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Integration of the Back End of the Nuclear Fuel Cycle

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ABSTRACT

Management of spent nuclear fuel and high-level radioactive waste consists of three main phases – storage, transportation, and disposal – commonly referred to as the back end of the nuclear fuel cycle. Current practice for commercial spent nuclear fuel management in the United States (US) includes temporary storage of spent fuel in both pools and dry storage systems at operating or shutdown nuclear power plants. Storage pools are filling to their operational capacity, and management of the approximately 2,200 metric tons of spent fuel newly discharged each year requires transferring older and cooler spent fuel from pools into dry storage. Unless a repository becomes available that can accept spent fuel for permanent disposal, projections indicate that the US will have approximately 136,000 metric tons of spent fuel in dry storage systems by mid-century, when the last plants in the current reactor fleet are decommissioned.

Current designs for dry storage systems rely on large multi-assembly canisters, the most common of which are so-called "dual-purpose canisters" (DPCs). DPCs are certified for both storage and transportation, but are not designed or licensed for permanent disposal. The large capacity (greater number of spent fuel assemblies) of these canisters can lead to higher canister temperatures, which can delay transportation and/or complicate disposal. This current management practice, in which the utilities continue loading an ever-increasing inventory of larger DPCs, does not emphasize integration among storage, transportation, and disposal. This lack of integration does not cause safety issues, but it does lead to a suboptimal system that increases costs, complicates storage and transportation operations, and limits options for permanent disposal.

This paper describes strategies for improving integration of management practices in the US across the entire back end of the nuclear fuel cycle. The complex interactions between storage, transportation, and disposal make a single optimal solution unlikely. However, efforts to integrate various phases of nuclear waste management can have the greatest impact if they begin promptly and continue to evolve throughout the remaining life of the current fuel cycle. A key factor that influences the path forward for integration of nuclear waste management practices is the identification of the timing and location for a repository. The most cost-effective path forward would be to open a repository by mid-century with the capability to directly dispose of DPCs without repackaging the spent fuel into disposal-ready canisters. Options that involve repackaging of spent fuel from DPCs into disposal-ready canisters or that delay the repository opening significantly beyond mid-century could add 10s of billions of dollars to the total system life cycle cost.

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ACRONYMS AND DEFINITIONS

Abbreviation	Definition
AAR	Association of American Railroads
AEC	Atomic Energy Commission
ANA	advanced neutron absorber
ANL	Argonne National Laboratory
ANS	American Nuclear Society
ATF	accident tolerant fuel
BAM	Bundesanstalt für Materialforschung und -prüfung (Germany)
BATS	Brine Availability Test in Salt
BRC	Blue Ribbon Commission
BWMi	Federal Ministry for Economic Affairs and Energy (Germany)
BWR	boiling water reactor
CFR	Code of Federal Regulations
CIEMAT	Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (Spain)
CIRFT	cyclic integrated reversible fatigue tester
CISF	consolidated interim storage facility
CoC	certificate of compliance
CSFP	US Commercial Spent Fuel Projection tool
CSNF	commercial spent nuclear fuel
CHLW	commercial high-level radioactive waste
DCSS	dry cask storage system
DHC	delayed hydride cracking
DHLW	DOE-managed high-level radioactive waste
DOE	US Department of Energy
DOE-NE	DOE Office of Nuclear Energy
DOT	US Department of Transportation
DPC	dual-purpose canister
DRZ	disturbed rock zone
DSNF	DOE-managed spent nuclear fuel
DTS	dry transfer system
EBS	engineered barrier system
EIA	US Energy Information Administration
ELEA	Eddy-Lea Energy Alliance

Abbreviation	Definition
ENSA	Equipos Nucleares Sociedad Anónima (Spain)
ENUN	ENSA UNiversal cask
ENUSA	Empresa Nacional del Uranio Sociedad Anónima (Spain) (now ENUSA Industrias Avanzadas)
EPA	US Environmental Protection Agency
EPRI	Electric Power Research Institute
ERDA	Energy Research and Development Administration
FEP	feature, event, and process
FY	fiscal year
GDSA	Geologic Disposal Safety Assessment
GE	General Electric
GTCC	Greater Than Class C (radioactive waste)
GWd	gigawatt-days
HAC	hypothetical accident conditions
HIP	hot isostatic pressing
HLW	high-level radioactive waste
HPC	high-performance computing
IAEA	International Atomic Energy Agency
IFBA	Integral Fuel Burnable Absorber
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
ISFSI	Independent Spent Fuel Storage Installation
ISP	Interim Storage Partners
IWM	DOE-NE Integrated Waste Management
kWh	kilowatt-hours
LLW	low-level radioactive waste
LWR	light-water reactor
LWT	legal-weight truck
MCO	multicanister overpack
MIT	Massachusetts Institute of Technology
MMTT	Multi-Modal Transportation Test
MOX	mixed oxide (fuel)
MPC	multi-purpose canister
MRS	Monitored Retrievable Storage
MTHM	metric tons of heavy metal

Abbreviation	Definition
MTU	metric tons of uranium
Nagra	National Cooperative for the Disposal of Radioactive Waste (Switzerland)
NAS	National Academy of Sciences
NCT	normal conditions of transport
NEA	Nuclear Energy Agency
NEI	Nuclear Energy Institute
NEUP	Nuclear Energy University Program
NFST	DOE-NE Nuclear Fuel Storage and Transportation Planning Project
NGSAM	Next-Generation System Analysis Model
NNPP	Naval Nuclear Propulsion Program
NNSS	Nevada National Security Site
NPP	nuclear power plant
NRC	US Nuclear Regulatory Agency
NSNFP	National Spent Nuclear Fuel Program
NTS	Nevada Test Site
NWMO	Nuclear Waste Management Organization (Canada)
NWPA	Nuclear Waste Policy Act of 1982
NWPAA	Nuclear Waste Policy Amendments Act of 1987
NWTRB	US Nuclear Waste Technical Review Board
OCRWM	Office of Civilian Radioactive Waste Management
ORNL	Oak Ridge National Laboratory
OWL	Online Waste Library
PA	performance assessment
PCT	peak cladding temperature
PFS	Private Fuel Storage consortium
PIRT	Phenomena Identification and Ranking Table
PNNL	Pacific Northwest National Laboratory
PWR	pressurized water reactor
R&D	research and development
RCT	Ring Compression Test
RSSF	Retrievable Surface Storage Facility
SCC	stress corrosion cracking
SFOE	Swiss Federal Office of Energy
SFWST	DOE Office of Spent Fuel and Waste Science and Technology
SNF	spent nuclear fuel

Abbreviation	Definition
SNL	Sandia National Laboratories
SRS	Savannah River Site
SSC	structure, system, and component
STAD	standardized transportation, aging, and disposal (canister)
TAD	transportation, aging, and disposal (canister)
THCM	thermal-hydrologic-chemical-mechanical
THM	thermal-hydrologic-mechanical
TMI-2	Three Mile Island Unit 2 reactor
TRU	transuranic
TSLCC	total system life cycle cost
TSPA	total system performance assessment
UO_2	uranium oxide (fuel)
UQ/SA	uncertainty quantification and sensitivity analysis
URL	underground research laboratory
US	United States
USGS	US Geological Survey
W	watts
WCS	Waste Control Specialists
WIPP	Waste Isolation Pilot Plant
WTP	Waste Treatment and Immobilization Plant (Hanford Vitrification Plant)

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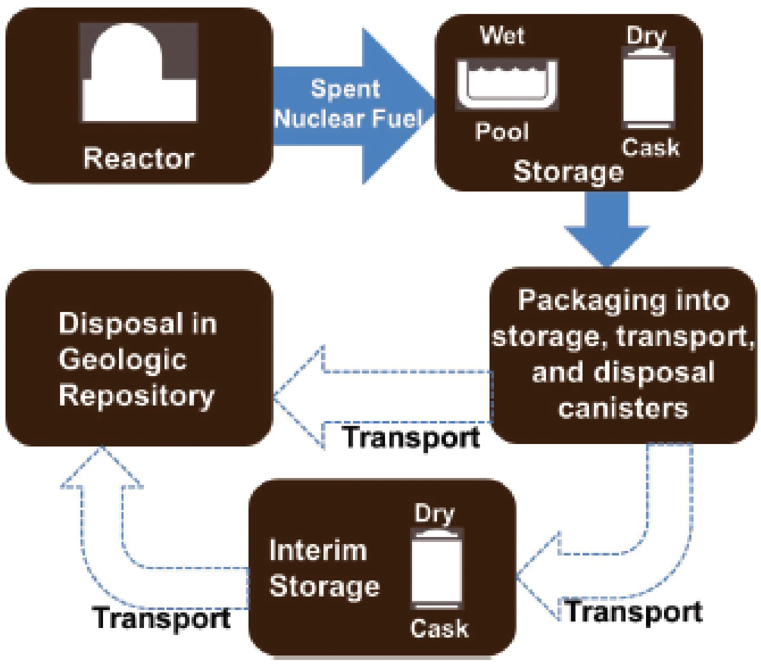
1. INTRODUCTION

Nuclear waste has been generated in the United States (US) since the early 1940s, first as a byproduct of nuclear weapons research and production, and later as a byproduct of the civilian nuclear power industry (Bonano et al. 2018; BRC 2012, Chapter 3.4). Management of nuclear waste, in the form of spent nuclear fuel (SNF)[1] and high-level radioactive waste (HLW)[2], consists of three main phases ─ storage, transportation, and disposal ─ commonly referred to as the back end of the nuclear fuel cycle. Figure 1-1 illustrates the potential stages of the fuel cycle for commercial nuclear power generation.



Note: The back end of the nuclear fuel cycle includes storage (wet and dry), transportation, and disposal.

(Source: NWTRB 2017a, Figure 1-1)

Figure 1-1. Nuclear Fuel Cycle for Commercial Power Generation

The remainder of this section summarizes the evolution of nuclear waste management in the US, from the initial passage of the Atomic Energy Act of 1946, through the revision to the Atomic Energy Act in 1954, the Energy Reorganization Act of 1974, the Nuclear Waste Policy Act

[1] "spent nuclear fuel" is defined as "fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing" (NWPA 1983, Sec. 2(23)).

[2] "high-level radioactive waste" is defined as "(A) the highly radioactive material resulting from the reprocessing of spent nuclear fuel, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (B) other highly radioactive material that the Commission, consistent with existing law, determines by rule requires permanent isolation" (NWPA 1983, Sec. 2(12)).

(NWPA) of 1982, the amendments to the NWPA in 1987, and up to the current federal policies and nuclear waste management strategy.

Section 2 identifies the current and projected future inventory of nuclear waste in the US and Section 3 summarizes the current practices for storage, transportation, and disposal in the US. Section 4 discusses the technical implications of these current practices and the regulatory, legal, and stakeholder considerations for the management of the back end of the nuclear fuel cycle. Section 5 identifies ongoing and future activities that could contribute to more efficient integration of the back end of the nuclear fuel cycle and introduces spent fuel management strategies and comparative costs for integrating storage, transportation, and disposal going forward. Section 6 provides a summary and conclusions.

Bonano et al. (2018) provides a good high-level summary of the needs and potential options for integration of the back end of the nuclear fuel cycle. As a result, certain parts of the Bonano et al. (2018) paper are excerpted, without citation, in this report. In addition, this report (1) provides further details and background on spent nuclear fuel management needs and options, and (2) identifies specific activities that could contribute to more efficient integration of the back end of the nuclear fuel cycle.

1.1 History of Nuclear Waste Management in the US

In the 1940s, during the early days of nuclear weapons development in the US, national security considerations were a very high priority. With an emphasis on rapid production of plutonium for use in weapons, storage in large, underground steel tanks was deemed adequate as an interim means (but not a long-term solution) of isolating the highly radioactive liquid waste that remained after acid was used to dissolve irradiated nuclear fuel as part of the plutonium separation process (BRC 2012, Chapter 3.4.1).

Following the end of World War II, the Atomic Energy Act of 1946 led to the establishment of the US Atomic Energy Commission (AEC) in 1947 for the peacetime development of atomic science and technology. This included the development, production, and control of atomic energy, and the associated responsibility for ownership and ultimate disposition of fissionable material and nuclear waste (Buck 1983; Rechard et al. 2015, Section 3.1). A revision to the Atomic Energy Act in 1954 enabled the AEC to encourage private industry development of nuclear power plants (NPPs), using fissionable material leased from the federal government, and gave the AEC responsibility for licensing and regulating commercial atomic activities (Buck 1983).

By the time the first commercial power plant (Shippingport) came into service in the US in 1957, there was a growing recognition that deep geologic disposal would be the best available option for permanently isolating highly radioactive wastes (NAS 1957). Every nation that has pursued nuclear power has subsequently come to the same conclusion ─ deep geologic disposal is the preferred option for isolating SNF and HLW (NEA 1999; NAS 2001; NWTRB 2011; BRC 2012).

At the onset of the commercial nuclear power industry, the US government supported the commercialization of SNF reprocessing because it was generally accepted that uranium resources were limited and would become increasingly expensive (MIT 2011, Chapter 2). Therefore, for most of the early history of commercial nuclear power in the US, including the period of rapid expansion in the late 1960s and 1970s, electric utility companies operated reactors with an expectation that SNF would be reprocessed to recover unused fissionable material (^{235}U and ^{239}Pu) for use in new reactor fuel (Bonano et al. 2018). HLW produced from reprocessing would be immobilized in borosilicate glass (e.g., a standardized waste form) and placed in standardized stainless-steel canisters (Rechard et al. 2015, Section 3.1).

With government support, a small privately-owned reprocessing plant at West Valley, New York began operations in 1966, but was shut down in 1972 for process improvements and never reopened because of increased costs and newly imposed regulatory requirements (MIT 2011, Chapter 2; Rechard et al. 2015, Section 3). General Electric (GE) started construction on a small demonstration reprocessing plant at Morris, Illinois in 1967. Due to problems identified during testing it was never operational, but it still has some SNF in a licensed pool storage facility. A larger commercial plant at Barnwell, South Carolina was constructed in the mid-1970s but never accepted or reprocessed SNF and is now closed (MIT 2011, Chapter 2).

In 1974, India detonated a nuclear explosive, and non-proliferation considerations led to concerns that reprocessing technology could be used to separate ^{239}Pu for use in nuclear weapons. This led President Ford to announce "deferral" of commercial reprocessing in 1976 which was extended indefinitely by President Carter in 1977 (MIT 2011, Chapter 2; Vandenbosch and Vandenbosch 2007). At about the same time, new uranium discoveries indicated that global uranium resources were not nearly as limited as was previously thought and the costs of reprocessing were higher than originally estimated. Although the decision to defer reprocessing was later reversed by President Reagan in 1981 (Reagan 1981), the confluence of non-proliferation and economic factors led the US to adopt a "once-through" nuclear fuel cycle where SNF is not recycled (MIT 2011, Chapter 2), and no new reprocessing facilities have been constructed in the US since then (Bonano et al. 2018).

In parallel with, but independent from, the reprocessing decision, there was criticism that the AEC regulated the very same energy source that it helped to produce and operate. Also, the energy crisis of 1973 contributed to a recommendation by President Nixon to coordinate all major national energy programs in a new agency, whose primary purpose would be to assist the US in achieving energy independence. This led to the Energy Reorganization Act of 1974. As a result, in 1975, the AEC's research and development (R&D) responsibilities became part of the newly formed Energy Research and Development Administration (ERDA) and the AEC's regulatory and licensing functions were assumed by the new Nuclear Regulatory Commission (NRC) (Buck 1983). Then, in 1977, the Department of Energy (DOE) was formed and became the successor to ERDA.

In 1982, recognizing the need for a national policy regarding wastes from both commercial and defense-related nuclear enterprises, Congress passed the NWPA (NWPA 1983). The NWPA assigned responsibility for permanent disposal of all SNF and HLW to the federal government, required the DOE to evaluate multiple repository sites and to license and construct a disposal facility that would begin operations in 1998, assigned the EPA to set health protection standards, and assigned regulatory and licensing responsibilities to the NRC. The first repository would be limited to a disposal inventory of 70,000 metric tons of heavy metal (MTHM), either in the form of SNF or an equivalent quantity of HLW, and, as prescribed in the NWPA (1983, Section 114(a)), a second repository would be licensed and constructed three years after the first to accommodate additional wastes. The NWPA (1983, Section 304) established the Office of Civilian Radioactive Waste Management (OCRWM) within the DOE to "be responsible for carrying out the functions of the Secretary under this Act".

The NWPA provided for the development of repositories for the disposal of SNF and HLW from "civilian nuclear activity" (i.e., commercial SNF (CSNF)). More precisely, the NWPA applies to any repository not used exclusively for the disposal of SNF and HLW from "atomic energy defense activities, research and development activities for the Secretary, or both" (NWPA 1983, Section 8(c)). However, the NWPA also states that one or more civilian repositories should be used for the disposal of HLW from "atomic energy defense activities" (i.e., defense HLW) unless the President finds that a repository for the disposal of HLW from "atomic energy defense activities only is required" based on an evaluation of six factors: "cost efficiency, health and safety, regulation, transportation, public acceptability, and national security" (NWPA 1983, Section 8(b)).

In 1985, President Reagan determined that a separate defense HLW repository was not needed, based on analysis that showed that there could be large cost savings (~$1.5 billion) to using the civilian NWPA repository for the defense wastes, and that no other factors distinguished significantly between the options (Rechard et al. 2015, Section 2.3.4).

The NWPA (1983) also states:

- in Section 111(a)(5) that waste generators and owners (i.e., the nuclear utilities) "have the primary responsibility to provide for, and the responsibility to pay the costs of, the interim storage" of the waste until it is "accepted by the Secretary of Energy" and at Section 111(a)(4) "the Federal Government has the responsibility for the permanent disposal" of waste,
- in Section 302(a)(1) "the Secretary is authorized to enter into contracts with any" waste generator or owner "for the acceptance of title, subsequent transportation, and disposal of" the waste, and at Section 302(a)(2) "Such contracts shall provide for payment to the Secretary of fees … for electricity generated … 1.0 mil per kilowatt-hour" ($0.001/kWh), and
- in Section 302(a)(5)(A) "following commencement of operation of a repository, the Secretary shall take title to the high-level radioactive waste or spent nuclear fuel involved as expeditiously as practicable ..., and at Section 305(a)(5)(B) "in return for the payment of fees established by this section, the Secretary, beginning not later than January 31, 1998, will dispose of the high-level radioactive waste or spent nuclear fuel ...".

These provisions of the NWPA (1) provide funding for waste management through a Nuclear Waste Fund supported by a fee on nuclear power borne by the ratepayers, and (2) establish Standard Contracts (codified in 10 CFR Part 961) between the federal government (i.e., the DOE) and nuclear utilities, whereby utilities would load CSNF into DOE-provided casks, and DOE would commence repository operations and begin taking title to the waste for transportation and disposal no later than January 31, 1998.

The NWPA (1983, Section 141) identifies long-term storage as an option for management of SNF and HLW and directs the federal government to evaluate the need for, and feasibility of, one or more federally-sited Monitored Retrievable Storage (MRS) facilities, but not in a state under consideration for a repository.

In 1987, with the Nuclear Waste Policy Amendments Act (NWPAA) (NWPAA 1987), Congress amended the NWPA to identify Yucca Mountain, Nevada, as the only site for further evaluation as a repository, to defer action on a second repository, and to preclude site-specific activities associated with any location other than Yucca Mountain without specific Congressional authorization.

The NWPAA, also put restrictions on the development of federal away-from-reactor storage facilities, under the amended provisions for MRS. Specifically, the NWPAA provides the following licensing conditions for an MRS (NWPAA 1987, Section 148(d)):

- "(1) construction of such facility may not begin until the Commission has issued a license for the construction of a repository under Section 115(d)" [corrected to Section 114(d) by footnote],
- "(2) construction of such facility or acceptance of spent nuclear fuel or high-level radioactive waste shall be prohibited during such time as the repository license is revoked by the Commission or construction of the repository ceases,
- "(3) the quantity of spent nuclear fuel or high-level radioactive waste at the site of such facility at any one time may not exceed 10,000 metric tons of heavy metal until a repository under this Act first accepts spent nuclear fuel or solidified high-level radioactive waste", and
- "(4) the quantity of spent nuclear fuel or high-level radioactive waste at the site of such facility at any one time may not exceed 15,000 metric tons of heavy metal."

