSF reference fuel bundles. The skyshine contribution to these dose rates was also calculated. Figure 3 shows the dose rates inside the storage buildings assuming the buildings are full. The contribution from roofshine was taken into account.

Calculated dose rates estimates have been demonstrated to be conservative compared with actual dose rates measured during PWMF and WWMF operations. For DSCs loaded with 10-year cooled or older fuel, measured contact dose rates to date are about 9 to 13 µSv/h. This compares with estimates of 67 µSv/h contact does rates for 10-year cooled fuel (at the DSC side or front). At a 1-m distance, measured dose rates are about 5 to 7 µSv/h, compared with calculated dose rate estimates of 48 to 56 µSv/h. Dose rates inside the PWMF Processing and Storage Buildings have been demonstrated to be acceptably low (most working areas are normally at or near ambient radiation levels).

Figure 3 shows the thermal power per reference fuel bundle as a function of the fuel age. A thermal analysis of a DSC loaded with 37-element fuel bundles with a decay heat of 6.8 W per bundle was carried out. The ambient temperature of 38°C in the proximity of DSCs was assumed. The maximum internal and external DSC surface temperatures were calculated to be 78°C and 47°C, respectively. The stresses generated in the concrete by this thermal gradient have been assessed to result in no significant cracking of the concrete over a storage period of 50 years. Ten-year cooled Darlington reference used fuel has a lower heat load of 5.86 W/bundle (see Figure 3), therefore the thermal analysis is considered to envelope the conditions for storage of Darlington used fuel in DSCs.

A thermal analysis of the DSC storage building was carried out. The results show that the storage building design is adequate to maintain temperatures below 38°C in the proximity of the DSCs given the buoyant air movement within the facility. Temperatures inside the storage building were found to be well below the design temperatures in most areas.

Under normal operating conditions, no airborne emissions are expected from DSCs during transfer from the Darlington fuel bays to the DUFDSF. This is because the uranium dioxide matrix, the used fuel sheath and the transfer elastomeric seal provide multiple barriers towards preventing the release of radioactive materials.

Because the DSCs are unclamped and unwelded at some stages during processing such as welding and vacuum drying, there is a small potential for airborne emissions resulting from DSC processing operations. Therefore an assessment of the chronic radioactive emission during processing was carried out. Conservatively, it was assumed that one fuel element in 1% of fuel bundles is damaged during handling (4 elements per DSC), and for each failed fuel element, the free inventory of tritium and krypton-85 is released into the DSC cavity. OPG fuel performance experience has demonstrated that cladding defects are present in less than 0.02% of fuel bundles (representing < 0.001% of fuel elements). The barrier provided by the DSC lid is ignored and these radionuclides are released into the environment. Since each DSC has the capacity to hold 384 bundles and the facility will process about 70 containers per year, it is postulated that a total of 280 fuel elements fail during one year under normal operating conditions (a very conservative assumption). The chronic off-site dose consequences from this scenario, for a member of the public at the Darlington NGS site boundary, are estimated to be about 0.001% of the CNSC regulatory dose limit of 1 mSv/year.

There are no mechanisms for airborne releases to occur under normal operating conditions during storage of seal-welded DSCs.

Operating experience has shown that emissions have remained below the regulatory limits and the facilities routinely operate contamination free.

6.2 ASSESSMENT UNDER MALFUNCTIONS AND ACCIDENTS CONDITIONS

The results from this assessment are shown in Table 1, Table 2, and Table 3 for transfer, processing and storage of the DSCs respectively.

A bounding accident during loading, transfer and processing of the DSCs was postulated assuming the failure of 30% of a DSC's used fuel content i.e. 30% of the fuel elements in all the 384 fuel bundles, for a total of 4,262 failed fuel elements. Realistically, fuel sheath failure is not expected to result from an accidental DSC drop from the low lift height of the transporter and processing building. The free inventory of tritium and Krypton-85 in the damaged fuel elements is assumed to be released into the DSC cavity. Ignoring the barriers provided by the transfer clamp seal and the sub-atmospheric pressure inside the DSC cavity, it is assumed that these radionuclides are released at once into the environment. The total dose to the public due to this event was assessed to be 1.5 μSv for an adult and 1.1 μSv for an infant at the Darlington site boundary. The dose to an individual in the immediate proximity of the DSC is assessed to be 4.5 mSv.

