CHAPTER 10

RADIOLOGICAL RISKS FOR DEEP GEOLOGICAL DISPOSAL AND SURFACE STORAGE OF SPENT NUCLEAR FUEL

10.1 BACKGROUND INFORMATION

In September 1996, the United States Senate (Senate Report 104-320, p.98, September 12, 1996) requested an evaluation by EPA of alternatives to the disposal of radioactive materials in a deep geologic repository at Yucca Mountain, as well as an evaluation of public health risks of these alternatives against standards proposed for deep geologic disposal. Alternatives to be considered included: (1) storage of nuclear wastes at each site where it is currently stored and (2) one or more centralized above-ground storage sites.

Spent nuclear fuel (SNF) from the operation of commercial nuclear power reactors is currently stored at more than 70 nuclear generating sites around the country. It is expected that existing nuclear power plants will produce approximately 87,000 metric tonnes of spent fuel during their operational lifetimes. Approximately 40,000 metric tonnes of spent nuclear fuel were stored at commercial nuclear power reactors as of 1999. By the year 2003, this amount is expected to increase to 48,000 metric tonnes (NPJ97).

To date, most SNF is stored in water-filled pools at the reactor sites where it was generated. However, space is not available in existing pools to store all of the spent fuel expected to accumulate over the lifetime of the reactors. When the pool capacities were established, it was expected that the SNF would be removed from the reactor site for reprocessing about five years after discharge from the reactor. After national plans for reprocessing were terminated, removal of SNF from the reactor sites for central interim storage or disposal was expected to begin in January 1998, but these programs have been delayed. Consequently, additional SNF storage capacity is therefore needed.

Facilities for interim storage of SNF prior to disposal have been under consideration since 1972. In February 1972, the Atomic Energy Commission (AEC) began consideration of surface storage facilities at the Hanford site in the State of Washington. The facility would be used "...for high-level commercial wastes and low-level wastes from both commercial and AEC activities (HEW87)." In June 1972, the AEC revealed plans to develop a Retrievable Surface Storage Facility (RSSF), which would be an array of mausolea or vaults where waste or spent fuel canisters would be stored (CAR87).

The decision to choose the surface storage option "...was a response to the dilemma of irretrievability" and seemed a "practical answer to a difficult political and technical problem (HEW87)". The AEC concluded that such storage would be satisfactory for decades or centuries (USC72, USC75). The Congress' Joint Committee on Atomic Energy reported in 1972 that the radioactive waste management program "...now includes the conceptual design of manmade surface facilities of an expected lifetime of several centuries (USC72)."

Three technical concepts for the RSSF were considered by the AEC: (1) stainless steel canisters in water basins for heat removal and shielding; (2) canisters in concrete basins, cooled by circulating air; and (3) a canister within a two-inch thick container with doubly-contained waste in a three-foot thick concrete cask cooled by circulating air.

Recently, members of Congress, other public officials, environmental groups, and private citizens have expressed concern that a surface storage facility might be regarded as a *de facto* repository, thereby reducing the impetus for building a geologic repository as expeditiously as possible. To allay these concerns, proposed Monitored Retrievable Storage (MRS) facilities have consistently limited the total storage capacity to well below total SNF quantities projected for permanent deep geologic disposal. MRS designs have been proposed from a few thousand up to 15,000 metric tonnes uranium (MTU). This chapter compares the potential impacts of continued storage to those associated with disposal at a geologic repository at Yucca Mountain. The comparison is presented in terms of applicable regulatory limits (Section 10.2) and anticipated (estimated) risks (Sections 10.3 and 10.4).

10.2 REGULATORY LIMITS

The Energy Policy Act of 1992 (EnPA) directs the Administrator to establish, after consultation with the National Academy of Sciences (NAS), a maximum individual dose standard for the proposed repository at Yucca Mountain. The NAS found that such an approach would provide protection for all exposed individuals and suggested that levels of a few millirems/year to a few tens of millirems per year (mrem/yr) would provide a reasonable point of departure for rulemaking. Therefore, for the purposes of comparison, it is assumed that the Yucca Mountain standard will establish a maximum individual dose standard. Further, absent the Administrator's final decision on the level of the standard, this comparison uses the upper end of the NAS'suggested range (i.e., a few tens of millirems per year) to characterize the allowable exposure limits for the proposed repository. This limit, which encompasses all exposure pathways, will apply to releases from the repository over an extended period of time. The NAS has recommended that it apply during the period of geologic stability for the site, a time frame that could extend to one million years.

The regulatory limits that would apply to continued surface storage depend upon the specific site at which the wastes are stored. For continued storage at the sites at which the spent fuel and high level wastes were generated or are currently stored, different limits apply to the following types of facilities:

- Power reactors
- Research reactors
- Independent spent fuel storage installation (ISFSI)
- DOE facilities

If the government opts to store wastes at a centralized facility such as an an ISFSI or an MRS facility, the applicable limits would likely be those for an ISFSI. However, if the ISFSI or MRS were constructed on a site owned by DOE prior to 1983, the limits applicable to DOE facilities would likely apply.