These provisions restrict construction of a federal interim storage facility until a license for construction of a repository has been issued, ensuring that the interim storage facility does not become a de facto disposal site in the absence of a repository, or delay progress towards a repository decision.

The NWPAA (1987, Section 503) also established the Nuclear Waste Technical Review Board (NWTRB) as an independent entity to "evaluate the technical and scientific validity of activities undertaken by the Secretary after the date of the enactment of the Nuclear Waste Policy Amendments Act of 1987."

While repository development focused on Yucca Mountain, CSNF was starting to accumulate at reactor sites. It was originally expected that spent fuel, upon being removed from the reactor core, would be held in pools at nuclear power plants until it had cooled sufficiently and then transported for reprocessing or disposal. In the early years of the commercial nuclear power industry, on-site pool storage times were anticipated to be on the order of one year or less before spent fuel would be sent for reprocessing, and most at-reactor storage pools were originally designed to hold one full core plus one or two refueling discharges (NRC 2014a, Section 2.1.2.1). However, by the late 1970s when reprocessing ceased to be a national policy objective, utilities began reconfiguring reactor pools and storage practices to accommodate substantially more spent fuel assemblies. As pools approached the revised capacity limits, utilities moved forward with implementing on-site dry storage systems, the first of which (at Surry) was loaded in 1986.

By the early 1990s, as it became clear that the proposed Yucca Mountain repository would not open in 1998, utilities began planning for greater on-site dry storage capacity. When the DOE failed to open the repository in 1998 and did not begin taking title of CSNF, as required by the NWPA, the utilities sued the DOE for breach of contract and were eventually awarded ongoing damages and penalties associated with their need to maintain on-site storage. These payments to the utilities come from the US Treasury via a "Judgment Fund" (31 CFR Part 256; 31 CFR Section 1304) rather than from the Nuclear Waste Fund and are a taxpayer liability that continues to the present day.

In 2008, DOE submitted a License Application for Yucca Mountain to the NRC (Sproat 2008; DOE 2008a; DOE 2009a[3]), twenty years after the date envisioned in the 1982 law, but the DOE withdrew support for the project in 2010, and Congress suspended funding for the licensing process.

1.2 Current Strategy for Nuclear Waste Management in the US

The provisions of the NWPA, as amended, remain in effect today, but all of what the law envisioned has not yet come to pass. The separation of responsibilities in the NWPA (at-reactor interim storage of CSNF by the utilities; transportation and disposal of CSNF by the DOE) makes integration inherently difficult. Utilities, driven by economic and worker safety considerations, favor loading CSNF into large multi-assembly canisters, the most common of which are so-called "dual-purpose canisters" (DPCs). However, the large capacity of DPCs can lead to higher canister temperatures, which can delay transportation and/or complicate disposal. Furthermore, DPCs are not designed or licensed for permanent disposal.

The suspension of the Yucca Mountain licensing process, and the associated termination of funding for OCRWM, led to various proposals for updated strategies for managing the back end of the nuclear fuel cycle.

In 2011, the Massachusetts Institute of Technology (MIT) published a report identifying important near-term decisions that could have far reaching long-term implications about the evolution of the nuclear fuel cycle – what type of fuel is used, what types of reactors, what happens to irradiated fuel, and what method of disposal is used for long-term nuclear wastes (MIT 2011). General conclusions related to the back of the nuclear fuel cycle included:

- **Fuel Cycle Option** – For the next several decades, a "once-through" or open fuel cycle using light-water reactors (LWRs) is the preferred economic option for the US and is likely to be the dominant feature of the nuclear energy system for much of this century (MIT 2011, Executive Summary). The once-through LWR fuel cycle is summarized in Appendix A.
- **Storage** – Planning for long-term managed storage of SNF – for about a century – should be an integral part of nuclear fuel cycle design. Managed storage can be done safely at operating reactor sites, centralized storage facilities, or geological repositories designed for retrievability (an alternative form of centralized storage). The possibility of storage for a century, which is longer than the anticipated operating lifetimes of nuclear reactors, suggests

[3] DOE initially submitted the License Application in June 2008. Updates to certain Sections were submitted in February 2009.

that the US should move toward centralized SNF storage sites – starting with SNF from decommissioned reactor sites. The federal government should take ownership of the SNF under centralized storage. (MIT 2011, Executive Summary and Chapter 1).

- **Disposal** – Permanent geological isolation will be required for at least some long-lived components of SNF, and so systematic development of a geological repository needs to be undertaken. Furthermore, the US currently has significant inventories of defense HLW and small quantities of commercial HLW glass that are ready for geological disposal. (MIT 2011, Chapter 4).

In 2012, President Obama's Blue Ribbon Commission on America's Nuclear Future (BRC) was chartered to recommend a new strategy for managing the back end of the nuclear fuel cycle. The recommended strategy has eight key elements, three of which are directly relevant to the integration between storage, transportation, and disposal (BRC 2012, p. vii):

#4: Prompt efforts to develop one or more geologic disposal facilities.
#5: Prompt efforts to develop one or more consolidated storage facilities.
#6: Prompt efforts to prepare for the eventual large-scale transport of spent nuclear fuel and high-level waste to consolidated storage and disposal facilities when such facilities become available

In 2013, consistent with BRC recommendations (and similar to the MIT recommendations[4]), DOE published a strategy for the management and disposal of SNF and HLW (DOE 2013). The DOE strategy outlined a long-term plan for interim storage (a pilot interim storage facility operating by 2021, with an initial focus on stranded (orphaned) waste, and a subsequent full-scale interim storage facility, available by 2025), transportation (options for SNF, including from shutdown reactors), and disposal (a geologic repository available by 2048). The strategy was intended to provide a basis for the Administration to work with Congress to design and implement a program to meet the government's obligation to take title to and permanently dispose of SNF and HLW (DOE 2013).

Implementation of the DOE strategy was assigned within the DOE Office of Nuclear Energy (DOE-NE) to the Used Fuel Disposition (UFD) Campaign (Swift et al. 2015) and Nuclear Fuel Storage and Transportation (NFST) Planning Project (Wagner et al. 2015). However, full implementation of the DOE strategy was contingent on (1) appropriate authorizations from Congress, and (2) enactment of new legislation or revision/amendment to the NWPA. Congress has appropriated some annual funding, but no new or revised legislation has been enacted. At the start of Fiscal Year (FY) 2017, DOE-NE was re-organized and UFD became the Office of Spent Fuel and Waste Science and Technology (SFWST) and NFST became the Office of Integrated Waste Management (IWM).

As a result of the 1985 finding of President Reagan (see Section 1.1), the Yucca Mountain Repository License Application (DOE 2008a) for a NWPA repository included both commercial and DOE-managed waste. However, in 2014, DOE performed an assessment of options for the permanent disposal of DOE-managed SNF and HLW (DOE 2014), which included consideration of whether DOE-managed SNF and HLW should be commingled with, or separate from,

[4] Contributors to the MIT report (MIT 2011) and members of its advisory committee also served on the BRC.

commercial SNF and HLW. The report recommended that "DOE pursue options for disposing of some DOE-managed HLW and SNF separately from commercial SNF and HLW" (DOE 2014, Section 3.4). This recommendation was confirmed in DOE (2015), which concluded that, "A geologic repository for permanent disposal of Defense HLW could be sited, licensed, constructed, and operated more quickly than a Common NWPA Repository and would provide valuable experience to reduce the cost of a future repository and the time needed to develop it. In consideration of the six statutory factors cumulatively, this report concludes that a strong basis exists to find that a Defense HLW Repository is required."

Based on the conclusion in DOE (2015), President Obama issued a finding that "the development of a repository for the disposal of high-level radioactive waste resulting from atomic energy defense activities only is required" (Obama 2015). However, work on the Defense HLW repository is no longer funded.

In 2015, the NRC staff completed its review of the Yucca Mountain Repository License Application by issuing the five-volume Safety Evaluation Report (NRC 2015), concluding that, although certain land ownership and water rights requirements have not yet been met, the repository meets its requirements for preclosure and postclosure safety.

However, as of the date of publication of this report, the Yucca Mountain repository licensing process remains suspended, Congress has made no change to the law, and the US government is considering no other sites for disposal. The DOE has still not begun to take title to CSNF, which continues to accumulate in dry storage at commercial reactor sites, and taxpayers continue to pay utilities for costs of on-site management of CSNF. The trend of the last decade for older and less profitable nuclear power plants to close seems likely to continue, leaving increasing quantities of CSNF stored at sites where there is no operating reactor.

Consolidated interim storage, which could accommodate some CSNF from both decommissioned and operating sites, has been contemplated. However, the viability of a federal site is uncertain without modifications to portions of the NWPA that link federal interim storage to repository licensing and operation; private sector sites are not yet NRC-licensed and their economic viability is uncertain.

If Congress chooses to restart the licensing process for Yucca Mountain, extrapolation from DOE's 2009 projection of a 2020 date for first disposal operations (DOE 2009a) indicates that disposal operations are unlikely to begin any sooner than eleven years after hearings resume, assuming issuance of a construction authorization by the NRC and full Congressional funding. If Congress chooses to permanently abandon the Yucca Mountain site and pursue as-yet-unidentified alternatives, which would require revisions or amendments to the NWPA, a repository would not likely be available before mid-century.

In summary, more than six decades after the beginning of commercial nuclear power, the US still does not have an operating deep geologic disposal facility for commercial or DOE-managed SNF or HLW. The current management practice, in which the utilities continue loading an ever-increasing inventory of larger DPCs, does not emphasize integration among storage, transportation, and disposal. This lack of integration does not cause safety issues, but it does lead to a suboptimal system that increases costs, complicates storage and transportation operations, and limits options for permanent disposal. Furthermore, the future is uncertain, with few options available other than continued at-reactor storage until such time as the Yucca Mountain Project is restarted and licensed or the NWPAA is further amended or replaced by new legislation that would allow the siting of one or more geologic repositories.

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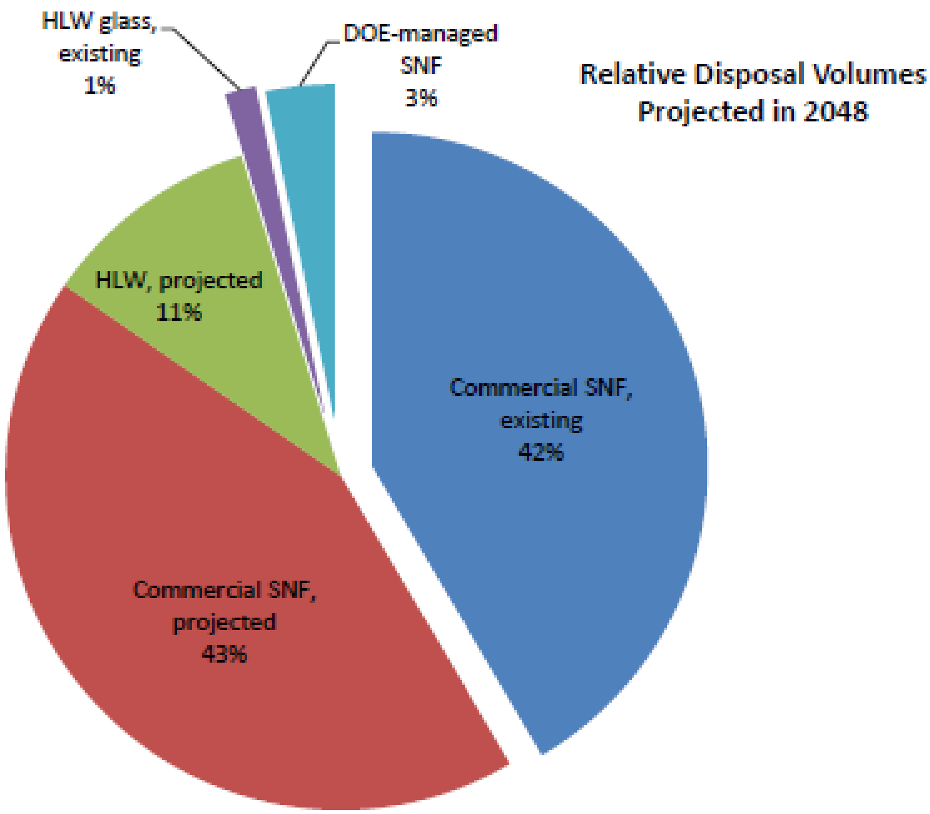
2. INVENTORY OF SNF AND HLW IN THE US

The US inventory of SNF and HLW includes nuclear waste from commercial nuclear power generation, generally categorized as CSNF (although there is also a small amount of HLW from commercial reprocessing (i.e., CHLW) at the shutdown West Valley plant)[5] (see Section 2.2), and from defense-related or other government-sponsored nuclear enterprises, generally categorized as DOE-managed (see Section 2.3). DOE-managed waste can be further categorized as DOE-managed SNF (DSNF) or DOE-managed HLW (DHLW), sometimes also called defense HLW.

As a result of the 1985 finding of President Reagan (see Section 1.1), the Yucca Mountain Repository License Application (DOE 2008a) included both commercial and DOE-managed waste. The commingling decision was re-examined in DOE (2014) and DOE (2015) and subsequently reversed by the finding of President Obama (Obama 2015). In either case, both commercial waste and DOE-managed waste requires a defined disposal path.

Recent (as of 2014) and projected (to 2048) inventories for commercial and DOE-managed SNF and HLW are shown in Figure 2-1.



(Source: SNL 2014a, Figure ES-2)

Figure 2-1. Relative Projected Volumes of US SNF and HLW in 2048 (from 2014 Data)

[5] This categorization of the DOE-managed West Valley HLW as CHLW is consistent with Peters et al. (2020) and with the Yucca Mountain waste allocations (see Section 3.3.2). Other documents (e.g., DOE 2014) categorize the West Valley HLW as DHLW.

Although the inventories shown in Figure 2-1 are based on data from 2014, the fact that 85% of the total projected volume of US SNF and HLW in 2048 is CSNF prioritizes the focus on CSNF as a key part of the integration of the back end of the nuclear fuel cycle. Updated current and projected inventories for commercial and DOE-managed SNF and HLW are summarized in the following subsections.

Given the preponderance of the CSNF inventory over defense HLW and defense SNF, the focus of this paper will be on CSNF generated from commercial nuclear power reactors. For completeness, however, all SNF and HLW will be discussed from an inventory and current status basis. In addition, considerations for spent fuel from alternative/advanced fuel cycles and accident tolerant fuels (ATF) are discussed where appropriate, with the acknowledgement that, under current plans, they would represent only a small portion of the projected CSNF inventory.

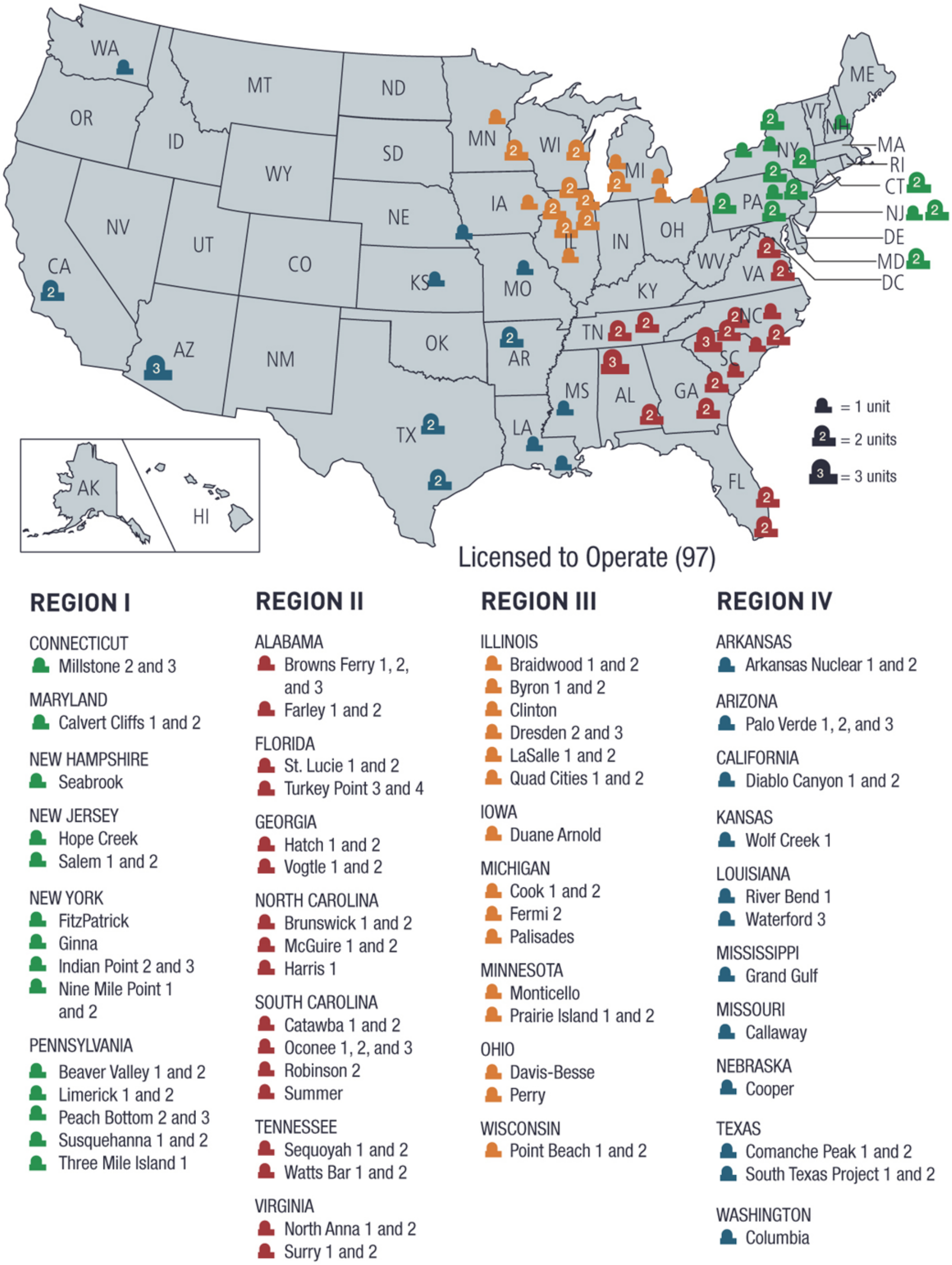
2.1 Commercial Nuclear Power Plants

As of December 2020, 94 commercial nuclear reactors were licensed and operating in the US (under 10 CFR Part 50) at 56 sites in 28 states (Figure 2-2 and Appendix B, Table B-1). These "counts" include the recent shutdowns of Pilgrim 1 in May 2019, Three Mile Island 1 in September 2019, Indian Point 2 in April 2020, and Duane Arnold in August 2020. All 94 of these NPPS are LWRs, 63 are pressurized water reactors (PWR) and 31 are boiling water reactors (BWR). Of the 94 operating reactors, 82 have received a first 20-year license renewal (all of which expire between 2025 and 2050), 4 (Peach Bottom 2 and 3, Turkey Point 3 and 4) have received a second "subsequent" 20-year license renewal (all expiring between 2052 and 2054), 4 (Comanche Peak 1 and 2, Perry, and Clinton) have announced intentions to submit renewal applications between 2022 and 2024, 2 (Diablo Canyon 1 and 2) have plans to shut down before a renewal is needed, and the 2 newest reactors (Watts Bar 1 and 2) have yet to announce an intention (Appendix B, Table B-1). In addition, two new reactors, Vogtle 3 and 4, are under construction at an existing site.

However, decisions to proceed with other new builds (Appendix B), and further license extensions, are currently constrained by economics (e.g., lower costs for natural gas and coal plants, the lack of carbon emission incentives) (MIT 2011, Chapter 1) and by the lack of a repository.