The internationally accepted criterion for assuring subcriticality in such storage facilities is that k_{eff} should be less than 0.95. Consistent with expectations for irradiated natural uranium fuel, the earlier analyses and assessments have yielded adequate subcriticality margin, and have demonstrated that there is no criticality of used CANDU fuel even in DSCs filled with water.

The specific cases analysed using the WIMS-AECL code for a DSC containing used fuel are given below.

Environment inside DSC	k_{∞}	k_{eff}
Dry inert atmosphere	0.4734	0.2114
Flooded with H_20, PuO_2 spheres	0.7815	0.7327
Flooded with H_20, Pu diluted	0.8119	0.7610

7.0 CONCLUSIONS

Safety assessments of OPG's dry storage facilities have demonstrated that these facilities can be operated safely and without undue risk to workers, members of the public, or the environment. Operating experience achieved at the Pickering Waste Management Facility and the Western Waste Management Facility has validated these assessments.

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Table 1 Credible Postulated Malfunctions or Accidents during DSC On-Site Transfer

Malfunction or Accident	Potential Maximum Dose Consequence to the Public (μSv)		Potential Maximum Occupational Dose Consequence (mSv)
	Adult	Infant	
Transporter Failure	1.5×10^{-3}	1.1×10^{-3}	4×10^{-3}
DSC drop during on-site Transfer	1.5	1.2	4.5
Earthquake	$<1.5 \times 10^{-3}$	$<1.1 \times 10^{-3}$	$<4 \times 10^{-3}$
Thunderstorms	$<1.5 \times 10^{-3}$	$<1.1 \times 10^{-3}$	$<4 \times 10^{-3}$
Tritium Removal Facility Explosion	$<1.5 \times 10^{-3}$	$<1.1 \times 10^{-3}$	$<4 \times 10^{-3}$
Hazardous Material Building Explosion	$<1.5 \times 10^{-3}$	$<1.1 \times 10^{-3}$	$<4 \times 10^{-3}$

Table 2 Credible Postulated Malfunctions or Accidents during DSC Processing

Malfunction or Accident	Potential Maximum Dose Consequence to the Public (μSv)		Potential Maximum Occupational Dose Consequence (mSv)
	Adult	Infant	
Drop of a DSC during handling	1.5	1.2	4.5
Earthquake	<1.5	<1.2	<4.5
Thunderstorms	0	0	0
Toxic Corrosive Chemical Rail Line Accident	1.5	1.2	4.5
Tritium Removal Facility Explosion	0	0	0
Hazardous Material Building Explosion	0	0	0

Table 3 Postulated Malfunctions or Accidents during DSC Storage

Malfunction or Accident	Potential Maximum Dose Consequence to the Public (μSv)		Potential Maximum Occupational Dose Consequence (mSv)
	Adult	Infant	
Seal-weld Failure during storage	0.21	0.16	0.6
Earthquake	0	0	0
Tornado	0	0	0
Thunderstorms	0	0	0
Rail Line Blast	0	0	0
Toxic Corrosive Chemical Rail Line Accident	0	0	0
Tritium Removal Facility Explosion	0	0	0
Hazardous Material Building Explosion	0	0	0