10.2.1 Power Reactors

For waste stored at power reactors, the applicable regulatory limits would be those established by EPA for the nuclear fuel cycle (40 CFR Part 190), and the NRC (10 CFR Parts 20, 50, and 100). 40 CFR Part 190 establishes exposure limits from normal facility operations of 25 mrem/yr to the whole body or any organ (except the thyroid, which is allowed 75 mrem/yr). These limits, established under the old "whole body/critical organ" protection concept, are roughly equivalent to a 15 mrem/yr limit under the current "effective dose equivalent (EDE)" protection concept. The 40 CFR Part 190 limits consider all exposure pathways and require the site to consider potential exposures from all facilities that are part of the nuclear fuel cycle. Under 10 CFR Part 20, power reactors are required to maintain exposures as low as is reasonably achievable (ALARA), consistent with maximum individual exposures of 100 mrem/yr “total effective dose equivalent” (TEDE) at an exposure rate not to exceed 2 mrem/hr. The limits for normal operations at power reactors can be characterized as ALARA, with maximum individual exposures limited to less than two millirem per hour (mrem/hr) TEDE and not to exceed 75 mrem/yr to the thyroid or 25 mrem/yr to the whole body or any other organ. The EPA’s recent evaluation of the airborne emissions from power reactors during its reconsideration of the National Emission Standards for Hazardous Air Pollutants (NESHAPS) (40 CFR 61 Subpart I) found that the ALARA design objectives and limiting conditions of operations set forth in Appendix I to 10 CFR Part 50 are being met by power reactor licensees and that actual exposures from the air pathway are a fraction of the regulatory limits.

In addition to the operating limits on power reactor licensees, 10 CFR Part 100 establishes siting criteria that assure that no member of the public will receive an exposure greater than 25 rem to the whole body or 300 rem to the thyroid over a two-hour period of exposure to a fission product release associated with a hypothetical major accident at the facility.

10.2.2 Research Reactors

The regulatory limits for research (non-power) reactors are established by 10 CFR Part 20. Exposures via all pathways to any member of the public from normal operations are also required to be as low as reasonably achievable, consistent with a maximum individual exposure limit of 100 mrem/yr TEDE at an exposure rate not to exceed 2 mrem/hr. Additionally, under the recently adopted "constraint rule," corrective actions must be initiated should exposures via the air pathway exceed 10 mrem/yr TEDE. Given this and a 50/50 split of the 100 mrem/yr limit between the air and liquid pathways outlined in 10 CFR Part 20, the effective maximum individual dose limit for research reactors from normal operations is 60 mrem/yr TEDE. No quantitative limits are imposed on research reactors for demonstrating ALARA for non-airborne exposure pathways, nor are quantitative criteria given for exposures from accidental releases. However, site suitability is considered during licensing.

10.2.3 Independent Spent Fuel Storage Installations (ISFSIs)

The regulatory requirements for ISFSIs are those established by the NRC in 10 CFR Parts 20 and 72. The limits established in Part 20 require exposure to be maintained as low as reasonably achievable, consistent with a maximum individual exposure limit of 100 mrem/yr TEDE at an exposure rate not to exceed two mrem/hr. Additionally, Part 72 imposes the nuclear fuel cycle exposure limits of 40 CFR 190. Therefore, like power reactors, the limits for normal operations of an ISFSI can be characterized as ALARA with maximum individual exposures limited to less than two mrem/hr TEDE and not to exceed 75 mrem/yr to the thyroid or 25 mrem/yr to the whole body or any other organ.

10 CFR Part 72 also establishes an exposure limit of five rem to the whole body or any organ in the case of an accident. This limit is applied at the boundary of the controlled area and cannot be exceeded by any design basis accident.

It should be noted that the limits for an ISFSI currently apply to dry storage of spent fuel at operating nuclear power plants, but not the original wet fuel pools. However, as reactors reach the end of their operating lives and are decommissioned, it is likely that long-term storage of all spent fuel would be conducted under a license issued pursuant to 10 CFR Part 72.

10.2.4 DOE Facilities

Relevant exposure limits for DOE facilities are those established by EPA for airborne releases of radionuclides (40 CFR 61, Subpart H) and by DOE Order 5280. 40 CFR 61, Subpart H, limits airborne releases of radioactive materials (excluding radon and its decay products) to quantities that do not cause any member of the public to receive an exposure greater than 10 mrem/yr EDE. The limits established by DOE Order 5280 mirror 10 CFR Part 20; i.e.,, exposures are to be as low as reasonably achievable, consistent with a maximum individual exposure limit of 100 mrem/yr TEDE from all pathways at an exposure rate not to exceed two mrem/hr. Given the constraint rule for the airborne pathway and a 50/50 split of the 100 mrem/yr limit between the air and liquid pathways, the limits for DOE facilities can be characterized as ALARA with maximum individual exposures not to exceed 60 mrem/yr TEDE at a dose rate of less than two mrem/hr (10 CFR Part 20).

No quantitative exposure criteria for accidental releases from DOE facilities are established by DOE Order 5280 or 40 CFR Part 61.

10.2.5 Summary of Regulatory Limits

As the above subsections have detailed, different regulatory limits would apply at existing or future storage sites, depending upon the specific use of the site. However, given the ALARA requirement imposed on all sites and the various limits on maximum annual exposures, it is not unreasonable to expect that exposures from undisturbed storage would be on the order of a few tens of millirems per year at any of these facilities. This level of exposure is consistent with the limits that will likely be promulgated for Yucca Mountain.

Such comparability cannot be assumed in the event of accidents. The Yucca Mountain standards explicitly require that natural phenomena be evaluated and factored into the design of the facility. Thus, a maximum individual exposure limit of a few tens of millirems per year would apply to exposures caused by both accidents and natural phenomena. By contrast, no explicit criteria exist for accidents at research reactors or DOE facilities. Consequently, the limits for storage at power reactors and ISFSIs could potentially allow individual exposures of up to five rem EDE in the case of an accident.