In addition to the 94 operating commercial reactors, there are also 37 shutdown reactors as of December 2020 that were used for civilian power generation, 11 of which were for demonstration or prototype purposes (Appendix B, Table B-3). Of the 37 shutdown commercial reactors, 25 have utility-owned CSNF on-site; the other 12 either no longer have CSNF on-site, or, in the case of Fort St. Vrain and Three Mile Island 2 (TMI-2), have transferred all SNF to DOE ownership.

Of the 25 shutdown commercial reactors with utility-owned CSNF on-site, 4 are on sites with actively operating reactors; CSNF at the other 21 reactors (at 18 different sites) is referred to as "stranded" or "orphaned" waste because the reactor is being, or has been, decommissioned and there is no longer a licensed operating reactor on-site (Appendix B). In addition, 8 of the 94 operating reactors have announced plans for shutdown (Indian Point 3 in 2021, Byron 1 and 2 in 2021, Dresden 2 and 3 in 2021, Palisades in 2022, and Diablo Canyon 1 and 2 in 2024 and 2025); these future shutdowns would create stranded CSNF at 5 more sites (Appendix B).



Note: Figure is current as of July 2019. Since then Three Mile Island 1 (September 2019), Indian Point 2 (April 2020), and Duane Arnold (August 2020) have ceased operation.

(Source: NRC 2019a; NRC 2020a, Figure 12)

Figure 2-2. Operating Commercial Nuclear Power Plants in the US

2.1.1 SNF from the Once-Through Fuel Cycle

The stages of the once-through LWR fuel cycle, including the front end (uranium mining, milling, and enrichment, fuel fabrication, and LWR operations) and the back end (storage, transportation, and disposal), are described in Appendix A. Management of SNF from a commercial LWR across the phases of the back end of the nuclear fuel cycle includes wet (pool) storage, dry storage (including canister-based and non-canistered systems), transportation systems, and permanent disposal facilities (repositories). Current US waste management practices in these phases are described in Section 3.1 (storage), Section 3.2 (transportation), and Section 3.3 (disposal). Specific system components, which are described in more detail, and illustrated, in Appendix A, include:

- **Fuel Rod** – LWR fuel consists of enriched uranium oxide (UO_2) pellets stacked inside long zirconium alloy (e.g., Zircaloy, ZIRLO^{TM}, M5^{TM}) tubes or "cladding" to form fuel rods. Fuel rods are welded at both ends to encapsulate the fuel. A fuel rod is sometimes called a fuel pin. See Section A.2.1 for additional details.
- **Fuel Assembly** – Fuel rods are bundled together with steel hardware (e.g., spacer grids, top nozzle, bottom nozzle) into fuel assemblies. Fuel assemblies vary in physical configuration, depending on reactor type and manufacturer, but are typically configured in square arrays (generally ranging from 14x14 to 17x17 for PWR assemblies and 7x7 to 11x11 for BWR assemblies) to accommodate the fuel rods as well as a smaller number of reactivity control rods. An assembly is sometimes called a fuel bundle or fuel element. See Section A.2.2 for additional details.
- **Waste Form** – The composition and physical and chemical state of the waste. For CSNF, the waste form is primarily the spent PWR and BWR fuel (i.e., the UO_2 pellets that have oxidized and hydrated, and the cladding). For DSNF, the waste form is primarily uranium metal (N Reactor) fuel, but also includes other DOE and naval spent fuels (Sections 2.3.1 and 2.3.2). HLW is primarily borosilicate glass (DOE 2009a, Section 2.1.1.2), but also includes calcine and sodium-bearing waste (Section 2.3.3).
- **Canister** – A sealed (typically welded), unshielded cylindrical metal (stainless steel) vessel containing the waste form, and, in some cases, associated hardware. The internals of a canister include a metal basket with compartments/cells for spent fuel assemblies (or, in some cases, damaged fuel). The basket provides structural strength and neutron absorber materials. A canister is the inner container in a canister-based storage and/or transport cask system, and secures the physical location of the spent fuel assemblies within a cask (Greene et al. 2013, Introduction). See Section A.2.3 for additional details.
- **Cask (or Overpack)** – An outer metal or concrete container surrounding one or more canisters, typically with a bolted closure. The cask/overpack provides radiation shielding, structural protection and containment, and thermal management during storage, transportation, and/or disposal operations (Hardin 2017).
- **Bare Fuel Cask** – A metal container designed for individual ("bare") spent fuel assemblies (i.e., assemblies not contained in large multiple-assembly canisters) (Greene et al. 2013, Table 4). A bare fuel cask is a non-canistered storage system; some bare fuel casks can also be used for transport. See Section A.2.3 for additional details.
- **Transfer Cask** – A shielded cask used locally to transfer unshielded canister(s) within various surface facilities (e.g., from spent fuel pools to storage casks, or from storage casks to transportation casks) (Hardin 2017).

- **Storage Cask/Overpack** – The outer cask for storage of canister(s). See Section A.2.3 for additional details.
- **Transportation Cask/Overpack** – The outer cask for transportation of canister(s). See Section A.2.4 for additional details.
- **Disposal Cask/Overpack –** The outer cask for disposal of canister(s). A disposal overpack is commonly part of the design of a waste package.
- **Waste Package** – The entire package of materials to be disposed in a repository. Existing concepts typically include a single or multi-layer outer shell (i.e., the disposal overpack and shielding) and waste package internals. The waste package internals include the canister materials (e.g., CSNF, naval SNF, DSNF or DHLW canisters and internals) and the waste form. See Section A.2.5 for additional details.

A typical PWR assembly weighs approximately 700 kg (Wagner et al. 2012, Figure A-1), including, on average, 431 kg of uranium (i.e., 0.43 metric tons of uranium (MTU)) (SNL 2014b, Table A-2), with lengths most commonly ranging from about 137.1 in (3.5 m) to 165.7 in (4.2 m) and widths most commonly ranging from about 7.76 in (0.20 m) to 8.54 in (0.22 m) (Peters et al. 2020, Table A-1). The weight of the cladding and steel hardware accounts for the non-uranium mass of the assembly.

A typical BWR assembly weighs approximately 300 kg (Wagner et al. 2012, Figure A-2), including, on average, 179 kg of uranium (0.18 MTU) (SNL 2014b, Table A-2), with lengths most commonly ranging from about 134.4 in (3.4 m) to 176.2 in (4.5 m) and widths most commonly ranging from about 4.28 in (0.11 m) to 5.44 in (0.14 m) (Peters et al. 2020, Table A-2).

The amount of SNF accumulated at a reactor over its licensed life depends on factors such as how long the reactor operates each year, the duration of outages, spent fuel burnup, and operating lifetime. Average fuel burnups have increased from around 35 GWd/MTU two decades ago to over 45 GWd/MTU today (Figure A-14) and are expected to eventually exceed 60 GWd/MTU (NRC 2014a, Section 2.8 and Appendix I). High-burnup fuel, typically defined as fuel with a burnup greater than 45 GWd/MTU, is thermally hotter, more radioactive, and may need to be cooled longer, than low-burnup fuel (Section A.3.1).

The amount of spent fuel discharged from a typical LWR operating at low burnups is about 20 MTU per year (NRC 2014a, Section 2.8). After 80 years of reactor operation at low burnups, this amounts to about 1,600 MTHM of spent fuel. Under 10 CFR Part 50, a reactor could operate for 80 years if the licensee requested, and the NRC granted, two 20-year renewals of its initial 40-year operating license. For plants at which higher fuel burnups are authorized, the annual discharge of spent fuel is reduced to about 15 MTU per year (NRC 2014a, Section 2.8). Should a nuclear power plant operate for up to 80 years with high-burnup fuel, it would generate about 1,200 MTHM of spent fuel.

In wet storage (Section 3.1.1), CSNF assemblies are cooled in pools without any packaging. Dry storage is predominantly in canister-based systems, although there are a smaller number of non-canistered systems (e.g., bare fuel casks) (Section A.2). The focus of this report is on canister-based systems.

Following sufficient cooling in wet storage, CSNF assemblies are loaded into sealed canisters (or bare fuel casks), which are then placed in storage casks for dry storage (Section 3.1.2). In the US, the most common type of CSNF canister is a large-diameter, stainless-steel DPC[6], that can be used for both storage (inside a storage cask or overpack) and transportation (inside a transportation cask or overpack) of CSNF. A typical DPC has a diameter of ~2 m and a length of ~5 m and may accommodate up to 37 PWR spent fuel assemblies or 89 BWR spent fuel assemblies; a fully loaded DPC can weigh between 32 and 53 metric tons (Section A.2.3). Dry storage systems (i.e., the various canister and cask designs) currently in use are described in Section A.2.3. Transportation casks are described in Section A.2.4.

2.2 Commercial SNF and HLW

Current and projected inventory estimates are summarized for CSNF (Section 2.2.1) and CHLW (Section 2.2.2).

Inventory estimates are necessary because pool and dry storage inventories change on a regular basis and "real-time" data is not immediately available. Inventory estimates come from a number of sources. Historical SNF discharges from commercial reactors from 1968 through 2012 on an assembly basis are available from the US Energy Information Administration (EIA) on GC-859 "Nuclear Fuel Data Survey" forms collected in June 2013 (EIA 2015). The historical GC-859 data provides yearly totals for number of assemblies discharged, the initial MTU, and the average burnup and is categorized by reactor type (PWR and BWR). The GC-859 data includes CSNF stored at commercial sites but may not accurately reflect CSNF discharges that were transferred to away-from-reactor off-site facilities (EIA 2015); these transfers are most commonly now categorized as DSNF or CHLW. Subsequent projections of CSNF discharges to the end of 2019 are made using the US Commercial Spent Fuel Projection (CSFP) tool, based on operating assumptions about each reactor (Vinson 2015).

Estimates of the corresponding dry inventory (assemblies, initial MTU[7], number of casks) and pool inventory (assemblies, initial MTU[7]) are described in Peters et al. (2020, Sections 2.1 and 2.1.1). Dry storage quantities (assemblies and casks loaded) are available monthly from UxC LLC (for example, StoreFUEL 2021). Pool inventories are estimated as the remaining total inventory not in dry storage.

[6] The term "dual-purpose canister (DPC)" is used in this report to generally refer to all multi-assembly canisters; most, but not all, are licensed for storage and transportation (see Section A.2.3). Different vendors use different terminology, such as multi-purpose canister (MPC), dry shielded canister (DSC), or transportable storage canister (TSC).

[7] Peters et al. (2020) infer waste quantities from initial MTU. In this report, waste quantities are reported as MTHM.

2.2.1 Commercial SNF

The operating and shutdown commercial reactors have produced an estimated 84,446 MTHM of SNF (more than the 70,000 MTHM regulatory capacity of Yucca Mountain) through the end of 2019 (Table 2-1). These estimates include 84,272 MTHM of CSNF (83,598 MTHM at operating and shutdown commercial reactor sites and 674 MTHM that was transferred from commercial reactor sites to an off-site pool at the GE Morris facility in Illinois (Peters et al. 2020, Table 2-3)) and 174 MTHM of SNF of commercial origin that is now in the possession of DOE (TMI-2 core debris, SNF discharged from the decommissioned Fort St. Vrain gas-cooled reactor, SNF from commercial sites transferred to DOE for R&D purposes). An additional ~260 MTHM of SNF of commercial origin was transferred to West Valley for commercial reprocessing; it is classified as CHLW and is not included in these totals (Section 2.2.2).

The 174 MTHM of DSNF of commercial origin is described in Section 2.3.1 along with other DNSF (e.g., SNF inventories from non-commercial reactors such as some early power reactor demonstration program reactors).

Table 2-1. Inventory of SNF from Commercial Reactors (through December 31, 2019)

Type	Location	Dry Inventory			Pool Inventory		Total Inventory[b]	
		Assem.	MTHM	Casks[a]	Assem.	MTHM	Assem.	MTHM
CSNF	*Reactor Sites – PWR*	*60,830*	*26,169*	*2,050*	*68,550*	*28,917*	*129,380*	*55,086*
CSNF	*Reactor Sites – BWR*	*73,477*	*13,038*	*1,108*	*86,792*	*15,474*	*160,269*	*28,512*
CSNF	Reactor Sites – Total	134,307	39,207	3,158	155,342	44,391	289,649	83,598
CSNF	GE Morris				3,217[c]	674.3	3,217[c]	674
CSNF	**Total**	**134,307**	**39,207**	**3,158**	**158,559**	**45,065**	**292,866**	**84,272**
DSNF	*INL (from TMI-2)*	*177*	*81.6*	*29*			*177*	*82*
DSNF	*INL (for R&D)*		*68.4*					*68*
DSNF	*INL (from Fort St. Vrain)*		*8.9*					*9*
DSNF	INL – Total	177	158.9	29			177	159
DSNF	Fort St. Vrain	1,464	14.7				1,464	15
DSNF	**Total**	**1,641**	**173.6**	**29**			**1,641**	**174**
	TOTAL	**135,948**	**39,381**	**3,187**	**158,559**	**45,065**	**294,507**	**84,446**

[a] "Cask" refers to a dry storage system, including canister/cask systems and bare fuel casks.
[b] Totals do not include ~260 MTHM of SNF of commercial origin that was transferred to West Valley for commercial reprocessing. It is classified as CHLW.
[c] Consists of 352 PWR assemblies (132.9 MTHM) and 2,865 BWR assemblies (541.4 MTHM) (Peters et al. 2020, Table 2-3).

(Source: Peters et al. 2020, Table 2-5 and Table 2-6 and Section 2.1.2; StoreFUEL 2020, Table 16)

The estimated end-of-2019 inventory of 84,272 MTHM of CSNF includes 292,866 assemblies and consists of 55,219 MTHM (66%) in 129,732 PWR assemblies (0.43 MTHM/assembly) and 29,053 MTHM (34%) in 163,134 BWR assemblies (0.18 MTHM/assembly). Wet storage (Section 3.1.1) of CSNF is all in spent fuel pools at reactor sites (operating or shutdown), except for the small amount at the GE Morris facility. Dry storage (Section 3.1.2) of CSNF is all in dry cask storage systems (DCSSs) at operating and shutdown reactor sites.

Year-end estimated CSNF inventory for prior years is shown in Table 2-2. Since 2012, the annual increase in total estimated inventory has been quite consistent, with an average over the past six years of 2,165 MTHM/yr. Dry inventory has been steadily increasing over that time, whereas pool inventory has been slowly decreasing, driven by activities at NPPs to accelerate spent fuel pool decommissioning (e.g., the transfer of CSNF from pools to dry storage at recently shutdown reactor sites) (Peters et al. 2020, Section 2.4).

Table 2-2. Estimated Inventory of SNF from Commercial Reactors (2012 - 2019)

End of Year	Dry Inventory		Pool Inventory		Total Inventory[a]		Source
	Assem.	MTHM	Assem.	MTHM	Assem.	MTHM	
2019	134,307	39,207	158,559	45,065	292,866	84,272	Peters et al. 2020, Table 2-5
2018	124,887	36,133	160,843	46,155	285,730	82,288	Vinson and Carter 2019, Table 2-5
2017	112,156[b]	32,480[b]	165,679[b]	47,583[b]	277,835	80,063	Vinson and Metzger 2018, Table 2-5
2016	100,775[c]	29,118[c]	169,753[c]	48,729[c]	270,528	77,847	Carter et al. 2016a, Table 1-2
2015	91,785[e]	26,659	170,791	48,968	262,576	75,628	Peters et al. 2020, Figure 2-13[d]
2014	83,525[e]	24,251	171,141	49,140	254,666	73,391	Peters et al. 2020, Figure 2-13[d]
2013	74,197	22,000	175,499	49,699	249,695	71,699	Carter and Vinson 2014, Table 1-2
2012	66,190[e]	19,524[d]	173,722	49,593	239,912	69,117	Peters et al. 2020, Table 2-10

[a] Totals do not include ~260 MTHM of SNF of commercial origin that was transferred to West Valley for commercial reprocessing or ~174 MTHM of SNF of commercial origin now in possession of DOE (Table 2-1).
[b] Updated using inventory data from Peters et al. (2020, Figure 2-13) and StoreFUEL (2021). Dry inventory reported in Vinson and Metzger (2018) was current to May 2, 2017, not end of 2017.
[c] Updated using inventory data from Peters et al. (2020, Figure 2-13) and StoreFUEL (2021). Dry inventory reported in Carter et al. (2016a) was current to May 3, 2016, not end of 2016.
[d] Inventory data supporting Peters et al. (2020, Figure 2-13).
[e] From StoreFUEL (2021) as listed in Table 2-3 (CSNF only).

The commonly used term, DCSS, generally refers to a single large canister (e.g., a DPC) inside a single storage cask/overpack; it is also used to refer to a bare fuel cask in dry storage (see Section A.2 for details). The number of dry inventory “casks” reported in Table 2-1 includes all DCSSs, whether they are canister/cask systems or bare fuel casks. Table 2-3 shows the end-of-year DCSSs reported by UxC LLC going back to 2007. These quantities include CSNF and DSNF in DPC-based cask systems, CSNF in bare fuel casks, and Greater Than Class C (GTCC) waste in DPCs. The GTCC DPCs, which are not included in the totals in Table 2-1, are all at shutdown sites (StoreFUEL 2021, Table 2).

Since 2007, there has been a steady increase in the number of casks (DPCs and bare fuel casks) in dry storage. From 2016 to 2019 there were an average of about 230 casks loaded per year. The cask loading went down slightly in 2020, but the decrease was most likely due to the postponement of several loading campaigns as a result of the COVID-19 pandemic (StoreFUEL 2021).

Table 2-3. Dry Cask Storage Systems in Service (2007 – 2020)

End of Year	CSNF in DPCs[a]		CSNF in Bare Fuel Casks		DSNF in DPCs[a]		GTCC in DPCs[a]	Total		New Casks Per Year
	Assem.	Casks	Assem.	Casks	Assem.	Casks	Casks	Assem.	Casks[b]	
2020	131,776	3,102	10,822	232	1,641	29	16	144,239	3,379	176
2019	123,605	2,929	10,702	229	1,641	29	16	135,949	3,203	222
2018	114,382	2,710	10,498	226	1,641	29	16	126,521	2,981	261
2017	102,226	2,459	9,930	216	1,641	29	16	113,797	2,720	249
2016	91,285	2,217	9,490	209	1,641	29	16	102,416	2,471	194
2015	82,635	2,028	9,150	204	1,641	29	16	93,426	2,277	190
2014	74,847	1,851	8,678	195	1,641	29	12	85,166	2,087	210[c]
2013	65,361	1,645	8,406	191	1,641	29	12	75,408[c]	1,877[c]	167[c]
2012	58,364	1,490	7,826	180	1,641	29	11	67,831	1,710	168
2011								60,304	1,542	155
2010								54,046	1,387	147
2009								47,455	1,240	129
2008								41,856	1,111	152
2007								34,690	959	NA

[a] "DPC" refers all multi-assembly canisters. Most, but not all, are licensed for storage and transportation.
[b] "Cask" refers to a dry storage system, including canister/cask systems and bare fuel casks.
[c] Originally reported end-of-year data for 2013 (76,097 assemblies in 1,892 casks) was erroneous and was later corrected (StoreFUEL 2014).

(Source: StoreFUEL 2021)

The CSFP tool (Vinson 2015) was used to extend the projections of US CSNF inventory to 2080 (Peters et al. 2020, Section 2.3). The projections are based on a reference scenario for future nuclear power generation is the US. The reference scenario, "No Replacement Nuclear Power Generation", used the following assumptions[8]:

- No new reactors are constructed and operated (i.e., no replacement).
- The inventory includes the fuel discharged from the 94 operating reactors and 25 shutdown reactors at the end of 2019 (96 operating reactors at the end of 2019 with shutdowns of Indian Point 2 and Duane Arnold in 2020).
- 88 of the 94 operating reactors have one 20-year license extension and will be decommissioned after 60 years of operation (82 already have license renewals; 6 – Clinton, Perry, Comanche Peak 1 and 2, Watts Bar 1 and 2 – will receive one before the current license expires).