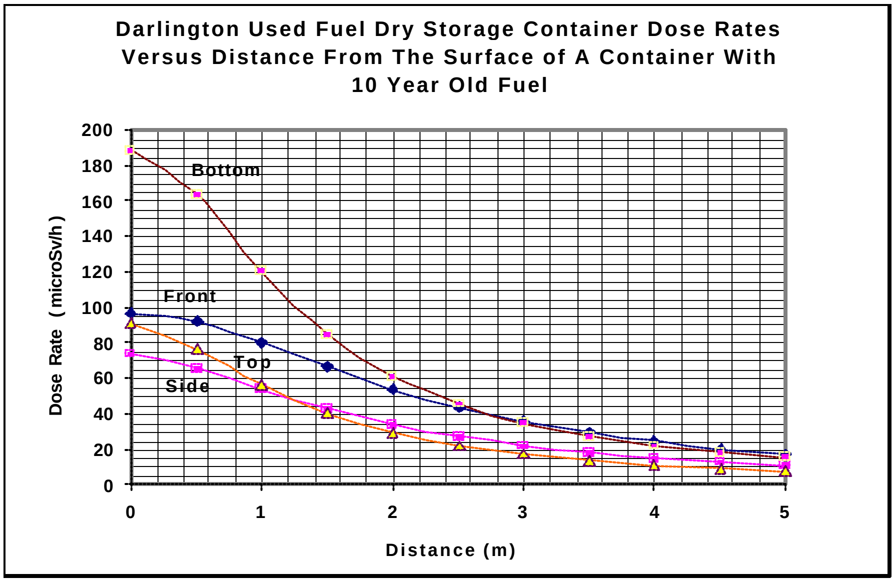

Figure 1. Dose rate versus distance from the surface of a single DSC with 384 reference fuel bundles 10-year cooled. The front corresponds to the wider face of the DSC facing the slightly enlarged cavity to accommodate long-long fuel bundles. The average concrete density of the DSC is 3.57 g/cc (taking into account the steel re-bar).

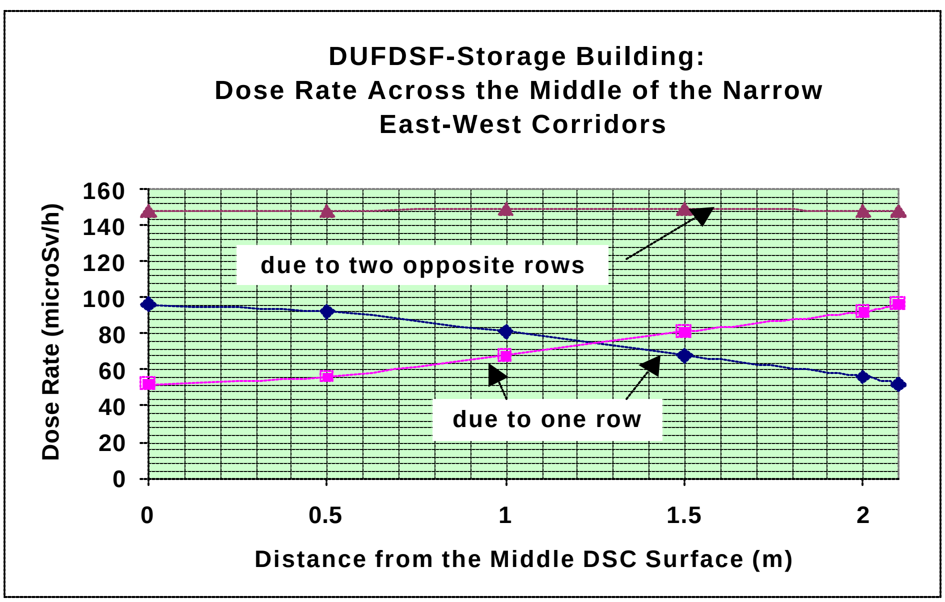

Figure 3. Dose rates across the narrow corridor inside the storage building.