10.3 REPORT BY THE MONITORED RETRIEVABLE STORAGE REVIEW COMMISSION

Information presented in this and the following section addresses the risks associated with storage of SNF at reactor sites and at a central interim storage facility, characterized as a Monitored Retrievable Storage Facility. These risks are compared to those associated with disposal at the proposed Yucca Mountain repository site in Nevada. A review of literature on the risks of SNF storage and disposal revealed a large body of information. However, no studies to date have specifically addressed the scope of the directives of the Senate Report. Past studies have focused on dry storage at reactors and on an MRS facility as part of a dynamic total waste management system configuration. Despite its limitations, this information was used as the basis for the data presented in this section of the BID.

In 1987, the Nuclear Waste Policy Amendments Act established the Monitored Retrievable Storage Review Commission. The Commission's charter was to compare storage of spent fuel at a Federal MRS facility to storage of spent fuel at the reactors at which it was generated. Through public hearings, the Commission solicited the views of private citizens, technical experts, and utility and government representatives. In addition, several contractors performed specific tasks to augment the Commission's technical work. The Commission's final report, issued in November 1989, examined each alternative's merits, including an assessment of radiological doses and risks to members of the general public. The report's principal findings are summarized below (MRS89).

10.3.1 <u>At-Reactor Storage Options</u>

Water-filled pools have been used for SNF storage since the earliest days of nuclear reactor operation and are universally used for storage of commercial Light Water Reactor (LWR) fuels today. Spent fuel pools employ a large amount of water for heat removal and radiation shielding. Pools remain a proven method for cooling LWR fuels for periods lasting from five to ten years. Thereafter, the fuel can be removed from the pool to make room for new inventories of hot fuel.

As stated above, existing pool capacities were not designed to accommodate SNF quantities generated from the 40-year operating life of a reactor. Expansion of storage capacity at the reactor sites includes options that are broadly categorized as wet storage and dry storage. (Rod consolidation for expanding on-site storage capacity is currently not considered a viable option and was therefore not considered.)

Two wet storage options commonly considered are spent fuel reracking and new pool construction:

- <u>Spent Fuel Reracking</u>. This option entails a reconfiguration for high density storage. Typically, this is done by manufacturing fuel racks that bring fuel elements closer together in order to create additional storage space. Most utilities have reracked their pools at least once. From the original typical design of 1-1/3 reactor core storage capacity, utilities have frequently increased storage to four to six reactor cores. However, even with these measures, pool-storage capabilities at most reactors will not be considered adequate.

- New Pool Construction. This option entails the construction of a pool for long-term storage of SNF. Due to high costs and operational/maintenance factors, new pool construction is not considered competitive to dry storage.

Since the early 1980s, demonstration projects at several utilities and research by the Electric Power Research Institute have demonstrated the viability of dry storage methods for SNF. Many different dry storage methods have been proposed and/or tested, including metal or concrete storage casks, air-cooled vaults, and universal multipurpose casks. (Note: In 1990, the NRC amended its regulations to authorize licensees to store spent fuel at reactor sites in storage casks approved by the NRC. To date, seven cask designs have received certificates of compliance (NPJ97).) A generic description of dry storage methods includes:

- Modular Concrete Vaults (MCV). MCVs consist of sealed metal tubes inside an above surface concrete structure. Inside the sealed metal tubes, the spent fuel is kept under an inert cover gas or air. The tubes are typically made of carbon steel and each tube contains a single fuel assembly or a single element. The MCV has received site-specific NRC licensing.

- Horizontal Concrete Vaults (Modules). Horizontal concrete modules keep the fuel inside a sealed stainless steel canister back-filled with an inert gas. The canister is protected and shielded by an above-surface concrete module. The heat generated by the spent fuel is removed by thermal radiation, conduction, and natural convection through air channels in the concrete module. The canister contains a basket for holding the fuel in place. The horizontal concrete vault has received site-specific NRC licensing.

- Metal Dry Casks (MDC). Metal dry casks are the most mature of the methods available for dry interim storage. Their use was successfully tested and demonstrated in 1984. The casks are large heavy vessels (100 to 125 tonnes loaded). They are equipped with an internal basket for holding the spent fuel assemblies or elements. The body is made from forged steel, modular cast iron, or lead and stainless steel with a double seal lid. The MDC has received site-specific NRC licensing.

- Concrete Dry Storage Casks. Concrete dry storage casks are similar to metal dry casks except that the body of the cask is made of steel-reinforced concrete with an inner metal liner for containment. Concrete dry storage casks have received site-specific NRC licensing.

- Dual Purpose Casks (DPC). The DPC was derived from the metal dry cask concept. The design and manufacture are very similar to that of the metal dry cask, but it would also be used to transport the fuel to a Federal spent fuel management facility. The fuel would be removed upon arrival at the spent fuel management facility, but would not be disposed of in a DPC. The DPC remains in development.

- Multi-Purpose Casks (MPC). This dry cask combines storage, transportation and disposal in one container. The MPC would potentially allow spent fuel to be stored, transported and disposed of in the same container in which it was originally placed. The use of the MPC would not require the fuel to be extracted from it prior to being placed in a repository.

10.3.2 Radiation Exposure Modeling Assumptions for At-Reactor Storage of SNF

To model public radiation exposures associated with at-reactor storage of SNF, the MRS Commission Report assumed that there were to be no new orders for nuclear power plants beyond those operating or being constructed as of December 1987. For post-1988 SNF, burnup rates of 36,600 megawatt-days per metric ton uranium (MWd/MTU) and 42,000 MWd/MTU were assumed for BWRs and PWRs, respectively.