[8] These assumptions, which were made at the end of 2019, are consistent with the current end-of-2020 status (see Section 2.1), but do not account for the following recent developments: (1) Peach Bottom 2 and 3 received subsequent 20-year license extensions in March, 2020; these extensions would increase the projected inventory by approximately 5,460 assemblies and 970 MTU (Peters et al. 2020, Section 2.3.1); (2) recently announced early shutdowns of Byron 1 and 2 in 2021 (~20 years early) and Dresden 2 and 3 in 2021 (~10 years early) (StoreFUEL 2021, Figure 22); these shutdowns would decrease the projected inventory; and (3) Vogtle 3 and 4 do not operate. The effects of these developments on the projected inventory, which is already subject to assumptions about future reactor operations, are largely offsetting; the net effect would be minimal.

- 2 operating reactors (Turkey Point 3 and 4) have received a subsequent 20-year license extension and will operate for 80 years.
- 4 operating reactors have utility-announced early shutdown dates (Indian Point 3 in 2021, Palisades in 2022, Diablo Canyon 1 in 2024, and Diablo Canyon 2 in 2025).
- No CSNF is reprocessed.
- There are no options for permanent disposal and all CSNF remains in storage.
- Transfers from pool to dry storage occur based on overall annual estimates, declining over time as reactors shut down: a total of 3,200 MTHM/yr from 2020 to 2025; 3,000 MTHM/yr from 2021 to 2046; 1,200-2,000 MTHM/yr from 2047 to 2052; <300 MTHM/yr after that.

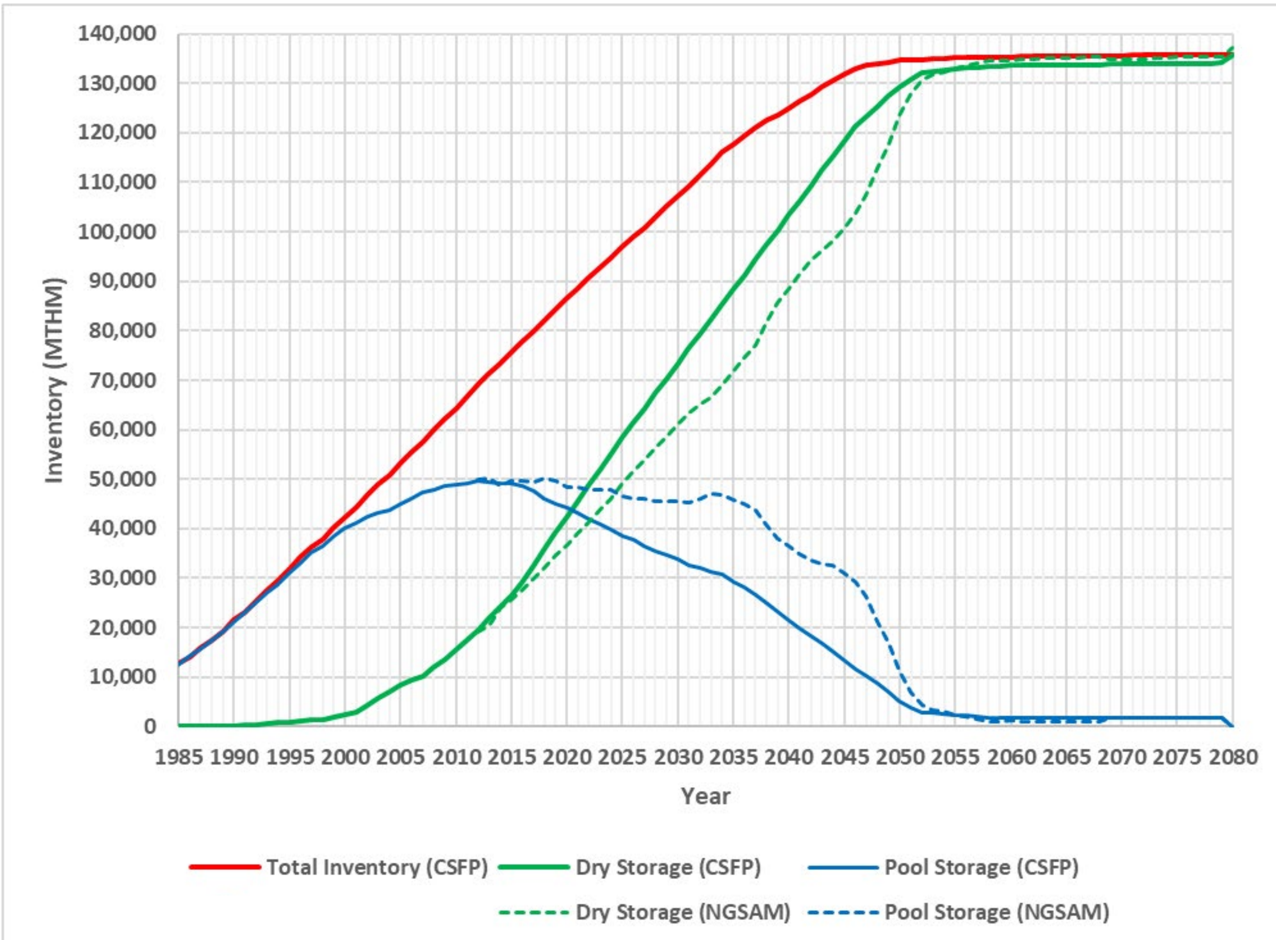
Under these assumptions, all currently operating reactors will be shut down by 2055, except for Watts Bar 2 which will be shut down in 2075 and have all transfers from pool to dry storage completed by 2080. Figure 2-3 shows the projected inventory of CSNF from 1985 through 2080 under this "no replacement" scenario. The projected inventory in 2080 totals 135,809 MTHM in 467,677 assemblies (Peters et al. 2020, Table 2-10), which includes 89,021 MTHM (66%) in 203,796 PWR assemblies and 46,789 MTHM (34%) in 263,881 BWR assemblies (adjusted[9] from Peters et al. 2020, Table 2-9). This is the projected CSNF total that is in storage at operating and shutdown reactor sites and includes the pool inventory at GE Morris – 84,272 MTHM projected at the end of 2019 plus an additional 51,537 MTHM projected to be generated from 2020 to 2075 (and all transferred to dry storage by 2080). This inventory does not include SNF generated at commercial reactors but subsequently transferred away from reactor sites for other purposes, and now classified as CHLW (see Section 2.2.2) or DSNF[9] (see Section 2.3 and Table 2-1).

In addition to the "no replacement" scenario inventory projections from the CSFP tool, Figure 2-3 also shows projections, made in 2018, from a similar no replacement scenario using the Next-Generation System Analysis Model (NGSAM) (Joseph et al. 2019; Simunich et al. 2020). NGSAM uses the projected discharges (i.e., total projected inventory) from the CSFP tool, but makes different assumptions about transfers from pool to dry storage. NGSAM also explicitly projects the numbers of specific types of canisters. The following are NGSAM assumptions that are different from the CSFP assumptions:

- Three Mile Island 1 operates until 2034, rather than shutting down in 2019.
- Turkey Point 3 and 4 have only one license extension and operate for 60 years rather than 80 years.
- Transfers from pool to dry storage are based on site-specific reactor-by-reactor operating conditions.

[9] The CSFP tool inventory projections typically include ~70 MTHM in 226 assemblies reported on GC-859 forms (or predecessor RW-859 forms) as having been transferred to DOE for R&D purposes prior to 2002 (Peters et al. 2020, Section 2.1.1 and Table D-1). Details of this DSNF inventory - predominantly from Surry 2, Ginna, Big Rock Point, and Turkey Point 3, and now stored at INL but not in NRC-licensed facilities - are provided in Carter and Vinson (2014, Tables 2-3 and 2-4). However, this is only a portion of the commercial SNF in DOE possession (see Table 2-1 and Section 2.3.1). The CSFP tool projections in this report (e.g., Table 2-2 and Figure 2-3) do not include any spent fuel now categorized as DSNF.

As shown in Figure 2-3, the NGSAM assumptions result in slower projected transfers from pool to dry storage as compared to the CSFP projections, but the overall trends are similar.



CSFP tool projections (Peters et al. 2020, Figure 2-13) assume 94 currently operating and 25 shutdown reactors (end of 2020):

- 88 operating NPPs receive one 20-year license renewal and are decommissioned after 60 years of operation
- 2 operating NPPs receive a second 20-year license renewal and are decommissioned after 80 years of operation
- 4 operating NPPs that have announced shutdown dates continue operating until those dates
- No new reactors are constructed and operated
- No CSNF is reprocessed
- There are no options for permanent disposal and all CSNF remains in storage
- Transfers from pool to dry storage are based on overall annual estimates

NGSAM projection assumptions differ from CSFP assumptions as follows:

- Three Mile Island 1 operates until 2034, rather than shutting down in 2019
- Turkey Point 3 and 4 have only one license extension and operate for 60 years rather than 80 years
- Transfers from pool to dry storage are based on site-specific reactor-by-reactor aging conditions

Figure 2-3. Projected Inventory of US Commercial Spent Nuclear Fuel in Storage

While each of the "no replacement" scenario assumptions are open to question, and details of the projection should be interpreted with caution, the overall trends shown in Figure 2-3 are important:

- Approximately 2,200 MTHM of CSNF are projected to be generated each year in the US. This rate will decrease after approximately 2035 as older reactors are shut down and decommissioned.
- Most reactor pools in the US have been filled to capacity since approximately 2012. Pool storage of newly discharged CSNF at most locations now requires transferring older and cooler fuel to dry storage.
- More CSNF will be in dry storage than in pools by about 2025.
- Pool storage capacity will decrease after approximately 2035 as older reactors are shut down and decommissioned, and by mid-century nearly all CSNF will be in dry storage.
- The total mass of CSNF generated by the existing US reactor fleet by mid-century will be greater than 130,000 MTHM, which is nearly twice the limit established by the NWPA for the proposed Yucca Mountain repository.

The assumption that none of the US inventory of CSNF will be reprocessed between now and 2080 warrants further discussion. US law and national policy do not preclude reprocessing, and it is possible that future nuclear power generation activities in the US will implement a closed fuel cycle that would change projections shown in Figure 2-3. However, no capacity for commercial reprocessing exists today in the US, and there is no plausible near-term path for constructing such capacity. The slope of the total inventory curve in Figure 2-3 is unlikely to change due to reprocessing within the next ten to twenty years. Even if reprocessing capabilities were available, there is no realistic scenario under which large amounts of existing SNF might be reprocessed, because the new discharges of 2,200 MTHM per year would be more than sufficient to provide initial feedstock for a new generation of reactors operating within a closed fuel cycle (Wagner et al. 2012). Reversing the trends shown in Figure 2-3 and decreasing the total US inventory of CSNF in storage will require permanent disposal in a geologic repository.

Peters et al. (2020) also developed three other scenarios that bound the reference scenario. Projected future inventories for those scenarios range from ~132,800 MTHM for scenario 2 (all reactors shut down at end of current license period) to ~142,300 for scenario 3 (addition of two new builds and eight additional license extensions) (Peters et al. 2020, Table 2-22).

2.2.2 Commercial HLW

There is also a small inventory of vitrified HLW from commercial reprocessing at the West Valley plant, which operated from 1966 to 1972 to recover plutonium and unused uranium. Of the 640 MTHM of SNF reprocessed at West Valley, about 260 MTHM was CSNF and about 380 MTHM was DOE N Reactor SNF (Peters et al. 2020, Section 2.2). During operations about 2,500 m^3 of liquid HLW was generated. The liquid HLW was vitrified between 1996 and 2001 producing 278 stainless-steel canisters, including 275 canisters of vitrified HLW (Peters et al. 2020, Section 2.2). The HLW from West Valley is classified as commercial HLW and not DHLW (DOE 2009a, Section 1.5.1.2.1).

2.3 DOE-Managed SNF and HLW

DOE manages a diverse inventory of SNF and HLW, primarily from defense programs (initiated by the AEC during World War II) and reprocessing operations at the Hanford Site in Washington, Savannah River Site (SRS) in South Carolina, and INL in Idaho. Smaller inventories come from a wide range of defense and R&D activities by DOE (including its predecessors AEC and ERDA), and from the Naval Nuclear Propulsion Program (NNPP). In the 1980s, reprocessing operations were ongoing at Hanford, SRS, and INL, and it was expected that DOE-managed SNF from all sources (weapons plutonium production reactors, naval propulsion reactors, and test reactors) would be reprocessed to recover usable materials, leaving HLW and transuranic (TRU) waste (waste contaminated with TRU elements and hazardous chemicals generated during the production of nuclear weapons) for long-term disposal. The cessation of most reprocessing operations by the early 1990s left DOE with a heterogeneous inventory of SNF and HLW (DOE 2014, Section 2.1).

DOE-managed HLW and SNF consists of two principal waste streams: (1) DSNF, primarily from atomic energy defense activities (weapons plutonium production reactors and naval propulsion reactors), but also including a smaller amount of SNF from DOE R&D activities and some DOE-managed SNF from commercial sources; and (2) DHLW, mostly resulting from atomic energy defense activities but also including a small amount of HLW of commercial origin (DOE 2014, Section 2.1).

Except for the relatively small volumes of SNF from naval and research reactors that will continue to operate, the inventory of DSNF and DHLW is derived from past activities and is therefore essentially known. However, much of the DSNF has not yet been packaged and much of the DHLW remains to be processed (e.g., tank wastes at Hanford and SRS and calcine at INL). As a result, estimates of the number and types of waste packages (i.e., canisters) required for disposal (and, in most cases, also for storage and transportation) continue to evolve.

Current and projected inventory estimates are summarized for DSNF (Section 2.3.1), naval SNF (Section 2.3.2), and DHLW (Section 2.3.3). The original source of most the DOE-managed inventory data is information collected in support of the Yucca Mountain Repository License Application (DOE 2008a). These projections made inventory and packaging assumptions specific to disposal in the proposed Yucca Mountain repository (e.g., waste that would have been deliverable in its originally projected 25 years of operation from 2010 through 2035 (SNL 2014a, Table 2-3)). Subsequent reports (Carter et al. 2012; SNL 2014a; SNL 2014b; DOE 2014; Carter and Vinson 2014; Carter et al. 2016a; NWTRB 2017a; Sassani et al. 2017; Vinson and Metzger 2018; Vinson and Carter 2019; Sassani et al. 2019; Peters et al. 2020) have provided updated inventory and packaging estimates based on new information and/or different repository assumptions. Specific details of canisters and waste packages for DOE-managed SNF and HLW are summarized in Section A.2.5.

2.3.1 DOE-Managed SNF

The cessation of most reprocessing operations by the early 1990s left DOE with a heterogeneous inventory of SNF, now stored at the Hanford Site, SRS, INL, and the Fort St. Vrain site in Colorado. There are several hundred distinct types of DSNF, which have been categorized into 34 DOE SNF groups based on fuel matrix, cladding, cladding condition, and enrichment (Peters et al. 2020, Section 3.1.1; SNL 2014a, Table 2-3; SNL 2014b, Table A-6).

Most of the DSNF is defense plutonium production fuel at the Hanford Site, but the inventory also includes fuel pieces and assemblies of various designs from a wide range of defense and DOE R&D activities (e.g., production, research, and other experimental reactors operated or sponsored by DOE both domestically and overseas) (Peters et al. 2020, Section 3.1). The inventory of DSNF has also been augmented by foreign research reactor fuel and SNF of commercial origin accepted by DOE under its Atomic Energy Act responsibilities (e.g., SNF from the Fort St. Vrain reactor in Colorado and the damaged core from the TMI-2 reactor) (DOE 2014, Section 2.1).

The majority of DSNF has been generated and is in storage; the current inventory (excluding naval SNF) is 2,447 MTHM (Table 2-4), based largely on data from the Spent Fuel Database maintained by the National Spent Nuclear Fuel Program (NSNFP) at INL (NSNFP, 2020)). The current DSNF inventory includes ~2,273 MTHM predominantly at Hanford, SRS, and INL (Peters et al. 2020, Section 3.1.3) and the ~174 MTHM of SNF of commercial origin that is now in the possession of DOE (TMI-2 core debris, Fort St. Vrain SNF, SNF from other commercial sites (e.g., Surry, Ginna) transferred to DOE for R&D purposes); these are listed in Table 2-1 and are primarily stored at INL. DOE continues to operate several research reactors and will be receiving SNF from universities and the DOE Foreign Research Reactor Spent Nuclear Fuel Return Program; an additional 14 MTHM is projected out to 2035 (Peters et al. 2020, Section 3.1.1).

Table 2-4. Current and Projected Inventory of DSNF

Location	Current (end of 2019) (MTHM)	Projected (2020-2035) (MTHM)	Total (MTHM)
Hanford Site	2,127		2,127
Savannah River Site	27		27
Idaho National Laboratory	273[a]		273
Fort St. Vrain	15[a]		15
Other	5		5
Research Reactors (US and Foreign)		14	14
TOTAL	**2,447**	**14**	**2,461**

[a] Includes ~174 MTHM of SNF of commercial origin that is now in the possession of DOE (~159 MTHM at INL and ~15 MTHM at Ft. St. Vrain) (see Table 2-1)

(Source: Peters et al. 2020, Section 3.1.3 and Appendix F)

Table 2-4 also provides a breakdown of the current and projected DSNF inventory by site. The storage configurations vary for each of the sites and include both dry and wet storage (Peters et al. 2020, Section 3.1.3). The largest single contributor to the inventory is Hanford N Reactor SNF (2,096 MTHM in 388 multicanister overpacks (MCOs) (SNL 2014a, Table 2-3; NWTRB 2017a; Table A1-2). Standardized canisters for DSNF have been designed, but not deployed, to accommodate the remainder of the DSNF inventory (Section A.2.5 and Table A-3). Since most of the remainder of DSNF has not yet been packaged, ultimate canister counts can only be estimated. DOE estimates a minimum of 2,500 to a maximum of 5,000 with a point estimate of 3,500 canisters (~400 MCOs and ~3,100 standard DSNF canisters) (DOE 2009a, Section 1.5.1.3).

NWTRB projects 3,332 canisters (413 MCOs and 2,919 standard DSNF canisters) (NWTRB 2017a, Table A1-3).

2.3.2 Naval SNF

The NNPP has generated SNF from operation of nuclear-powered submarines and surface ships, operation of land-based prototype reactor plants, operation of moored training ship reactor plants, early development of commercial nuclear power, and irradiation test programs (Peters et al. 2020, Section 3.2).

As summarized in Table 2-5, approximately 37 MTHM of naval SNF currently exists and is in temporary storage in 177 naval SNF canisters at the Naval Reactors Facility at INL pending shipment (Peters et al. 2020, Section 3.2.1; SNL 2014a, Section 2.1.3). The inventory of naval SNF increases as fuel is withdrawn from Navy vessels. The projected inventory of naval SNF is ~65 MTHM in ~400 naval SNF canisters in 2035 (Peters et al. 2020, Section 3.2.1); however, it will continue to increase throughout the operational lifetime of the nuclear Navy (DOE 2014, Section 2.2.1).

Table 2-5. Current and Projected Inventory of Naval SNF

Location	Current (end of 2019) (MTHM)	Projected (2020-2035) (MTHM)	Total (MTHM)
Idaho National Laboratory	37	28	65
Other	0	0	0
TOTAL	**37**	**28**	**65**

(Source: Peters et al. 2020, Section 3.2)

2.3.3 DOE-Managed HLW

DHLW derives from defense programs and associated reprocessing operations (the small amount of HLW of commercial origin at West Valley is counted as CHLW (Section 2.2.2)). Aqueous reprocessing of DSNF has occurred at the Hanford Site, INL, and SRS. Liquid HLW from aqueous reprocessing has historically been stored in underground metal storage tanks. Some stabilized DHLW has been produced from the tank wastes (vitrified HLW, also called HLW glass, at SRS and calcine at INL), but much of the DHLW remains to be processed (e.g., vitrification of tank wastes at Hanford and SRS and further treatment of calcine at INL to a waste form suitable for disposal) (Peters et al. 2020, Section 3.3). There is also a small volume of sodium-bonded SNF at INL that requires treatment.