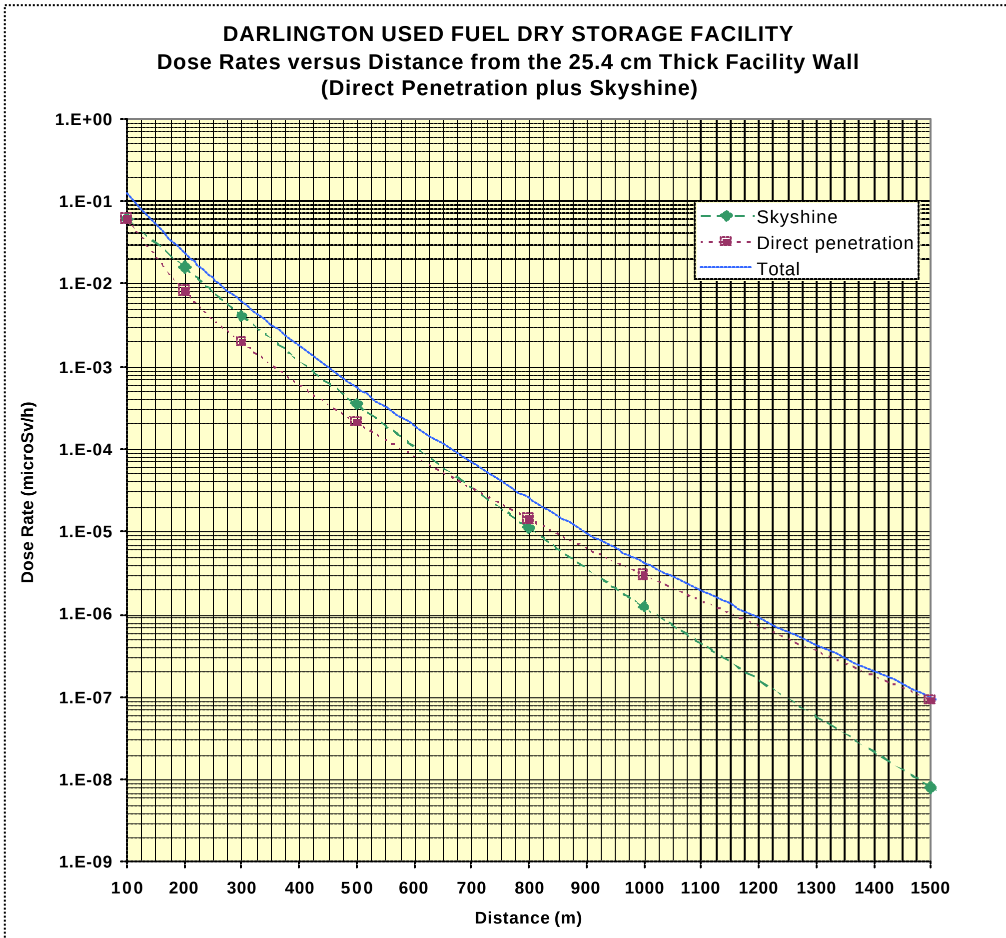

Figure 2. Total dose rates outside the 25.4 cm (10 in) concrete DUFDSF wall from 100 to 1500 m (direct penetration plus skyshine).

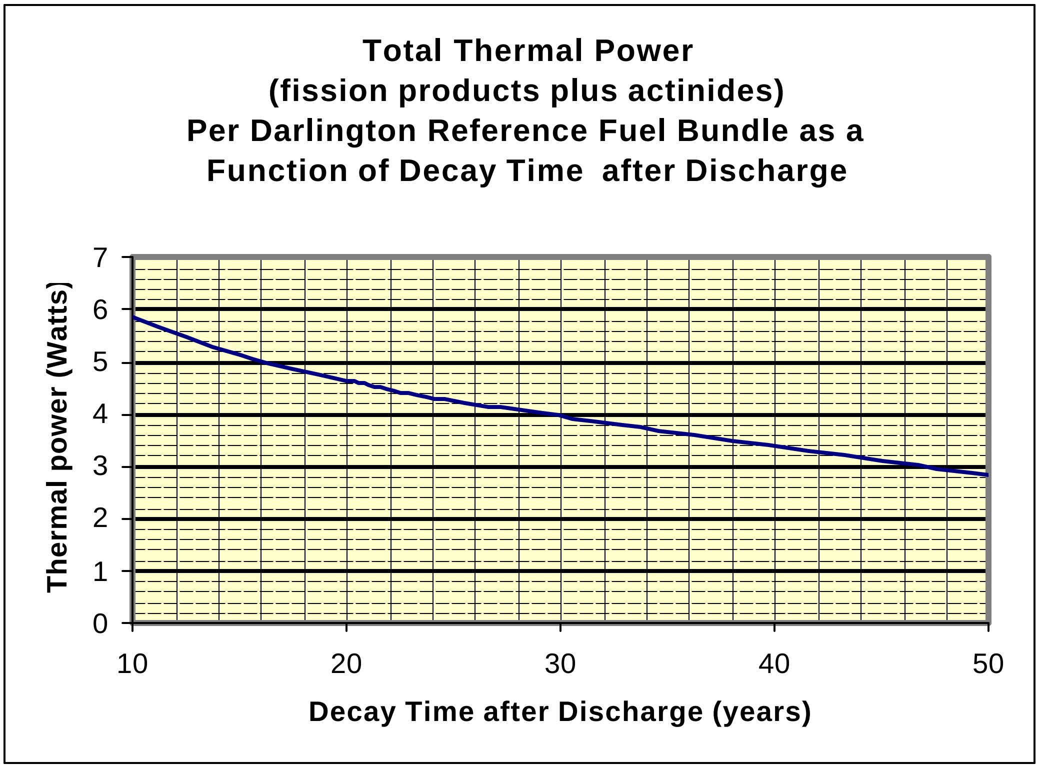

Figure 4. Total thermal power per reference fuel bundle as a function of decay time

after discharge from the reactor.