It was also assumed that all fuel would be stored at the reactor sites until DOE was ready to accept the waste and ship it to the repository for disposal. The analysis assumed utilities would select from a number of currently available options to provide at-reactor storage that includes fuel-pool reracking supplemented by dry-storage. Since most utilities have already reracked their pools at least once, the Commission concluded that future reracking will be limited. Adding a second tier of racks was not considered a practical way to expand the current pool storage capacity. It was further assumed that there would be no transhipment of spent fuel from one reactor site to another to alleviate storage problems. Every utility would maintain enough pool storage capacity so that the full core of the reactor could be unloaded into the spent fuel pool, if necessary.

For the balance of life-of-plant SNF, dry storage, either at the reactor sites or on utility-owned land contiguous to the reactor, was assumed to involve metal or concrete casks. The typical metal dry storage cask is made of stainless steel or nodular cast iron that may hold from a few to 25 PWR fuel assemblies per module. For concrete casks with an inner metal liner, the unventilated type will hold nine PWR or 25 BWR fuel elements, while a ventilated type may hold 17 PWR or 50 BWR elements.

Although a few reactors have already been permanently shut down, the majority of currently licensed and operating facilities will reach their end-of-operating life between the years 2009 and

2030. Table 10-1 shows the amounts of five-year-old SNF that are expected to be stored in fuel pools and in dry storage.

Table 10-1. Spent Fuel Accumulation at Shutdown Commercial Light Water Power Reactors (Source: MRS89)

Year	MTU of Five-Year-Old Fuel		
	Dry Storage	**Pool Storage**	**Total**
2000	0	500	500
2005	0	500	500
2010	0	1,500	1,500
2015	500	4,000	4,500
2020	7,500	21,000	28,500
2025	12,000	29,000	41,000
2030	19,500	40,000	59,500
2035	27,000	54,000	81,000
2040	29,000	55,500	84,500
2045	30,000	57,000	87,000

10.3.3 Model Assumptions for MRS Storage of SNF

For the MRS alternative analyzed by the Commission, spent fuel is assumed to be stored at the reactors until an MRS facility becomes available. At this time, spent fuel from some reactors would be transported to and stored at an MRS until a repository is available for permanent disposal. The MRS would continue to operate until all the spent fuel has been emplaced in the repository.

The MRS facility as defined in the Nuclear Waste Policy Amendments Act of 1987 was analyzed. Given the MRS capacity limitation of 15,000 MTU, most spent fuel would be stored at the reactor site. The MRS would, therefore, only supplement at-reactor storage and reduce the need for dry storage at reactor facilities, as defined in Table 10-2. Dry modular storage, using technologies similar to those described in Section 10.3.1, was assumed for the MRS facility.

Table 10-2. Reduction in Dry Storage Needs At Reactor Facilities
with Linked MRS (Source: MRS89)

Cases	Maximum MT in Dry Storage	Difference From No-MRS Case
No-MRS	7,693	---
Linked MRS*	3,562	4,131

*MRS schedule linked to repository schedule.

10.3.4 Transportation Models for SNF With and Without MRS

The MRS Review Commission's dose assessment for members of the public also included radiation exposure that would result from transportation of wastes if an MRS facility were part of the spent fuel management system and if it were not. In the absence of an MRS facility, transit doses could result to members of the public along the paths of travel between individual reactor sites and a repository assumed to be at the Yucca Mountain Site. With an MRS facility, transit exposures could occur between: (1) reactor sites and the MRS facility, (2) reactor sites and the repository, and (3) the MRS facility and the repository. For SNF shipments originating from reactors, 54 percent would be shipped by rail and 46 percent by truck; 100 percent of the shipments between the MRS facility and the repository were assumed to be made by rail. These assumptions will hold despite the location of the MRS, since all waste that could be moved by rail (rail head at reactor site) were assumed to be moved by rail. The assumed MRS location was actually in the Eastern United States.

Table 10-3 identifies primary parameters that define transportation risk for each of the base cases. In effect, these parameters serve as surrogate measures to approximate risk. For example, to compare the relative radiological risk of the three base cases, the total number of shipment miles or cask miles was added. The underlying assumption was that public exposure was proportional to the number of casks and distances traveled.

Table 10-3. Life-Cycle Transportation Risk Measures*
(Source: MRS89)

Cases	Shipment-miles (in millions)	Cask-miles (in millions)
No-MRS	64.7	74.1
Linked MRS**	26.9	40.3

* Repository in 2013 (54% rail/46% truck from reactor; 100% rail from MRS facility)
** MRS facility to begin operations in 2010

10.3.5 Public Exposure From SNF Storage

Radiation doses to members of the public were assessed for spent fuel management operations at reactor sites, the MRS facility, and the repository. The computer model used to evaluate radiation doses for different system configurations was MARC: MRS Review Commission's Analysis of System Risk and Cost. MARC is a network model that incorporates DOE's computer code TRICAM. TRICAM describes how spent fuel moves through the system and is used for modeling transportation. The code also uses facility-specific data such as reactor spent fuel discharges, population data, reactor rail accessibility, repository capacity and demographics, and acceptance schedules.

Table 10-4 summarizes population doses predicted by MARC for members of the public. Population dose estimates were 130 person-rem for individuals living within a 50-mile radius of the 70 reactor sites; 4 person-rem for individuals living within a 50-mile radius of an MRS facility located in the eastern United States; and 0.125 person-rem for individuals living within a 50-mile radius of a deep geologic repository assumed to be located at Yucca Mountain.