The current (end of 2019) inventory of DHLW is summarized in Table 2-6, based on information presented in Peters et al. 2020, Section 3.3). At the Hanford Site, reprocessing of defense reactor SNF since the 1940s has generated about 220,000 m^3 of liquid HLW and 1,936 Cs and Sr capsules. At SRS, reprocessing of defense reactor SNF and nuclear targets since 1954 has produced more than 600,000 m^3 of liquid HLW. Through evaporation and vitrification, SRS has reduced this inventory to the current level of about 136,000 m^3 of liquid HLW. SRS began vitrifying liquid HLW in 1996 and through June 1, 2017 has produced 4,162 HLW canisters. At INL, reprocessed

SNF from naval propulsion reactors, test reactors, and research reactors has generated approximately 30,000 m^3 of liquid HLW. Between 1960 and 1997, all of the liquid HLW at INL was converted into about 4,400 m^3 of calcine (a granular solid). In addition, a demonstration of electro-chemical processing to treat 60 MTHM of sodium-bonded SNF is being pursued at INL.

Table 2-6. Current Inventory of DHLW

Location	Liquid Tank Waste (m^3)	Solid Waste
Hanford Site	220,000	1,936 capsules [a]
Savannah River Site	133,000	4,210 canisters [b]
Idaho National Laboratory (calcine)		4,400 m^3
Idaho National Laboratory (sodium-bonded SNF)		60 MTHM

[a] Hanford Cs and Sr capsules (diameter < 0.09 m and length ~0.5 m)
[b] SRS HLW canisters (see Table A-3 for dimensions)

(Source: Peters et al. 2020, Section 3.3.1)

The projected inventory of DHLW is summarized in Table 2-7. The projected inventory primarily involves processing of existing waste to waste forms more suitable for disposal (e.g., vitrification of tank wastes), rather than the production of additional waste. The projected inventories are reported in terms of canisters.

Table 2-7. Projected Inventory of DHLW

Location	Vitrified HLW Canisters	HIP Calcine Canisters	Electro-Chem Sodium Canisters	Total
Hanford Site (tank waste)	7,800 [a]			7,800 [b]
Savannah River Site (tank waste)	8,121 [c]			8,121
Idaho National Laboratory		3,700 [d]	102 [d]	3,802
TOTAL	**15,921**	**3,700**	**102**	**19,723**

[a] Hanford HLW canisters (see Table A-3 for dimensions)
[b] The Hanford total may or may not include the Cs and Sr capsules (Table 2-6). The Hanford tank waste will also produce 8,400 TRU waste drums to be disposed at WIPP.
[c] SRS HLW canisters (see Table A-3 for dimensions). Total includes the current inventory (Table 2-6).
[d] INL HLW canisters (see Table A-3 for dimensions).

(Source: Peters et al. 2020, Section 3.3.2 and Table 3-7)

At the Hanford Site, the Hanford Tank Waste Treatment and Immobilization Plant (WTP), or Vitrification Plant, is currently under construction to convert the HLW at the Hanford Site to an estimated 7,800 borosilicate glass HLW canisters and 8,400 TRU waste drums (to be disposed at the Waste Isolation Pilot Plant (WIPP)) based on a reference baseline scenario assumption (Peters et al. 2020, Section 3.3.2). Other scenarios are also being assessed.

At SRS, H Canyon is currently the only operating reprocessing facility in the US. It is estimated that an additional 12,000 m^3 of liquid HLW may be generated with continued canyon operations. The total SRS inventory, based on those HLW canisters in the current inventory and those projected from future operations, is estimated to be 8,121 vitrified HLW canisters (Peters et al. 2020, Section 3.3.2).

At INL, DOE is currently planning to treat the calcine using hot isostatic pressing (HIP) to produce an estimated 3,700 HIP canisters (Peters et al. 2020, Section 3.3.2). In addition, the electrochemical processing to treat sodium-bonded SNF is projected to ultimately produce 102 HLW canisters (Peters et al. 2020, Table 3-7).

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3. CURRENT PRACTICES FOR MANAGEMENT OF COMMERCIAL SNF IN THE US

As noted in Section 1.2, the US does not have an operating disposal facility for SNF and HLW, and CSNF continues to accumulate in at-reactor pools and dry storage facilities. Furthermore, the federal government (DOE) did not meet its obligations under the Standard Contracts to begin accepting CSNF by 1998, and is now liable for damages to some utilities to cover the costs of on-site, at-reactor storage (DOE 2013). The current practices for storage (Section 3.1), transportation (Section 3.2), and disposal (Section 3.3) resulting from these conditions, are summarized in the following subsections. Ongoing R&D supporting these activities is summarized in Appendix C. Excerpts from relevant regulations are provided in Appendix D.

Despite the lack of integration between storage, transportation, and disposal, CSNF is, and can continue to be, stored and transported safely in accordance with NRC regulations (NRC 2014a; NRC 2014b). Rather, the lack of integration leads to delays and/or added costs.

Due to the larger volume of CSNF as compared to DOE-managed waste, the remainder of this report focuses on CSNF management. However, special considerations for DSNF and DHLW, as well as for spent fuel from alternative/advanced fuel cycles[10] and ATF[10], are noted where they may be important.

3.1 Storage of SNF

As noted in Section 1.1, it was originally expected that spent fuel, upon being removed from the reactor core, would be held in pools at nuclear power plants until it had cooled sufficiently and then transported for reprocessing or disposal. However, by the late 1970s when reprocessing ceased to be a national policy objective, utilities began reconfiguring reactor pools and storage practices to accommodate substantially more spent fuel assemblies. As pools approached the revised capacity limits in the early 1990s, and as it became clear that the proposed Yucca Mountain repository would not open in 1998, utilities moved forward with implementing on-site dry storage systems. This resulted in on-site storage of much larger quantities of CSNF for much longer periods of time than policy-makers envisioned or utility companies planned for when most of the current fleet of reactors were built (BRC 2012, Chapter 5.1).

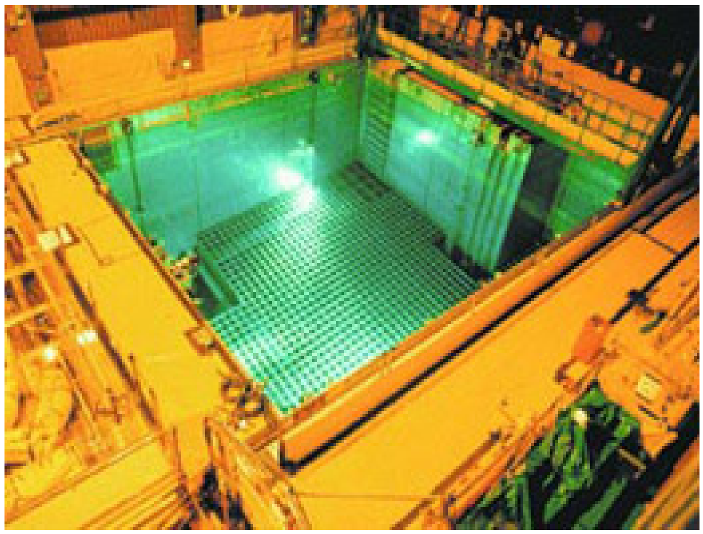
The current options for storage, shorter-term wet (pool) storage at the reactor (Section 3.1.1) and longer-term dry (cask) storage at the reactor (Section 3.1.2), are described below. Section 3.1.3 summarizes the current practices for storage of stranded CSNF at shutdown reactors. Options for away-from-reactor storage (i.e., consolidated interim storage) are discussed in Section 3.1.4. Ongoing R&D supporting storage system performance is summarized in Appendix C.1.

3.1.1 Wet Storage

Nuclear fuel typically remains in a commercial power reactor for about four to six years, after which it can no longer efficiently produce energy (BRC 2012, Chapter 3.1). Spent fuel removed

[10] There is a wide range of fuel types existing within the DOE complex; where spent fuel from alternative/advanced fuel cycles, including ATF, is substantially different from the existing spent fuel inventory, further consideration and characterization will be needed (Sassani et al. 2019, Section 2.2).

from a reactor is thermally hot and emits a great deal of radiation. To keep the fuel cool and to protect workers from radiation, newly discharged spent fuel assemblies are transferred to wet storage (a deep, water-filled pool) where they are placed in metal racks (Figure 3-1). Typically, spent fuel pools are at least 40 ft (12 m) deep, allowing the spent fuel to be covered by at least 20 ft (6 m) of water, which provides adequate shielding from the radiation for anyone near the pool NRC 2014a, Section 2.1.2.1). The assemblies are moved into the spent fuel pools from the reactor along the bottom of water canals, so that the spent fuel is always shielded to protect workers (NRC 2020b). As described further in Section 3.1.2, pools are also used for the transfer of spent fuel assemblies into canisters used for dry storage.



(Source: NRC 2020b)

Figure 3-1. Spent Nuclear Fuel in Wet Storage at a Reactor Site

Most wet storage currently resides at operating reactors and, thus, is licensed by NRC under 10 CFR Part 50 as part of the reactor license. An initial reactor license lasts up to 40 years, with renewal in increments of up to 20 years (see Appendix B).

Spent fuel resides in the pool for at least 3 years, depending on the design and license of the DCSS or transportation cask, but typically for at least 10 years (Hanson and Alsaed 2019, Section 2.1.1). The higher-burnup CSNF currently being discharged requires a few more years of cooling (NRC 2014a, Appendix I). Spent fuel at many US reactor sites has been in pool storage for several decades (BRC 2012, Chapter 3.1).

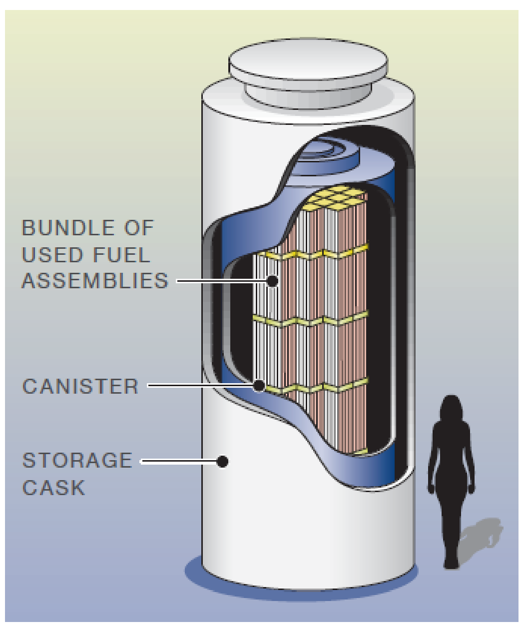
The general strategy for a commercial reactor is to use the entire wet storage capacity, minus the space required if it was necessary to completely empty the reactor core (Rechard et al. 2015, Section 2.1.2). Once the wet storage is full, CSNF must be moved from the pool into dry storage to accommodate newly discharged CSNF from the reactor.

Pools for early reactors were designed to hold slightly more than one full core load (~100 MTU) (NRC 2014a, Section 2.1.2.1; Rechard et al. 2015, Section 2.1.1). However, as CSNF continued to accumulate in pools and dry storage, NRC approved packing the CSNF assemblies more densely in the pools to increase the wet storage capacity, provided measures were included to prevent criticality. As a result, a typical spent fuel pool can hold anywhere from about four to ten reactor core loads, or about 400 to 1200 MTHM (StoreFUEL 2021, Table 13). There are also some multi-unit reactor sites with shared pools. These configurations mean that the spent fuel pool at a typical reactor reaches capacity after about 35 years of low-burnup operation (about 46 years for high-burnup operation) and spent fuel must be removed from the pool to ensure full core offload capability (NRC 2014a, Section 2.2.1).

At the end of 2019 there was an estimated 44,391 MTHM (155,342 assemblies) of CSNF in wet pool storage at operating and shutdown reactor sites (Table 2-1). This is down from a pool inventory of 50,195 MTHM at the end of 2016 (Table 2-2). The CSNF pool inventory has been slowly decreasing for the last several years and projected to continue to decrease in the future (see Figure 2-3), as reactors shut down and CSNF continues to be transferred to dry storage.

3.1.2 Dry Storage

Dry storage is considered to be the most economical and operationally efficient option available today for extended periods of storage (i.e., multiple decades up to 100 years or possibly more) (BRC 2012, Chapter 5.1). Multiple dry storage system designs are in use in the US today; these are summarized in Section A.2.3. The most common DCSSs are canister-based systems, where spent fuel assemblies are sealed in a large stainless-steel multi-assembly inner canister (e.g., a DPC), which is then placed in a concrete or steel storage cask (Figure 3-2). There are also a smaller number of non-canistered systems (e.g., bare fuel casks), where spent fuel assemblies are loaded directly into a metal cask.



(Source: BRC 2012, Figure 4)

Figure 3-2. Spent Nuclear Fuel in Dry Storage

Following sufficient cooling of spent fuel assemblies in wet storage, the assemblies are loaded into canisters (e.g., DPCs) in the pools. The canisters are then removed from the pool and dewatered, dried, sealed with welded lids, and backfilled with helium gas (Section A.2.3). Shielded transfer casks are used to move the canisters to storage pads, where they are placed in concrete (or concrete and steel) casks for shielding and physical protection during storage.

DCSSs may store canisters vertically or horizontally. Vertical DCSSs can be constructed above ground on concrete pads or below grade; horizontal DCSSs are designed for canisters to be emplaced in modular concrete storage units or "vaults" (Figure 3-3). Additional details about canisters and vertical and horizontal DCSSs are provided in Section A.2.3.

(a) Vertical Above Ground
(Maine Yankee, Wiscasset, ME)

(b) Vertical Below Ground
(Humboldt Bay, Eureka, CA)

(c) Horizontal
(Rancho Seco, Herald, CA)

(Source: Rechard et al. 2015, Figure 3)

Figure 3-3. Types of Dry Cask Storage Systems

During storage, an important consideration is to maintain the fuel cladding temperature below 400°C (NRC 2010, Sections 4.4.2 and 8.4.17) to minimize the potential for degradation of the cladding on the fuel rods. Cooling during the dry cask storage period is provided by the natural circulation of air through the annular space between the cask and the canister. The reliance on natural (passive) cooling makes DCSSs less vulnerable to system failures.

Individual DPC specifications vary, but the most commonly used ones range from 4.6 m to 5.0 m in length, about 1.7 m to 1.9 m in diameter, and have a loaded weight of about 32.1 metric tons to 52.8 metric tons (Greene et al. 2013, Table 2). Design capacities of DPCs have increased over time, and currently the largest ones hold 37 PWR assemblies or 89 BWR assemblies.

The most commonly used storage casks (exclusive of the horizontal cask systems) range from about 5.7 m to 5.9 m in length (height), about 3.4 m to 3.5 m in diameter. They weigh from about 99.3 metric tons to 122.5 metric tons empty and 140.8 metric tons to 163.3 metric tons loaded (Greene et al. 2013).

At the end of 2019 there was an estimated 39,207 MTHM (134,307 assemblies) of CSNF in 3,158 DCSSs at operating and shutdown reactor sites (Table 2-1). The dry inventory is projected to increase in the future (see Figure 2-3), as reactors shut down and CSNF continues to be transferred from wet to dry storage. The 3,158 DCSSs include DPCs and bare fuel casks with CSNF. There are also 29 casks with DSNF and 16 casks with GTCC waste, for an overall total of 3,203 DCSSs (Table 2-3).

As noted previously, the term DPC is used in this report to generally refer to all multi-assembly canisters. Although some of the older-design canisters are NRC-certified for storage only, ~85% of the canisters in use today are "true" DPCs that are NRC-certified for both storage (inside a storage cask) and transportation (inside a transportation cask or overpack) of CSNF.

NRC general requirements for dry storage of CSNF are in 10 CFR Part 72, applicable to the licensing of an Independent Spent Fuel Storage Installation (ISFSI)[11]. The NRC authorizes construction and operation of ISFSIs by general and site-specific licenses. As summarized in Appendix B and NRC (2014a, Sections 2.1.2, 2.1.3 and Appendix G):

- a general license (described in 10 CFR Part 72, Subpart K) can be used by a nuclear power plant licensee to store CSNF in NRC-approved casks (i.e., with a certificate of compliance) at a site licensed to operate a reactor under 10 CFR Part 50, and
- a site-specific license (described in 10 CFR Part 72, Subparts A through I) can be used at or away from a reactor site based on an application that contains technical requirements and operating conditions for the ISFSI.

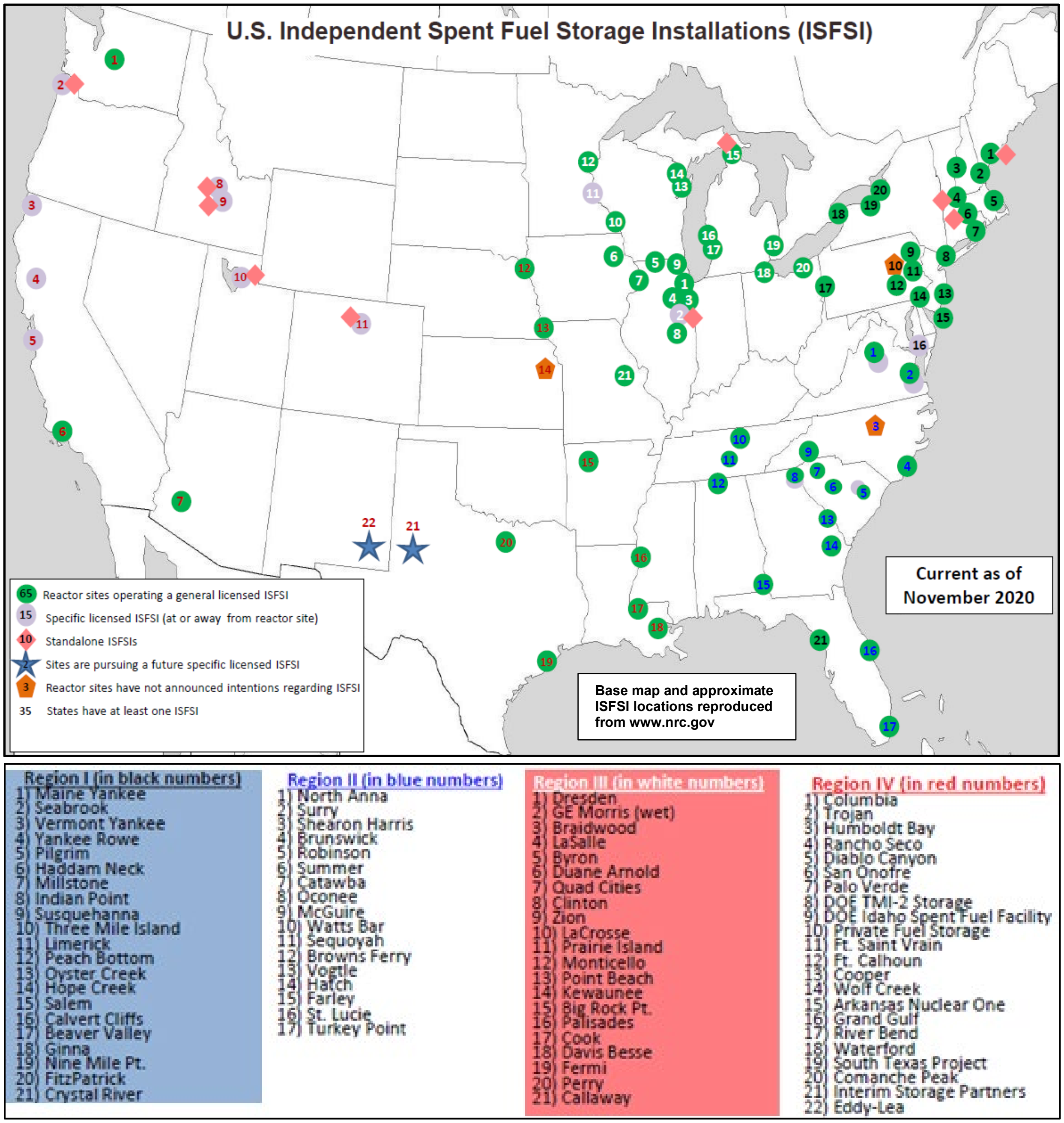
An ISFSI may be licensed for up to 40 years with options to renew in up to 40-yr increments (10 CFR Part 72.42 and 72.240).

As of November 2020, there were 81 ISFSI licenses (66 general licenses and 15 site-specific licenses) at 77 sites in 35 states[12]. 62 sites have general licenses, 11 sites have site-specific licenses, and 4 sites have both a general and site-specific license (Figure 3-4). ISFSIs are most commonly co-located with reactors. Of the 66 general licenses, 51 are at operating reactor sites and 15 are at shutdown reactor sites with stranded waste (Appendix B). Of the 15 site-specific licenses, 7 are at operating reactor sites (4 of these sites also have a general license), 3 are at shutdown sites with stranded waste, and 5 are at away-from-reactor locations.

At the 56 sites with operating reactors, 54 have ISFSIs and 2 (Shearon Harris and Wolf Creek) have not yet made a decision on licensed storage (Appendix B). Each of the 18 shutdown sites with stranded waste also has an ISFSI (Section 3.1.3 and Appendix B). The remaining 5 sites are at away-from-reactor locations (GE Morris, Fort St. Vrain, 2 sites at INL, and Private Fuel Storage (PFS)). The two away-from-reactor licenses at INL[13] are the DOE-TMI-2 ISFSI, which contains core debris from the damaged TMI-2 reactor, and the DOE Idaho Spent Fuel Facility, which has not opened. PFS is a private consortium that was interested in operating an MRS facility in Utah. As described further in Section 3.1.4, PFS received a site-specific license, but it did not receive

[11] 10 CFR Part 72 is also applicable to licensing for an MRS facility, but all storage licenses to date are for ISFSIs.
[12] Eight states (Colorado, Idaho, Iowa, Maine, Massachusetts, Oregon, Utah, and Vermont) have ISFSIs but no operating reactors; Kansas has an operating reactor (Wolf Creek), but no ISFSI (currently only at-reactor pool storage).
[13] DOE stores SNF at three locations on the INL site: the Idaho Nuclear Technology and Engineering Center (INTEC), the Naval Reactors Facility, and the Materials and Fuels Complex. INTEC includes the DOE-TMI-2 ISFSI as well as some other storage facilities that are not NRC licensed (NWTRB 2017b).

the necessary land use approvals from the federal government and never moved past the design phase.



Note: Three Mile Island was issued a general license in August 2020 (NRC-2020-0174 Docket No. 72-77) that is not reflected on this Figure. As a result, there are 66 sites operating a general licensed ISFSI and 2 sites that have not announced intentions.

(Source: NRC 2020c)

Figure 3-4. Independent Spent Fuel Storage Installations (ISFSIs) as of November 2020

The NRC has also received recent applications from two other private entities, one in west Texas and one in southeastern New Mexico, for away-from-reactor consolidated interim storage. These facilities, described in more detail in Section 3.1.4, would be site-specific ISFSIs under 10 CFR Part 72.

3.1.2.1 Future Projection of Canisters in Dry Storage

Between 1986 and 1998, CSNF was most commonly loaded into dry storage canisters holding 24 PWR assemblies; no BWR fuel had yet been transferred to dry storage. In the early 2000s, canisters holding 32 PWR assemblies and 68 BWR assemblies were introduced. By the end of 2012 there were 1,670 canisters of CNSF in dry storage (Table 2-3); comprised of 1,130 canisters of PWR assemblies with an average of ~27 assemblies/canister and 11.6 MTHM/canister and 540 canisters of BWR assemblies with an average of ~65 assemblies/canister and 11.5 MTHM/canister (StoreFUEL 2013)[14].

In 2013 and 2014, canisters holding 37 PWR assemblies and 89 BWR assemblies were introduced. By the end of 2019, there were an estimated 39,207 MTHM of CSNF in dry storage at operating and shutdown reactor sites in 3,158 DCSSs; the totals correspond to 2,050 canisters of PWR assemblies with an average of ~30 assemblies/canister and 12.8 MTHM/canister and 1,108 canisters of BWR assemblies with an average of ~66 assemblies/canister and 11.8 MTHM/canister (Table 2-1). From 2018 to 2019 the PWR canisters loaded averaged 33.5 assemblies/canister and 14.77 MTHM/canister and the BWR canisters loaded averaged 65.1 assemblies/canister and 11.49 MTHM/canister; the overall average for all canisters loaded in 2019 was 42.5 assemblies/canister and 13.84 MTHM/canister (based on Peters et al. 2020, Tables 2-5 and 2-6 and Vinson and Carter 2019, Tables 2-5 and 2-6).

As noted in Section 2.2.1 and Figure 2-3, the mass of CSNF in dry storage will steadily increase under the "no replacement" scenario to an estimated 135,809 MTHM in 2080, when all CSNF from the final shutdown reactor has been transferred to storage. The projected inventory in 2080 includes 89,021 MTHM in 203,796 PWR assemblies and 46,789 MTHM in 263,881 BWR assemblies (Section 2.2.1). The estimated dry storage inventory to be loaded from 2020 to 2080 is 96,602 MTHM, consisting of 62,851 MTHM in 142,966 PWR assemblies and 33,751 MTHM in 190,404 BWR assemblies.

Estimates of the number of canisters in dry storage associated with these inventory projections require assumptions to be made about the size and loading of the canisters (assemblies and/or MTHM).

Using a simple assumption that the future loading of 96,602 MTHM will all be at the 2019 overall annual rate of 13.84 MTHM/canister results in an estimate of 6,979 additional canisters loaded from 2020 to 2080, for a total of 10,137 canisters.

Using an assumption that BWR assemblies and PWR assemblies will be loaded at annual rates similar to 2019 (66 assemblies/BWR and 34 assemblies/PWR) results in an estimate of 7,090 additional canisters (2,885 BWR and 4,205 PWR) loaded from 2020 to 2080, for a total of 10,248

[14] StoreFUEL (2013) provides information about canisters and assemblies in dry storage. MTHM are extrapolated back from Carter and Vinson (2014, Appendix B).

canisters (3,993 BWR and 6,255 PWR). Assuming future loadings that approach 68 or 89 assemblies/BWR and 37 assemblies/PWR could reduce the estimate to as few as 9,500 total canisters.

3.1.3 Stranded CSNF Storage

As described in Section 2.1 and Appendix B, there are currently (end of 2020) 21 shutdown reactors at 18 sites with stranded CSNF. In addition, 8 of the 94 operating commercial reactors have announced plans for shutdown; these future shutdowns would create stranded CSNF at 5 more sites.

Stranded CSNF at older "legacy" shutdown sites (all shut down by 1997 and have not had an operating reactor on-site for at least 20 years) includes 10 shutdown reactors at 9 sites (Appendix B). At the end of 2019, the ISFSIs at these 9 sites had a projected total CSNF inventory of 2,815 MTHM, all in dry storage in a total of 248 casks (Peters et al. 2020, Table 2-5).

Stranded CSNF at the more recently (after 2010) shutdown sites includes 11 shutdown reactors at 9 sites (Appendix B). At the end of 2019, the ISFSIs at 8 of these sites (Duane Arnold is not included in these totals as it did not close until 2020) had a projected total inventory of 6,202 MTHM in pool and dry storage (Peters et al. 2020, Table 2-5). This CSNF is in the process of being transferred to dry storage; 298 casks are currently loaded, and it is projected that 480 casks will eventually be required for all of the CSNF (Peters et al. 2020, Table 2-12).

In addition to the 18 commercial sites there is also stranded waste at the ISFSIs at GE Morris, Fort St. Vrain, and INL.

All of the sites with stranded waste are monitored and well-guarded and hence are not thought to present immediate safety or security concerns; nonetheless, the continued presence of CSNF at shutdown reactor sites is problematic and costly (BRC 2012, Chapter 5.2.1). For example, it prevents the shutdown sites from being reclaimed for economically productive or otherwise desirable uses that would benefit the surrounding communities and most of the shutdown sites no longer have the capability to remove CSNF from storage canisters (e.g., spent fuel pools or other fuel handling facilities no longer exist) for inspection if long-term degradation problems emerge that might affect the ability to transport the canisters (BRC 2012, Chapter 5.2.1).

3.1.4 Consolidated Interim Storage

A consolidated interim storage facility (CISF) could accommodate spent fuel from both operating and decommissioned sites. It would likely be dry storage (although wet storage is not precluded) at an off-site location, not at a reactor site. Under the current practices for the management of the back end of the nuclear fuel cycle in the US, a CISF could (BRC 2012, Section 5.2):

- allow for the removal of stranded waste from shutdown reactor sites,
- enable the federal government to begin meeting waste acceptance obligations under the terms of the Standard Contracts, and,
- simplify repository operations by providing thermal management (e.g., buffer storage, aging, blending hot and cool waste) and/or repackaging.

The NWPA, as amended, anticipated the possibility of long-term away-from-reactor storage under the provisions for MRS (see Section 1.1). However, as noted in Section 1.1, these provisions restrict the federal government from constructing MRS (which would be limited to 10,000 MTHM while the repository is being constructed and subsequently to 15,000 MTHM once the repository begins operations) until a license for construction of a repository has been issued. The purpose of these provisions is to ensure that an interim storage facility does not become a de facto disposal site in the absence of a repository, or delay progress towards a repository decision. This legislative restriction would need to be removed to allow the construction of a federal CISF independent of the progress on a repository site (MIT 2011, Chapter 4).

In 1986, DOE submitted a proposal to locate MRS at Clinch River in Oak Ridge, Tennessee. DOE also proposed that an MRS facility include more than just storage; a central feature was fuel rod consolidation to reduce the number of waste packages required for disposal (Rechard et al. 2015, Section 3.2.2). However, Congress nullified the DOE proposal for siting an MRS facility at Clinch River in the NWPAA (1987, Section 142(a)). The NWPAA instead established an MRS Commission to prepare a report on the need for a federal MRS facility as part of the US national nuclear waste management system (NWPAA 1987, Section 143) and established the Office of the Nuclear Waste Negotiator (NWPAA, Section 402) to attempt to find a state or tribe willing to host a repository or MRS facility (although, with the identification of Yucca Mountain as the repository site, the focus was on siting MRS).

The MRS Commission report, published in 1989 (Radin et al., 1989), concluded that a MRS facility, as described in the NWPAA, was not justified at that time. However, the MRS Commission stated in the report that its recommendation could change depending on several factors, one of which was if the opening of a geologic repository was delayed considerably beyond its initial scheduled date of operation. The Secretary of Energy chose to continue to explore development of federal MRS through the Office of the Nuclear Waste Negotiator because of potential benefits to the overall waste management system (DOE 1991). Three communities and 20 tribes showed interest (Jenkins-Smith et al. 2013, Appendix A), but, in 1995, Congress closed the Office of the Nuclear Waste Negotiator, due in part to friction between tribes exploring interest and states wishing to deny access (Rechard et al. 2015, Section 3.2.3).

After the Nuclear Waste Negotiator Office was closed, one of the tribes that had been interested, the Skull Valley Band of the Goshute Indian Tribe in Utah, signed an agreement with a consortium of 11 utilities, called PFS, to lease and operate a dry cask storage site for 25 years on the reservation, with an option to renew for another 25 years (Rechard et al. 2015, Section 3.3). PFS submitted a license application to NRC to build a facility to store 40,000 MTHM in 1997. Because PFS was private, it was not subject to the NWPA limitations on the development of federal CISFs; the utilities would maintain title and responsibility for the CSNF.

In 2006, NRC authorized PFS to store CSNF for 20 years with an option to renew for another 20 years as a site-specific ISFSI under 10 CFR Part 72 (one of 15 shown in Figure 3-4). Construction of PFS was blocked through strong opposition by the State of Utah and its Congressional delegation, accompanied by adverse rulings by the US Department of Interior. PFS asked NRC to terminate the license in December 2012, after spending ~$70 million (Rechard et al. 2015, Section

3.3). Despite the PFS facility not being constructed, the issuance of the NRC license supports the assumption that an away-from-reactor storage facility can be licensed.

More recently, Interim Storage Partners (ISP) and Holtec International Texas have separately explored the development of private CISFs and have each submitted an application for a site specific ISFSI license. ISP, a joint venture between Waste Control Specialists (WCS) and Orano, has proposed a CISF for up to 40,000 MTU in Andrews County, Texas. Holtec International, supported by the Eddy-Lea Energy Alliance (ELEA) has proposed a CISF for up to 8,680 MTU in Lea County, New Mexico (NRC 2020d). Both applications are currently under review by the NRC.

In summary, federal CISF sites remain to be licensed and their viability is uncertain without modifications to the portions of the NWPA that link federal interim storage to repository licensing and operation. Private efforts for developing regional interim storage facilities also remain to be licensed. They have historically been stymied by national and state political opposition (MIT 2011, Chapter 4).

Another option for consolidated interim storage is at a repository site. While the provisions of MRS in the NWPA preclude development of a federal CISF in any state where a repository site is located, the repository design may include de facto interim storage. For example, the Yucca Mountain Repository License Application (DOE 2009a) included (1) surface aging and buffer storage for blending (2) and subsurface ventilation and aging during the period of operations and regulatory retrievability (see Section 3.3.2). While the License Application did not formally include MRS (in accordance with NWPA) or an ISFSI (in accordance with 10 CFR Part 72), these design features accomplished some of the desired functions of a CISF.

3.2 Transportation of SNF

The safe transport of SNF and HLW is an important aspect of the nuclear waste management system. In the US, spent fuel has typically been transported via truck or rail; other nations also use ships for spent fuel transport (BRC 2012, Chapter 3.1). Since the early 1960’s, transportation has occurred on an intermittent basis (e.g., return of foreign SNF to the US, shipment of experimental SNF for testing, CSNF transfers between reactors owned by the same company prior to 2002, and shipment of naval SNF for storage); more than 3,000 shipments of SNF have been made in the US for a total of ~1.7 million miles (Rechard et al. 2015, Section 2.2; DOE 2009b, Section I.B).

Transportation of CSNF requires removal of the canister from the storage cask using a transfer cask and emplacement in a reusable shielded transportation cask. Transportation casks provide structural protection and containment, radiation shielding, and thermal management. Currently available transportation cask systems for DPCs are summarized in Section A.2.4. Certified transportation casks exist for about 85% of the current CSNF inventory in dry storage (Section A.2.4). However, the existence of storage-only canisters (typically older) complicates the task of eventually transporting the spent fuel in them to a CISF site or to a disposal facility.

Due to the large diameter and weight, transportation is most likely to be by rail. A typical rail cask has a cask diameter of 2.4 m, an overall diameter including impact limiters of 3.4 m, and an overall length including impact limiters of 7.6 m. The largest rail casks can accommodate canisters with of up to 37 PWR assemblies or 89 BWR assemblies. With loaded DPCs (the heaviest of which may weigh more than 50 metric tons), the most commonly used rail cask systems may weigh from about 113.4 metric tons to 158.8 metric tons (Greene et al. 2013, Table 1). Truck casks, which might be needed at older shutdown sites that lack rail access, can only accommodate smaller canisters (up to 4 PWR assemblies or 9 BWR assemblies) and have a loaded weight of about 23 metric tons (Greene et al. 2013, Table 1).

NRC general requirements for transportation of SNF are in 10 CFR Part 71. Transportation packages (i.e., the outer cask and the internal canister and other materials) must, under normal conditions of transport (NCT) and hypothetical accident conditions (HAC), have "no loss or dispersal of radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging" and "no accessible surface of a package would have a temperature exceeding 50°C" (10 CFR Part 71.43).

NRC currently certifies transportation cask designs based primarily on numerical modeling of HAC. The numerical modeling has been validated with small-scale testing (Rechard et al. 2015, Section 2.2.2). As specified in the Standard Contract, the utility is responsible for preparing CSNF for shipment in transportation casks provided by DOE (10 CFR Part 961.11, Article IV).

In 2014, NRC performed a risk assessment for SNF transportation that examined doses and risks from routine transportation and from two accident scenarios, (1) cask is not damaged or affected and (2) cask is affected (NRC 2014b). The NRC findings, quoted verbatim below, included (NRC 2014b, Public Summary):

- The collective doses from routine transportation are very small. These doses are about four to five orders of magnitude less than collective background radiation doses.
- Radioactive material would not be released in an accident if the fuel is contained in an inner welded canister inside the cask.
- Only rail casks without inner welded canisters would release radioactive material and only then in exceptionally severe accidents.
- The certification process not only assures that casks will survive the HAC, but that they also survive 99.9999 percent of more severe accidents. Therefore, if there were an accident during a spent fuel shipment, there is less than a one-in-a-billion chance that the accident would result in a release of radioactive material.
- If there was a release of radioactive material in a spent fuel shipment accident, the dose to the maximally exposed individual would be less than 2 Sv (200 rem) and would not cause an acute fatality.
- The collective dose risks for the two types of extra-regulatory accidents (accidents involving a release of radioactive material and loss of lead shielding accidents) are negligible compared to the risk from a no-release, no-loss of shielding accident.

The results of the NRC risk assessment affirm that radiological impacts from SNF transportation conducted in compliance with NRC regulations are low, and that regulations for transportation of radioactive material are adequate to protect the public against unreasonable risk (NRC 2014b, Section 6.4).

Overall, the set of standards and regulations that currently exists to govern the transport of SNF, HLW, and other nuclear materials has functioned well and the safety record for past shipments of these types of materials has been excellent (BRC 2012, Chapter 9). However, the transportation of the projected ~136,000 MTHM of CSNF would require a larger-scale shipping campaign and infrastructure spread over several decades. Successfully developing and operating such a transportation system requires advance planning to address various technical and programmatic challenges, including (1) the design, fabrication, testing and licensing needed equipment (e.g., transportation casks and rail rolling stock, including the capacity for large DPCs), and (2) the early implementation and testing of institutional arrangements involving state, tribal and local officials (BRC 2012, Chapter 9; Appendix D).

This task is further complicated by the uncertainties involving future CSNF management practices and locations for a repository and potentially a CISF. For example, when shipping CSNF from an existing ISFSI to a CISF for further storage, the fuel would be first be subject to storage requirements at the ISFSI under 10 CFR Part 72, then to transportation requirements under 10 CFR Part 71, and finally to 10 CFR Part 72 requirements again for storage at the CISF. There are several regulatory, technical, and logistical challenges associated with this so-called “storage to transportation to storage” or “72-71-72” sequencing, such as: how is the canister, fuel, and cladding integrity confirmed prior to shipment; and how is the integrity again confirmed prior to acceptance. Transportation system logistics are discussed further in Section 3.2.3 and Appendix E. Additional considerations for integration of transportation with storage and disposal are discussed in Section 4.3.

3.2.1 Transportation Cask Testing

NRC initiated a Package Performance Study in 2001 to examine the response of full-scale rail casks in extreme transportation accidents (Sprung et al. 2002; NRC 2003). The objectives of the Package Performance Study were to demonstrate the safety of transportation casks, with public participation, and to provide high-fidelity data for validating modeling (Rechard et al. 2015, Section 2.2.2). However, plans to fund the Package Performance Study ceased in 2009 because of budget constraints (the estimated cost of the study was approximately \$15 million) and uncertainties about the Yucca Mountain project (BRC 2012, Chapter 9.2).

Between June and October 2017, a series of rail cask tests were conducted by Equipos Nucleares Sociedad Anónima (ENSA) of Spain, collaborators from the Republic of Korea, and DOE (Kalinina et al. 2018). The ENSA/DOE rail cask tests collected strain and acceleration data from surrogate PWR fuel assemblies and cladding in an ENSA UNiversal (ENUN) 32P rail cask for three different modes of transportation – heavy-haul truck (245 miles in Spain), ship (coastal barge from Santander, Spain to Zeebrugge, Belgium and ship from Zeebrugge to Baltimore, MD), and rail (1,950 miles from Baltimore to Pueblo, CO) – and during operations simulating the vertical placement of the ENSA ENUN 32P cask onto a surrogate storage pad. In addition, a series of controlled rail tests were performed at the Transportation Technology Center, Inc. (TTCI) in

Pueblo, CO. The combination of three modes of transportation, special rail tests in TTCI, and handling, referred to as the Multi-Modal Transportation Test (MMTT), provided an understanding of the cumulative effects of transportation and handling of SNF. All recorded strains and accelerations on the surrogate fuel rods were exceedingly low during the rail-cask tests for all the transport and handling modes. The test results suggest that spent fuel can withstand thousands of cross-country trips before approaching fatigue or shock failure limits and provide a compelling technical basis for the safe transport of spent fuel under NCT (Kalinina et al. 2018).