The Commission report found that public exposures from spent fuel were small at all locations associated with SNF management. By far, the largest source of public exposures was estimated to result from the transportation of SNF between reactor sites and the MRS facility and the repository and between the MRS facility and the repository. Table 10-4 shows that public transportation exposure is reduced by more than a factor of two (i.e., from 7,900 to 3,400 person-rem) when an MRS facility is included in the management of SNF. This is almost exclusively due to reduced shipping miles and its attendant shift from truck to rail services when an MRS facility is employed.

Table 10-4. Total Life-Cycle Doses in Person-Rem from Spent Nuclear Fuel Management With and Without MRS
(Source: MRS89)

Activity Center	At-Reactor Storage Only (person-rem)	At-Reactor Storage Plus MRS (person-rem)
All Reactors	130	13
MRS Facility	Not applicable	4
Repository	0.1	0.1
Transportation	7,900	3,400
TOTAL	~ 8,000	~3,500

Results reported by the MRS Review Commission (as summarized in Table 10-4 above) cannot be used directly to compare potential public risks from SNF stored at the proposed Yucca Mountain Site (YMS) with the alternatives of At-Reactor Storage and MRS storage. The dose estimates given in Table 10-4 correspond to <u>different</u> and <u>changing</u> SNF inventories at these facilities over different time periods. The variations in SNF quantities and locations with time associated with the MRS Review Commission evaluations are shown in Table 10-5.

By interpolation, average SNF quantities can be defined for each time period that, when added, yield time-weighted quantities of stored SNF at the reactor facilities, MRS, and repository in terms of metric tonnes of uranium-years (MTU-years):

- <u>For At-Reactor Storage,</u> which spans the 50-year period between 1995 and 2045, fuel pool and dry storage correspond to 1,672,067 MTU-years

- <u>MRS Storage</u> that is assumed to begin in 2010 and ends in 2045 represents a time-weighted SNF storage value of 106,400 MTU-years

- <u>Repository Storage and Disposal</u> that is assumed to begin in 2013 yields a cumulative value of 1,302,431 MTU-years

Table 10-5. Location of Spent Fuel With MRS in 2010 and Repository in 2013 (MTUs)
(Source: MRS89)

Year	Pool Storage At-Reactor	Dry Storage At-Reactor	MRS Storage	Repository
1995	28,680	1,286	---	0
2000	36,807	3,711	---	0
2005	42,026	8,019	---	0
2010	46,362	11,532	2,400	0
2015	49,914	7,907	12,100	1,149
2020	43,857	5,819	15,000	12,798
2025	36,799	4,208	15,000	27,715
2030	28,037	459	15,000	45,644
2035	18,949	0	9,800	57,596
2040	9,831	0	3,000	72,436
2045	0	0	1,000	86,756

From the MRS Review Commission's previous estimates of public exposures, normalized public dose estimates can be derived that provide a fair comparison for these three modes of SNF storage (Table 10-6). Based on normalized values, public exposures that would result from SNF stored at reactor sites and a designated MRS are nearly the same. For storage at a repository, however, public exposure is projected to be lower by two to three orders of magnitude.

The projected similarity of public doses for at-reactor and MRS storage is to be expected if it is assumed that: (1) release fractions of stored SNF for these two alternatives are either identical or very similar and (2) the population density and distribution for the hypothetical MRS facility are similar to the 0-50 mile populations that characterize each of the 70 reactor facilities expected to store SNF onsite.

The much lower collective population exposure estimated for repository storage is also to be expected. For deep geologic disposal, the release fraction from the waste package to the biosphere is likely to be reduced by at least two to three orders of magnitude, as suggested by the reduced normalized population dose estimates in Table 10-6. It can also be assumed that cited population dose estimates will decline to even lower levels when the repository progresses from its operational phase to post-closure that includes backfilling of all access shafts and repository penetrations (MRS89).

Table 10-6. Comparison of Public Exposures Resulting from Three SNF Storage Alternatives

SNF Storage Alternative (Collective Dose)*	Time-Weighted SNF Quantities (MTU-Yrs x 1,000)	Normalized Public Exposure	
		Person-rem per 1,000 MTU-Yrs	Person-rem per Year for 70,000 MTU
At-Reactor Storage (130 person-rem)*	1,672	7.8E-02	5.5E+00
MRS (4 person-rem)*	106	3.8E-02	2.7E+00
Repository (0.125 person-rem)*	1,302	< 9.6E-05	6.7E-03

*MRS Review Commission's public dose estimates (see Table 10-4).

10.4 OTHER INFORMATION SOURCES

A comprehensive literature review was performed to identify other potential sources of information concerning radiation exposures associated with SNF management and disposal. Additional data were needed to: (1) confirm and/or compare dose estimates cited by the Commission, and (2) supplement Commission's data that were limited to collective population doses from routine facility operation. Lacking in the Commission Report were dose data for the reasonably maximally exposed individual (RMEI) and doses linked to accidental releases.

Estimates of offsite doses that result from accidental releases of radioactivity to the environment are complex and require predictive risk analyses that include: (1) a facility-specific characterization, (2) identification of potential accident scenarios, (3) estimation of accident probability, and (4) pathway modeling that incorporates site-specific data on weather, population distribution, land use, hydrology, etc.

It was found that, for the studies reviewed, a simple comparison of reported dose estimates is made difficult by variations in the studies' scope and objectives, selection of accident scenarios, and model-parameter values. To provide a common basis for comparison, secondary information, when provided within each study, was used to convert reported data to a common normalized value that would permit comparison. Summarized below are the most relevant studies and their estimates for doses to members of the public.