The full range of ongoing R&D supporting transportation safety is summarized in Appendix C.2.

3.2.2 Transportation of CSNF from Stranded Sites

As noted in Section 3.1.3, there are nine "legacy" shutdown sites (sites that were shut down by 1997 and have not had an operating reactor on-site for at least 20 years) with an estimated total of 2,815 MTHM of stranded CSNF in dry storage in 248 canisters. For four of the nine sites, a heavy-haul truck or a barge may be needed to transport DPCs to the nearest rail node, because rail access was removed along with the reactor during decommissioning (Rechard et al. 2015, Section 2.2.4). Furthermore, the rail spurs to four other sites will need refurbishing (Maheras et al. 2013, Table S-1).

Stranded CSNF at several of the more recently (after 2010) shutdown sites will have high burnup. Early transportation of this high-burnup CSNF in existing transportation casks may require additional time for cooling and/or loading fewer CSNF assemblies in a cask to meet the NRC limits on accessible surface temperature and/or external surface radiation levels (10 CFR Part 71.43).

3.2.3 Transportation System Logistics

Key considerations for transportation system logistics modeling to support the planning for an efficient and resilient transportation system are described in Appendix E. A transportation system logistics model should balance competing priorities to evaluate the impact of operational decisions on system-level metrics such as resource utilization, safety, security, and cost. Implicit in logistics modeling is the need to address regulatory and technical challenges such as the "storage to transportation to storage" or "72-71-72" sequencing.

Important considerations for the transportation campaign are (1) identification of transportation routes (e.g., from reactor sites and shutdown sites to a repository, and possibly also to a CISF), (2) development of a transportation schedule, and (3) acquisition of the equipment (transportation fleet) required to move the fuel in accordance with the selected schedule. Model results for these aspects of transportation logistics are described in Appendix E. The results show significant sensitivity to factors such as: allocation strategy (how shipments from reactor sites are prioritized), transportation cask/canister system capacity, train "consist" (number of railcars) size, fuel age and burnup, and rolling stock availability.

3.3 Disposal of SNF and HLW

As described in Section 1.1, deep geologic disposal is the preferred option for long-term isolation of SNF and HLW. Many nations are working to develop mined geologic repositories for SNF and HLW, but none are yet in operation. Other disposal options (e.g., deep boreholes for smaller inventories) are also being considered but are at an earlier stage of development.

From 1955 through the 1960s, the AEC studied disposal of HLW and SNF in salt formations, culminating in consideration of a site at an abandoned salt mine near Lyons, Kansas (Rechard et al. 2014). Due to issues with site suitability (a large number of nearby resource exploration boreholes) and local opposition, the Lyons site was abandoned in 1972[15] and AEC announced plans for a Retrievable Surface Storage Facility (RSSF), in which waste could be stored "a minimum of 100 years" and thereby enable the US to "keep open all options" and to "move slowly" to permanent disposition (Rechard et al. 2014, Section 2.2.1; Walker 2009, p. 80). Plans for the RSSF were also abandoned, due to criticism from the Environmental Protection Agency (EPA) and nuclear opponents that the RSSF would be de facto permanent disposal (Rechard et al. 2014, Section 2.2.1).

In 1976, ERDA, successor to the AEC, formed the National Waste Terminal Storage[16] (NWTS) program. The NWTS program, buoyed by acceptance of the concept of using multiple barriers to build a robust disposal system, expanded the range of national repository research to include salt, argillite, and crystalline rock (Freeze et al. 2013, Appendix A-3.1; Rechard et al. 2014, Section 2.2.1).

Also in 1976, the US Geological Survey (USGS) noted that the region in southern Nevada around the Nevada National Security Site (NNSS), then known as the Nevada Test Site (NTS), had several advantages because of its closed groundwater basin, long groundwater flow paths to outflow basins, many types of suitable host rocks, remoteness, past nuclear testing, arid climate, and thick unsaturated zone (Rechard et al. 2014, Section 2.2.2). The NWTS program was subsequently expanded from the original three geologic media to also consider the bedded tuff at NTS and basalt at the Hanford Site.

The NWPA (1983) required DOE to evaluate multiple repository sites for a first and second repository. Nine sites were identified for screening for the first repository: three salt dome sites (Vacherie Dome in Louisiana, Richton Dome and Cypress Creek Dome in Mississippi), four bedded salt sites (Deaf Smith and Swisher in Texas, Davis Canyon and Lavender Canyon in Utah), the basalt site at Hanford, Washington, and the bedded tuff at the Yucca Mountain site at the NTS (Freeze et al. 2013, Appendix A-3.2).

By 1986, DOE had identified five of the sites as potentially acceptable for the first repository and performed a multi-attribute utility decision analysis to rank the sites. DOE then recommended

[15] The early salt formation studies for disposal of HLW and SNF were a precursor to the consideration of bedded salt deposits in southeastern New Mexico as a location for the Waste Isolation Pilot Plant (WIPP) for the mined geologic disposal of TRU waste (waste contaminated with transuranic elements and hazardous chemicals generated during the production of nuclear weapons) (Rechard 2000).

[16] In the 1970s, a mined geologic repository was categorized as a storage option, and a repository was called a terminal storage facility (Rechard et al. 2014).

three sites, in three different geologic media, for further characterization: volcanic tuff at Yucca Mountain (ranked 1st); bedded salt at Deaf Smith County (ranked 3rd); and basalt at Hanford (ranked 5th) (Rechard et al. 2014, Section 2.2.4). DOE also proposed evaluating 12 crystalline (granitic) rock sites in seven states (five in the east, two in the Midwest) for the second repository, but postponed their evaluation because of high characterization costs and because new reactors were not being built (Rechard et al. 2014; Freeze et al. 2013, Figure A-3).

The NWPAA (1987) specified that DOE evaluate disposal of SNF and HLW at Yucca Mountain for the first repository and deferred action on a second repository. As noted in Section 1.2, a License Application for Yucca Mountain was submitted to the NRC in 2008, but the DOE withdrew support for the project in 2010 and Congress suspended funding for the licensing process.

Both the NRC and the EPA have applicable regulations for the disposal of SNF and HLW. General regulations for disposal of SNF and HLW in geologic repositories (10 CFR Part 60 (NRC) and 40 CFR Part 191 (EPA)), were first promulgated in 1983 and 1985, respectively. These general regulations are for a geologic repository sited, constructed, or operated in accordance with the NWPA (but pre-dates the 1987 NWPAA). Site-specific regulations for Yucca Mountain (10 CFR Part 63 (NRC) and 40 CFR Part 197 (EPA) were initially promulgated in 2001 and remain in effect.

Currently, US disposal research, as part of DOE-NE SFWST, is focused on generic repositories in several different geologic media (Mariner et al. 2019), including bedded salt, argillaceous material (e.g., clay and shale), crystalline rock, and unsaturated alluvium. The full range of ongoing disposal research is summarized in Appendix C.3.

3.3.1 Repository Design Considerations

Internationally, a number of different repository designs, for a variety of waste types and volumes, are being considered in several different geologic media (e.g., salt, argillite, crystalline rock). Analyses of long-term repository performance indicate that all of these geologic media can safely host a repository, with appropriate design considerations (Freeze et al. 2013).

A key repository design consideration is thermal management, which is affected by several factors including: waste package temperature and aging, waste package size and spacing, drift/tunnel spacing, long-term ventilation, and geologic setting. Different repository designs can withstand different maximum temperatures, typically constrained by waste or buffer/backfill performance, and often controlled by the thermal conductivity of the host geology (see Section 4.3.2). As a result, thermal management is an important consideration for integration within the back end of the nuclear fuel cycle.

By mid-century, US repository disposal capacity will need to accommodate ~136,000 MTHM of CSNF (Figure 2-3), approximately twice the NWPA limit on the first repository. Based on current estimates, this would require waste packages (overpacks) for an estimated 10,000 large-diameter DPCs (e.g., 32 PWR or 68 BWR) (Section 3.1.2.1) with thermal power (projected to 2048) of 5 to 20 kW/canister (DOE 2014, Figure 3). The disposal capacity would also need to accommodate waste packages for ~13,000 MTHM of DSNF and HLW in smaller canisters with a lower thermal output (Section 2.3). These projected inventories are summarized in Table 3-1; details are provided in Sections 2.2, 2.3, and A.2.5.

Table 3-1. Projected US Inventory of SNF and HLW

Waste Stream	Waste Form	Mass (MTHM)	Canisters (Small Diam.) [a]	Canisters (Large Diam.) [b]	Year
CSNF	PWR and BWR assemblies (spent UO_2 fuel and cladding)	~136,000		~10,000 [c]	2055 – 2075 [d]
DSNF	U metal (from N Reactor) and other SNF waste forms	2,461	~3,500 [e]		2035
Naval SNF	naval spent fuel	65		400	2035
CHLW	borosilicate glass	640	275 [f]		Current
DHLW	borosilicate glass, HIP calcine, and sodium-bearing HLW	9,862 [g]	19,723		N/A

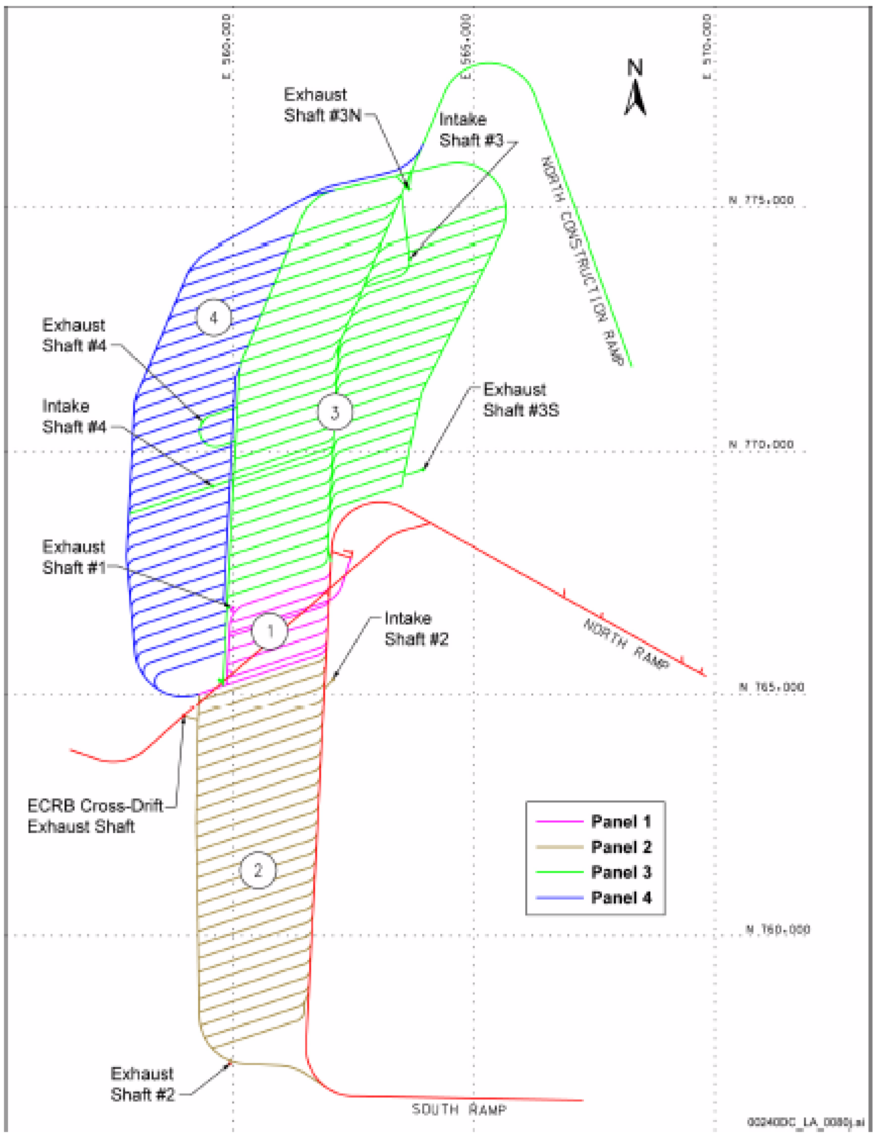
[a] Small diameter is ≤ 0.61 m (2 ft)
[b] Large diameter is ≥ 1.5 m (4.9 ft)
[c] Assumed to be large DPCs (e.g., 32 PWR or 68 BWR), see Section 3.1.2.1.
[d] CSNF Projection is to 2075. However, between 2055 and 2075 only 1 reactor is projected to be in operation, producing ~600 MTHM during that time
[e] The projected number of canisters ranges from 2,500-5,000 (Section 2.3.1)
[f] 275 out of 278 West Valley canisters contain vitrified HLW (Section 2.2.2)
[g] Based on an estimate of 0.5 MTHM/canister used by DOE for design purposes for the Yucca Mountain Repository License Application (DOE 2009a, Section 1.5.1.2.1)

3.3.2 Yucca Mountain Repository Design

The Yucca Mountain Repository License Application (DOE 2009a) was based on the assumption that CSNF would be loaded into large-diameter, stainless-steel transportation, aging, and disposal (TAD) canisters (diameter = 1.69 m, length = 5.38 m) that hold 21 PWR or 44 BWR spent fuel assemblies (Section A.2.5.2). When DOE submitted the License Application in 2008, the amount of CSNF in dry storage was small, and the proposed concept of loading spent fuel assemblies into disposal-ready TAD canisters at the reactor sites made sense (Bonano et al. 2018). Since 2008, the dry storage inventory of DPCs (that are larger-diameter and hotter than TAD canisters) has increased. Nonetheless, an overview of the TAD-based Yucca Mountain design provides insights into future repository design considerations.

Yucca Mountain is located on federal land (the southwest corner of the NNSS) in a remote area of Nye County in southern Nevada. The Yucca Mountain site is in an arid region of the US, approximately 130 km northwest of Las Vegas (DOE 2009a, Section 1.1.1.2). Yucca Mountain consists of successive layers of volcanic tuffs, which were formed approximately 14 to 11.5 million years ago by eruptions of volcanic ash from calderas to the north. The repository horizon is to be located in the unsaturated zone in the densely welded rocks of the Topopah Spring Tuff lithostratigraphic unit, 200 to 500 m below the surface at a depth ~300 m above the water table (DOE 2009a, General Information, Section 5.2). The deep water table and thick unsaturated zone result from a combination of low annual precipitation, low infiltration rates, and high rates of evaporation in the arid environment.

The emplacement areas include four distinct emplacement panels containing the emplacement drifts (Figure 3-5); the drift and panels are to be developed over a period of several years, with waste emplacement operations occurring concurrently with repository development. The emplacement drifts are horizontal, 5.5-m nominal diameter circular tunnels, spaced in a parallel pattern, center to center, at nominally 81 m (DOE 2009a, Section 1.3.1.1). The subsurface layout can accommodate approximately 108 emplacement drifts; they are, on average, ~600 m in emplacement length, with no drift longer than 780 m, and a total available emplacement length of 65,209 m (40.6 mi.) (DOE 2009a, Section 1.3.1.1).



(Source: DOE 2009a, General Information, Figure 2-3)

Figure 3-5. Yucca Mountain Emplacement Panels and Drifts

For a license (to construct) application in 2008, DOE initially estimated that waste receipt and emplacement operations could begin in 2017, following construction authorization (estimated in 2012) and a license to receive and possess (estimated in 2016) (DOE 2008a, General Information, Figure 2-1). DOE (2009a, General Information, Figure 2-1) revised the estimate to 2020 for first disposal operations, although that was prior to the suspension of the licensing process. Receipt and emplacement operations are projected to span 50 years (DOE 2009a, General Information, Section 2.2), followed by a 50-year period of additional preclosure ventilation (DOE 2009a, Section 1.1.8.4.2).

The Yucca Mountain Repository License Application included both commercial waste (63,000 MTHM) and DOE-managed waste (7,000 MTHM). Specifically, DOE allocated 10% of the mass of the 70,000 MTHM first repository to DSNF and HLW, with ~1/3 (2,333 MTHM) for DSNF and ~2/3 (4,667 MTHM) for HLW (Lytle 1995; Rechard and Voegele 2014). The naval SNF (65 MTHM) comes out of the 2,333 MTHM DSNF allocation; the commercial HLW at West Valley (640 MTHM) comes out of the 63,000 MTHM commercial allocation (DOE 2009a, General Information, Section 1.2.1). Inventories of CSNF, DSNF (including naval SNF), and HLW planned for disposal at the Yucca Mountain repository are summarized in Table 3-2.

Table 3-2. Projected Yucca Mountain Inventory of SNF and HLW

Waste Form	**Mass (MTHM)**	**Canisters (Small Diam.)** [a]	**Canisters (Large Diam.)** [b]	Waste Packages Needed[i]
CSNF	~63,000 [f]		~7,500 [c]	~7,500 CSNF
DSNF	2,268	~3,500 [d]		~3,500 codisposal [h]
Naval SNF	65		400	~400 Naval
CHLW [e]	640 [f]	275		[h]
DHLW	4,667	~9,334 [g]		[h]
Total	**70,000** [f]			**~11,400**

[a] Small diameter is ≤ 0.61 m (2 ft)
[b] Large diameter is ≥ 1.5 m (4.9 ft)
[c] TAD canisters (21 PWR or 44 BWR)
[d] Includes ~3,100 standard DSNF canisters and ~400 MCOs (DOE 2009a, Section 1.5.1.3.1.2).
[e] From commercial reprocessing at West Valley, 275 out of 278 canisters contain vitrified HLW (Section 2.2.2)
[f] CHLW counts against the 63,000 MTHM limit for CSNF
[g] Based on an estimate of 0.5 MTHM/canister used by DOE for design purposes for the Yucca Mountain Repository License Application (DOE 2009a, Section 1.5.1.2.1)
[h] Codisposal waste packages include combinations of DSNF and HLW
[i] The actual number of waste packages to be disposed in uncertain. The Yucca Mountain Repository License Application and supporting documents provide additional estimates (see Table A-4).

(Source: DOE 2009a, Table 1.5.1-1)

In accordance with the Yucca Mountain Repository License Application, the 63,000 MTHM CSNF inventory of PWR and BWR assemblies would be placed into approximately 7,500 TAD canisters (21-PWR or 44-BWR) before emplacement in the repository (DOE 2009a, General Information, Section 1.2.1). This is only about half of the projected (to 2075) US CSNF inventory (Table 3-1). Some CSNF, including DSNF of commercial origin, would also be received as uncanistered SNF in casks (e.g., bare fuel casks) and some CSNF may be received in DPCs (DOE 2009a, General Information, Section 1.1.3.1).

The heterogeneous DSNF inventory, predominantly uranium metal from the Hanford N Reactor with a smaller amount of other SNF waste forms, would be in standard DSNF canisters or MCOs, ranging in number from 2,500 to 5,000 (with a nominal value of 3,500) (DOE 2009a, Section 1.5.1.3). The naval SNF would be in 400 naval SNF canisters. The total projected (to 2035) US DSNF inventory of 2,461 MTHM (Table 3-1), includes approximately 174 MTHM of commercial origin, which is included in the 63,000 MTHM commercial allocation, leaving a DSNF inventory (including 65 MTHM of naval SNF) which is similar to the DSNF allocation of 2,333 MTHM (DOE 2009a, General Information, Section 1.2.1).