10.4.1 “An Assessment of LWRS Spent Fuel Disposal Options” (BEC79)

This study was conducted by the Bechtel Corporation for the DOE National Waste Terminal Storage Program and provided background documentation that dealt with three treatments of SNF prior to disposal at a repository:

Case 1: Simple encapsulation and disposal of spent fuel at the repository following storage at an ISFSI for nine years

Case 2: Encapsulation of fuel, end fittings, and secondary wastes after chopping the fuel bundle and removal of volatile materials

Case 3: Encapsulation of fuel, end fittings, and secondary wastes after chopping, removal of volatile materials, calcination, and vitrification

These base case scenarios assumed a spent fuel throughput of 5,000 MTU per year at a processing and encapsulation facility before the SNF was shipped to a repository for final disposal.

Risk analysis at the repository was further limited to the preclosure period, which defines the period of emplacement of the processed/encapsulated SNF. Estimates of public exposures were defined for normal operations and “shaft drop” accident conditions as summarized in Table 10-7.

Table 10-7. Public Doses for Normal Repository Operation and From Shaft-Drop Accident (Based on data from BEC79)

SNF Packaging	0-50 Mile Population Dose (person-rem/1,000 MTU-yr)
Case 1	
Normal Operation	1.0E-06
Accident	1.1E-02
Case 2	
Normal Operation	1.5E-06
Accident	1.1E-02
Case 3	
Normal Operation	2.0E-06
Accident	1.1E-02

10.4.2 “Generic Environmental Impact Statement, Management of Commercially Generated Radioactive Waste” (DOE80).

This study, known as the GEIS, was issued as a basis for reexamining the strategy for disposing HLW in a mined geologic repository with several alternative configurations. Reference repositories located in salt, granite, shale, and basalt were analyzed at a “reference” location in a midwestern state. Preclosure facility risk categories analyzed included routine exposures to the regional population and maximum individual exposure from potential worst-case accidents. Routine doses to the regional population from chronic radiological releases to the atmosphere were derived from the standard Gaussian dispersion model. Of a total of 207 potential accident scenarios considered, 116 had the potential for offsite exposure. Dropping a spent fuel canister was considered the most serious radiological event, with an estimated frequency of occurrence of 1.0E-05 per year.

Exposure to the 50-mile population from routine repository preclosure operation was considered “negligible,” with no quantitative estimate given.

For all accidents considered, public exposure was estimated at 5.0E-05 person-rem per year; and for any single worst-case accident, a maximum individual exposure of 1.1E-04 rem was estimated.

10.4.3 “Review of Dry Storage Concepts Using Probabilistic Risk Assessment” (ORV84)

This study provided a comparative risk analysis for dry storage of SNF at reactor sites. Assessed storage designs included drywall, storage cask, and vault. The reference reactor facility was a 1,000-MWe PWR that was assumed to discharge 60 spent fuel assemblies (~19 MTU) to the fuel pool. After five years of cooling, SNF was transferred to dry storage with a capacity of 2,400 fuel assemblies (~62 MTU).

To model accidental release fractions, a starting assumption was that one percent of the fuel was failed prior to any of the accident scenarios. Principal elements modelled included long-lived radionuclides of Kr, I, Cs, Sr, Cs, and actinides.

Conservative model parameters were used to estimate population doses out to a distance of 200 miles from the reference reactor site. For example, primary model parameters used included the relatively high population density distribution for the Zion nuclear power plant, one meter per second wind velocity, and stability class D. Table 10-8 defines population dose risks for 12

Table 10-8. At-Reactor Storage Accidents: Summary of Results (ORV84)

Accident Scenario	Frequency (events/yr)	Number of Assemblies	Dose Consequence (Person-rem/event)	Dose Risk (Person-rem/yr)
Fuel Assembly Drop During Loading	1E-01	1	4E-01	4E-02
Drop of Transport Cask During Loading				
Cask	4E-03	10	4E+00	2E-02
Drywall	7E-02	10	4E+00	3E-01
Venting of Cask During Transport				
Cask	2E-03	24	1E+03	2E+00
Drywall	3E-02	1	4E+01	1E+00
Collision During Transport				
Cask	2E-04	24	1E+03	2E-01
Drywall	2E-05	1	4E+01	8E-04
Collision with Fire During Transport				
Cask	2E-06	24	5E+03	1E-02
Drywall	2E-07	1	2E+02	4E-05
Canister Drop During Emplacement				
Drywall	1E-06	1	4E+01	4E-05
Canister Shear During Emplacement				
Drywall	2E-06	1	4E+01	8E-05
Cask Drop During Emplacement				
Cask	1E-05	24	1E+03	1E-05
Tornado Missile Penetration				
Cask	6E-06	10	4E+02	2E-04
Drywall	1E-04	10	4E+02	4E-02
Plane Crash Topples Cask with Fire				
Cask	6E-09	24	5E+03	3E-05
Plane Crash Plus Fire				
Cask	9E-09	24	5E+03	4E-05
Drywall	2E-07	1	2E+02	4E-05
	2E-08	10	2E+03	4E-05
Earthquake				
Cask	4E-06	24	1E+03	4E-03
	4E-08	2,400	1E+05	4E-03
Drywall	8E-06	1	4E+01	3E-04
	8E-07	10	4E+02	3E-04
	2E-08	2,400	2.4E+04	5E-04
Total Risk: Cask				2.3E+00
Drywall				1.4E+00

accident scenarios. For cask and drywall storage, the annual population doses of 23 and 14 person-rem, respectively, were estimated.

10.4.4 “Requirement for the Independent Storage of Spent Fuel and High-Level Radioactive Waste” (NRC84).

This environmental assessment study performed by the Nuclear Regulatory Commission analyzed a fuel storage installation (i.e., MRS) that was intended to accommodate 70,000 MTU for a period of 70 years using dry store technology. It was further assumed that the facility would receive 3,500 MTU per year for a period of 20 years, with an additional 50-year storage period at maximum capacity.

The NRC assumed MRS construction designs that effectively reduce potential air emissions to near zero levels. Public doses resulting from routine operation/storage were, therefore, assumed to be “insignificant.”

For accidental fuel canister failure containing 1.7 MTU, the Commission estimated a maximum individual dose of about 1×10^{-6} rem per year per event.

10.4.5 “Environmental Assessment Related to the Construction and Operation of the Surry Dry Cask Independent Spent Fuel Storage Installation” (NRC85)

Virginia Electric and Power Company conducted a study as part of an application for a license to construct and operate an onsite Dry Cask ISFSI. The function of the Dry Cast ISFSI was to provide on-site interim storage for about 420 MTU of spent fuel from its reactors, Surry 1 and 2.

Based on cask design and specifications, no liquid and gaseous releases were assumed and only direct irradiation was considered for an exposure pathway under normal facility operations. The estimated maximum annual dose to the nearest actual offsite person located at 2.5 km from a direct radiation source was estimated to be 6E-05 mrem/yr.

For a hypothetical worst case accident, an upper-bound individual dose of 1.35 mrem/yr was estimated at the site boundary.

10.4.6 “Environmental Assessment Deaf Smith County Site, Texas” (DOE86a)

This DOE study evaluated the suitability of an HLW repository site in salt as specified in the NWPA of 1982. The reference repository was assumed to have a capacity of 36,000 MTU SNF and 3,510 metric tonnes of DOE-HLW. SNF was assumed to be 6.5 years old with an average annual receipt rate of 634,000 rods per year.

Estimated offsite radiological effects for routine and accidental releases are cited in Table 10-9.

Table 10-9. Preclosure Exposure Associated with a Reference Salt Repository
(Based on data from DOE86a)

Scenario	Dose
Normal Operation	
50-Mile Population	390 person-rem/yr
RMEI	5.6E-03 rem/yr
Worst-Case Accident	
50-Mile Population	3,000 person-rem/event
RMEI	4.7E-02 rem/event

10.4.7 "Preliminary Assessment of Radiological Doses in Alternative Waste Management Systems Without an MRS Facility" (SCH86)

This study analyzed the effects of nine waste management system alternatives, excluding an MRS facility, specifically dealt with at-reactor facilities and the surface facilities of a deep geologic repository site. The nine alternatives largely involved transportation modes and options between reactors and the repository and resulted in nominal differences in public exposures.

Public doses from routine waste management activities at reactor facilities were estimated to be at less than one person-rem per year per 1,000 MTU. The potential drop of a fuel assembly was cited as the typical accident scenario, with an occurrence frequency estimated at 0.006 per year and an estimated population exposure of 0.1 person-rem per event.

Public doses under normal operating conditions at the surface facility of a repository would likely be due to effluents associated with cask venting and fuel consolidation. On a 1,000 MTU basis, public exposure was estimated to be 6 person-rem. Public doses from accidental fuel-assembly and shipping cask drop were estimated to vary between 0.03 and 0.006 person-rem per year. Table 10-10 summarizes the public-dose estimates cited in this study.

Table 10-10. Public-Dose Estimates for Reference Reactor and Repository Surface Facility (Based on data from SCH86)

Routine Activity	Public Dose (person-rem/1,000 MTU)
• SNF Handling At-Reactor	1
• SNF Handling/Consolidationat Repository Surface Facility	6
• Transportation Between Reactor and Repository	164
Accidental Conditions	**Public Dose (person-rem/yr)**
• At-Reactor	0.0006
• At Repository Surface Facility	0.03 - 0.006

10.4.8 "Monitored Retrievable Storage Submission to Congress" (DOE86b)

This comprehensive environmental assessment study was prepared by DOE and submitted to Congress under Section 141 of the Nuclear Waste Policy Act. The study assessed impacts to humans and the environment from construction, operation, and decommissioning of an MRS facility at three potential sites (Clinch River, Oak Ridge, and Hartville, Tennessee) and two possible designs for a storage system: storage casks and field dry wells. Radiological doses from routine emissions from the MRS facility were estimated, along with doses resulting from specific accident scenarios. The facility was assumed to have a 26-year operating life with a total throughput of 62,000 MTU.

For routine operations that assume a "store-only MRS" and a repository that containerizes intact fuel, the estimated population doses are given in Tables 10-11a and 10-11b.

Tables 10-11a and 10-11b indicate that: (1) exposures to members of the public are significantly greater for routine operations than for accidental releases, (2) exposures are consistent with values cited by the MRS Review Commission, and (3) exposures for the RMEI are well below all regulatory limits.

Table 10-11a. Public Doses From Routine Operations at MRS and Repository (DOE86a)

Routine Activity	Annual Offsite Population Dose (person-rem/1,000 MTU)	RMEI (rem/yr)
• MRS-Routine Release	0.1*	< 0.001
• Repository-Routine Release	< 0.1*	< 0.001
• Transportation-Normal		<0.005
- Reactors to MRS	64	
- MRS to Repository	16	

*Population dose assumes storage only without fuel consolidation.

Table 10-11b. Public Doses from Accidental Releases at MRS and Repository (DOE86a)

Accident Events	Annual Offsite Population Dose (person-rem/yr)	RMEI (rem/yr)
• MRS Facility		
- Fuel Assembly Drop	0.03	0.004
- Shipping Cask Drop	0.006	0.0009
- Storage Cask Drop	0.006	0.0009
• Repository	N/A	N/A

10.4.9 "The Safety Evaluation of Tunnel Rack and Dry Well Monitored Retrievable Storage Concepts" (LIG83)

This study evaluated the safety of dry rack storage of SNF at an MRS facility. Only the radiological risks to members of the public due to select accidents during MRS operations were addressed. The MRS facility was assumed: (1) to have a throughput of 900 MTU per year; (2) to receive SNF for 36 years; and (3) to store the fuel for 100 years before it was shipped to a repository for permanent disposal.

In total, the study analyzed 15 different accident scenarios that included transportation collision during emplacement/retrieval of SNF, canister drops during emplacement/retrieval, an airplane crash with and without fire, and pin failure resulting from seismic events. Accident frequencies for these scenarios ranged from a low of 4E-10 per year (plane crash) to 1.4E-01 per year (canister drop during retrieval).

For all 15 accident scenarios combined, the annual population dose risk was estimated to be 17 person-rem for the 0-50 mile population.

10.4.10 Summary Assessment of Available Data

This chapter of the BID has presented information from a variety of studies that provided quantitative estimates of population doses from future fuel management activities that may be conducted at individual reactor sites, at a centralized monitored retrievable storage facility, and at a deep geologic repository.

Because the MRS Review Commission Study provided data for all three facility types, it was regarded as the reference study. Table 10-12 presents population doses for a 50-mile area around a given facility based on normalized Commission data.

Table 10-12. Normalized Population Doses (Based on data from MRS89)

Facility	Person-rem/1,000 MTU-yr
At-Reactor Storage	7.8E-02
MRS (Eastern U.S. location)	3.8E-02
Repository (at Yucca Mountain)	< 9.6E-05

While acknowledging the much lower population dose associated with deep geologic disposal, the Commission concluded that short-term spent fuel storage at reactor facilities or at an MRS facility, albeit higher, yielded population doses that were safe. In association with this conclusion, the Commission offered the following summary statements:

- *... spent fuel management operations have been safely carried out at reactors for many years under NRC regulatory control and by trained personnel.... Although the inventory of spent fuel at reactors is increasing there is no reason to believe that safe management of fuel cannot continue or that the fuel will interfere significantly with safe reactor operations.*

- *It appears that most, if not all, reactor sites can safely store all of the spent fuel that would be generated during the reactor's 40-year operation life. ... This storage can be expanded as necessary to meet life-of-plant storage requirements.... At most sites, life-of-plant* *storage can be accomplished by reracking spent fuel pool and using dry storage.*

- *From a technical perspective, both the No-MRS and MRS options are safe...Although neither option is completely without risk, ...the risks are expected to be small and within regulatory limits, and the degree of difference in risks is so small that the magnitude of difference should not affect the decision as to whether there should be an MRS.*

The views of the MRS Review Commission are generally supported by other data. Radiological dose values from various other studies were reviewed and are summarized in Table 10-13.

As indicated by the brief overviews presented, the studies varied widely in scope, facility designs, primary assumptions regarding SNF processing, waste packaging, accident scenarios, modeling approaches, and other factors.

Table 10-13 illustrates the risk categories that have been addressed and the forms or units in which the data were reported. Although an attempt was made to present results in common units for unbiased comparison, necessary data were frequently lacking. For example, some studies presented accident risk in terms of resultant dose without specifying the probability of occurrence for a given accident scenario. In other instances, 50-mile population doses were cited for the entire preclosure period of a facility with full awareness that SNF inventories were variable/accumulating throughout that period. In the absence of detailed inventory data, a representative annual population dose could not be determined.

Table 10-13. Summary Data of Public Doses Associated with SNF Storage At-Reactor, MRS, and Repository

	At-Reactor				MRS				Repository			
	Population Dose		RMEI		Population Dose		RMEI		Population Dose		RMEI	
Study	Routine	Accident	Routine	Accident	Routine	Accident	Routine	Accident	Routine	Accident	Routine	Accident
Bechtel (1979)									1.5E-06[a]	1.1E-02[a]		
DOE (1980)									negligible		negligible	1.1E-04[f]
Orvis (1984)		2.3E+00[c]										
NRC (1984)	negligible[a]			1.0E-06[e]								
DOE (1985)	negligible[a]		6.0E-08[c]									
DOE (1986a)									3.9E+02[b]	3.0E+03[f]	5.6E-03[e]	4.7E-02[f]
Schneider (1986)	1.0E+00[a]	6.0E-04[c]							6.0E+00[a]			
DOE (1986b)					1.0E-01[a]	3.0E-02[c]	1.0E-03[c]	4.0E-03[e]	1.0E-01[a]		1.0E-03[c]	
Ligon (1983)						1.7E+01[c]						
MRS Review Commission (1989)	7.8E-02[a]				3.8E-02[a]				9.6E-05[a]			

Population Doses:
a = person-rem/1,000 MTU-yr
b = person-rem/SNF inventory-yr (see text)
c = person-rem/yr

RMEI Doses:
d = Rem/1,000 MTU-yr
e = Rem/yr
f = Rem/event

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