The DHLW inventory, predominantly vitrified borosilicate glass with a small amount of HIP calcine and sodium-bearing HLW, would be in HLW canisters (Table A-3). An estimated 9,334 DHLW canisters would fill the DHLW allocation of 4,667 MTHM (Table 3-2). This is less than half of the projected US DHLW inventory (Table 3-1). The majority of DHLW comes from the defense nuclear program, of which a small quantity (approximately 900 canisters) may also contain vitrified glass with plutonium arrayed within the vitrified HLW (DOE 2009a, General Information, Section 1.2.1). In addition, there are 275 canisters of HLW from reprocessed CSNF at the West Valley Demonstration Facility, which come out of the 63,000 MTHM commercial allocation (DOE 2009a, General Information, Section 1.2.1).

DOE expects the repository to accommodate the following approximate annual waste receipt rates: 3,000 MTHM of CSNF and commercial HLW, 763 canisters of DHLW, up to 24 canisters of naval SNF, and 179 canisters of DSNF (DOE 2009a, General Information, Section 2).

Disposal of each of these waste forms and canisters requires a disposal overpack (waste package). For the Yucca Mountain Repository License Application, three waste package types were designed to accommodate the waste (Figure 3-6).

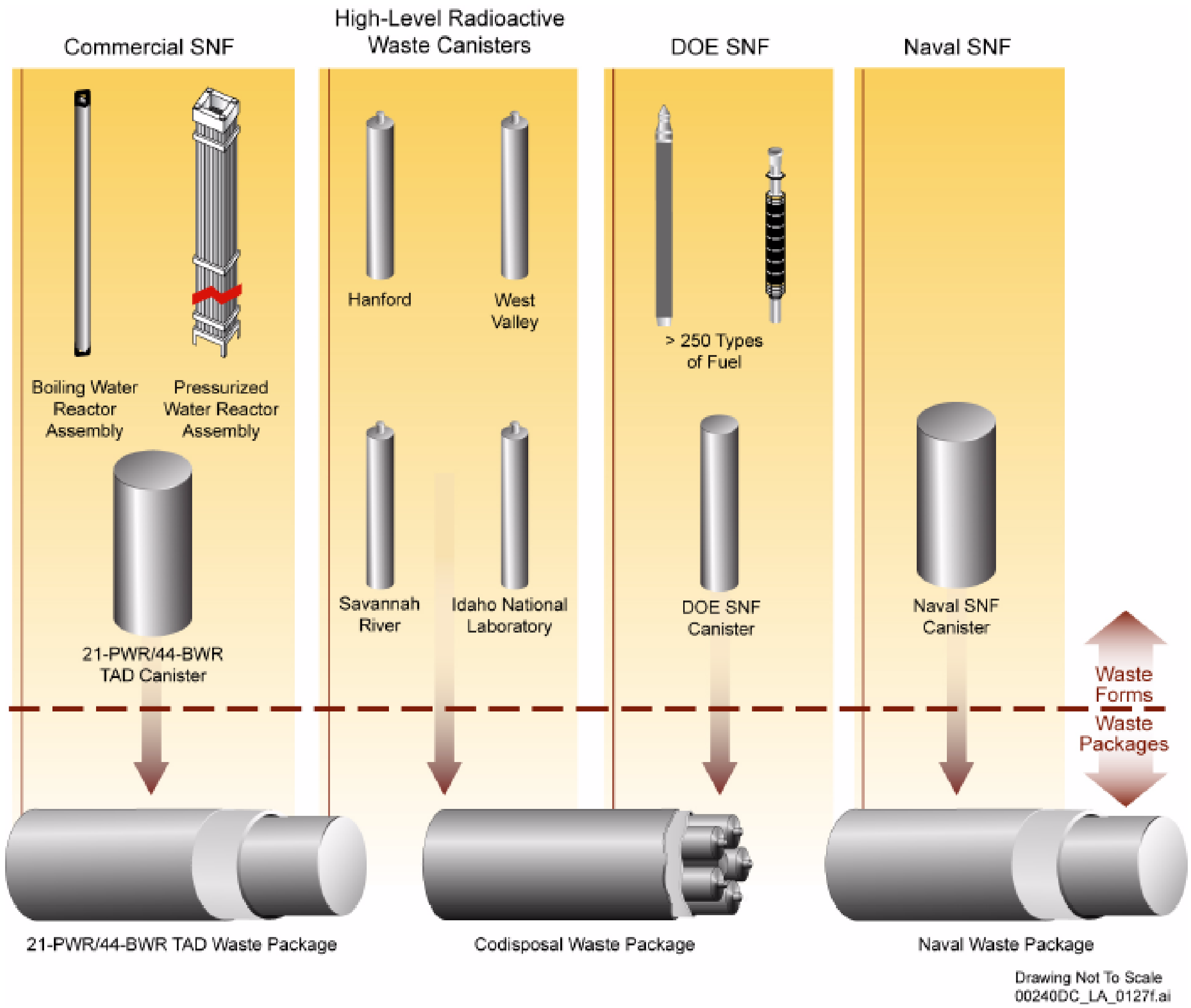
The different waste package types have multiple configurations and internal structures and different external dimensions to allow acceptance of the various waste forms and canisters (see Section A.2.5 and Table A-4):

- CSNF Waste Package – CNSF in a single TAD canister (1 configuration)
- Naval SNF Waste Package – naval SNF in single naval canister (2 configurations: Short and Long)
- Codisposal Waste Package – multiple canisters of DSNF and/or HLW (3 configurations: 5-DHLW/1 DSNF Short; 5-DHLW/1 DSNF Long; 2 DHLW/2 MCO)

Each type of waste package has a similar design, consisting of two concentric cylinders in which the waste canisters are placed. The inner cylinder is stainless steel and includes welded top and bottom inner lids. The outer cylinder is an Alloy 22 (a corrosion resistant, nickel-based alloy) corrosion barrier and likewise includes welded top and bottom outer lids.

The Yucca Mountain Repository License Application design did not include disposal of CSNF in DPCs. Ongoing R&D is examining the feasibility of disposing CSNF in DPCs (Section 5.1.2), with a corrosion-resistant disposal overpack that could be of similar design to Yucca Mountain

CSNF waste package, as an alternative to repackaging the CSNF into TAD or other disposal-specific canisters (SNL 2021a).



Note: the TAD canister and TAD waste package for CSNF and the codisposal waste package for DSNF and DHLW are based on the Yucca Mountain License Application design (DOE 2009a). Future designs may be different (e.g., they might include canisters such as DPCs for CSNF).

(Source: DOE 2009a, Figure 1.5.2-1)

Figure 3-6. Representative Waste Forms, Canisters, and Packages

The preclosure safety analysis (PCSA) (DOE 2009a, Section 1.8) and postclosure total system performance assessment (TSPA) (DOE 2009a, Section 2.4; SNL 2010) demonstrated compliance with the repository performance objectives. The NRC staff review of the Yucca Mountain Repository License Application, including the PCSA and postclosure TSPA, concluded that, although certain land ownership and water rights requirements have not yet been met, the repository meets its requirements for preclosure and postclosure safety. (NRC 2015),

3.3.2.1 Thermal Management

Thermal management of the waste packages is necessary to maintain the postclosure integrity of engineered barrier components and minimize altered conditions in the near-field host rock. Yucca Mountain repository subsurface temperature design limits include (DOE 2009a, Table 1.3.1-2):

- Drift wall ≤ 200°C
- WP surface ≤ 300°C for the first 500 years
  ≤ 200°C for the next 9,500 years
- CSNF clad ≤ 350°C at emplacement

Part of the thermal management strategy for the Yucca Mountain repository is to (1) emplace codisposal waste packages in the same drifts with CSNF waste packages, using the lower heat output of interspersed DHLW and DSNF waste to dilute the higher heat output of CSNF, and (2) include an aging pad (buffer storage) to facilitate cooling of CSNF and "blending" of hotter and cooler CSNF waste packages for disposal (Rechard et al. 2015, Section 2.3). The Yucca Mountain design also includes a Wet Handling Facility for fuel repackaging, as needed (DOE 2009a, Section 1.2.5).

The design reference case for loading waste packages in a drift is based on a seven-waste-package segment (six whole waste packages and two half waste packages), consisting of different types of codisposal and CSNF waste packages with different thermal power content (DOE 2009a, Section 1.3.1.2.5). The design included the following thermal parameters (DOE 2009a, Section 1.3.1.2.5):

- Maximum waste package thermal power at emplacement = 18.0 kW
- Maximum linear heat load at emplacement over the length of a seven-waste-package segment = 2.0 kW/m
- Nominal spacing between adjacent waste packages, averaged over a seven-waste-package segment = 10 cm

The aging facilities provide safe cooling of CSNF within TAD canisters and DPCs, in aging overpacks or horizontal aging modules, until the thermal heat load of the CSNF has decayed to a level low enough to be placed in a waste package. The aging facilities will have a maximum total aging capacity of 21,000 MTHM in 2,500 aging spaces (DOE 2009a, General Information, Section 1.1.2.1).

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4. OBSERVATIONS ON CURRENT US PRACTICE FOR THE MANAGEMENT OF COMMERCIAL SNF

The current state of the US nuclear waste management program is characterized by a lack of integration among storage, transportation, and disposal. The driving factor is the failure to develop permanent disposal capability, which is not only at the heart of the government's inability to honor its waste management obligations under the Standard Contract, it is – especially after the 2011 events at Fukushima – a source of renewed concern to the general public, a growing liability to taxpayers, and a burden to nuclear utilities, their ratepayers, and the nuclear energy industry's prospects going forward (BRC 2012, Chapter 4).

Three observations about current US practice for the management of CSNF provide a foundation for storage, transportation, and disposal planning:

- Current practice is safe and secure
- Current practice is optimized for at-reactor storage operations
- Current practice is not optimized for transportation or disposal of SNF

Two additional factors also influence the integration of storage, transportation, and disposal:

- Regulatory and stakeholder considerations
- Total system life cycle costs

Each of these observations and factors is discussed in more detail in the following sections.

4.1 Current Practice is Safe and Secure

As determined by the NRC, current storage practices for CSNF are safe and secure, and associated risks to the public are extremely small (NRC 2007; NRC 2014a). So long as storage facilities are properly maintained, institutional controls remain in place, and regulatory oversight and enforcement remains effective, public safety standards can and will be met.

Similarly, the results of the NRC risk assessment affirm that, under NCT and HAC, radiological impacts from SNF transportation conducted in compliance with NRC regulations are low, and that regulations for transportation of radioactive material are adequate to protect the public against unreasonable risk (NRC 2014b).

All 3,158 DPCs containing CSNF in storage at the end of 2019 are NRC-certified for storage and ~85% are NRC-certified for transport (Section A.2.3). There is no question, however, that extending storage operations decades longer than originally intended raises questions about spent fuel and canister performance and subsequent transportability over longer time periods. Current trends in industry are to irradiate fuel longer in reactors, achieving higher levels of burnup; higher-burnup fuel has higher thermal output, is more radioactive, and may result in different fuel and cladding behavior during storage, transportation, and disposal. Furthermore, much of the available data on fuel performance in storage are from older relatively low-burnup fuel.

In recent updates to the "Waste Confidence Decision," the NRC findings for "Continued Storage of Spent Nuclear Fuel" (NRC 2014a) suggest that commercial spent fuel can be stored safely and without significant environmental impacts for at least 60 years beyond the licensed life for operation, which could include an initial license period of up to 40 years and license extensions of up to 40 more years (Appendix B). These findings, which suggest safe storage times exceeding 100 years, are based on the assumptions that (1) spent fuel canisters and casks would be replaced approximately once every 100 years, (2) a dry transfer system (DTS) would be built at each ISFSI location for fuel repackaging, and (3) ISFSI and DTS facilities would also be replaced approximately once every 100 years. The technical basis for these findings is that fuel and cladding are expected to have low degradation rates in dry storage and that dry cask storage systems are inherently robust and resistant to damage (NRC 2014a, Appendix B.3.2). However, the NRC also notes that high-burnup fuel may be subject to increased degradation and that ongoing research into the extended storage of spent fuel is part of the effort to continuously evaluate and update safety regulations (NRC 2014a, Appendix B.3.2.1).

Industry, NRC, and the DOE are conducting ongoing R&D activities to confirm storage-system performance, to identify monitoring and aging-management practices needed to ensure continued safety of future operations (NRC 2019b), and to assess changing needs for site-specific physical security as fuel ages and reactors are decommissioned. There is DOE-NE SFWST sponsored R&D focused on evaluating (1) the integrity of cladding, including high-burnup fuel, during extended storage, (2) the integrity of storage canisters during extended storage, including the potential for chloride-induced stress corrosion cracking (SCC), (3) the transportability of CSNF, including high-burnup fuel, following extended storage, and (4) the potential to dispose of CSNF in DPCs in a repository without repackaging (Hanson and Alsaed 2019; Teague et al. 2019; Saltzstein et al. 2020; Sassani et al. 2020; SNL 2021a). This ongoing R&D is described in more detail in Appendix C.

4.2 Current Practice is Optimized for At-Reactor Storage Operations

Utilities that operate nuclear power plants in the US must meet all operational safety and security requirements while fulfilling obligations to public utility commissions, ratepayers, and shareholders to provide electric power at competitive costs. Activities associated with the at-reactor storage of CSNF must therefore ensure full safety for site workers and the general public while causing the least disruption to the cost-effective operations of the reactor. In practice, these constraints favor loading large DPCs that require the fewest on-site handling operations to minimize reactor down time. Canister sizes have increased in the past decade; the largest DPCs, which currently hold 37 PWR or 89 BWR assemblies, are made possible by advanced cask/canister system designs that are very efficient in passively removing heat from the canister. Canister size is ultimately limited by practical considerations associated with on-site handling and transportation. DPCs with up to 37 PWR assemblies can initially produce ~40 kW per canister (Hardin et al. 2015, Section 1).

The current situation, where utilities own CSNF for at-reactor storage and DOE takes title for transportation and disposal, does not provide any incentive to the utilities to implement waste packaging and/or storage practices that would also contribute to more efficient transportation or disposal practices. This situation is exacerbated by the payments to the utilities from the US

Treasury Judgment Fund (Section 1.1) to maintain on-site storage as a result of the federal government not meeting the obligations of the Standard Contracts.

4.3 Current Practice is Not Optimized for Transportation and Disposal

The current practice in the US of loading large DPCs is appropriate given the financial, legal, and regulatory constraints imposed on the utilities that generate and manage CSNF. However, the practice may not be an optimal solution for the full back end of the nuclear fuel cycle, including interim storage at operating reactor sites, subsequent transportation to either consolidated interim storage sites or repositories, and eventual permanent disposal. In part, this potential for a suboptimal CSNF management system has its roots in the NWPA, which does not transfer ownership of, and management responsibility for, CSNF from the utilities to the federal government until it is accepted for permanent disposal (i.e., a repository is licensed). Long delays in the availability of a repository have also contributed to the lack of integration. When the NWPA was written, no CSNF was in dry storage, and planning for transportation and disposal assumed handling operations would begin with bare fuel assemblies. As shown in Figure 2-3, when DOE submitted the Yucca Mountain Repository License Application in 2008, the total amount of CSNF in dry storage was still small (~12,000 MTHM in about 1,100 DPCs), and the proposed concept of loading spent fuel assemblies into disposal-ready TAD canisters at the reactor sites made sense. In the absence of a federal commitment to Yucca Mountain or any other disposal concept, utilities will continue to load large DPCs and the disconnect between the system envisioned in the NWPA and operational reality will continue to grow.

The large DPCs are designed to meet thermal constraints during dry storage (e.g., maximum fuel cladding temperature of 400°C); however, transporting and disposing of these large canisters can be problematic because of their size. Placing a large number of spent fuel assemblies in a single canister increases the temperature and thermal output of the canister. As a result, some of the larger DPCs can be too hot to transport because the thermal output of the waste produces surface temperatures on the transportation cask that exceed licensing limits (50°C). Thus, the waste must be stored, sometimes for several decades, until it is cool enough to be transported (see Section 4.3.1).

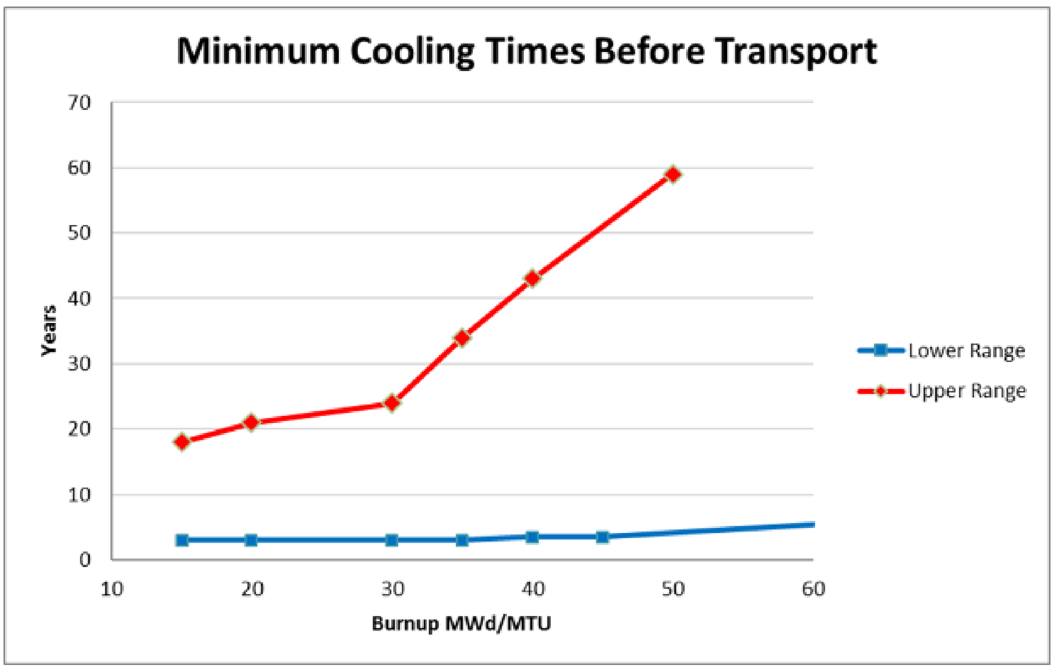
Furthermore, higher-burnup fuel exacerbates the thermal issues outlined above because the thermal output of SNF increases as burnup increases. The fuel cooling time for any given value of burnup depends primarily on the canister loading configuration and the cask/overpack design. Small canisters can be loaded sooner. A large DPC with high-burnup fuel may require a significantly longer cooling time in storage prior to transportation. Higher-burnup fuel is also more radioactive; larger DPCs with high-burnup fuel may produce a higher radiological dose, requiring additional storage time to meet the dose limit for transportation.

In general, extended storage operations (decades longer than originally intended) and higher-burnup fuel in storage raises questions about spent fuel and canister performance and subsequent transportability over longer time periods. Ongoing research to address these considerations is described in Sections C.1 and C.2.

4.3.1 Implications of Current Practice for Transportation

DPCs are large and are both hot and heavy when fully loaded. Technical challenges associated with moving massive casks are relatively straightforward and can be addressed using available technology, but the magnitude of the enterprise should not be underestimated. Even if consolidated interim storage or disposal sites were available and ready to receive CSNF, some dry storage containers in use today are not certified for transport, and transportation routes from some storage sites are uncertain (Jones 2016; Maheras et al. 2017). Rail cars remain to be designed, tested, and built (NWTRB 2019), some transportation casks remain to be certified by the NRC, and in some cases DPCs may need to remain on-site for further decay cooling before they reach thermal and dose limits established for transportation (Jones 2016).

As an example of the thermal implications of DPC loadings with respect to transportation, Figure 4-1 shows the range of required aging times established by the NRC in its certificates of compliance for various transport cask configurations. Aging times are specified as a function of fuel burnup, and the upper limits assume maximum loading density in the canisters. Aging times vary with the specific details of the cask/canister system (some are more efficient at heat transfer than others) and the assumptions made regarding loading schemes.



As reported in NRC certificates of compliance for specific designs as of 2013. Variation in times is due to the diversity of cask/canister system designs, and is dominated by DPC size and heat transfer capabilities.

(Source: Modified from Stockman and Kalinina 2013)

Figure 4-1. Minimum Cooling Times for Multiple Cask/Canister Systems

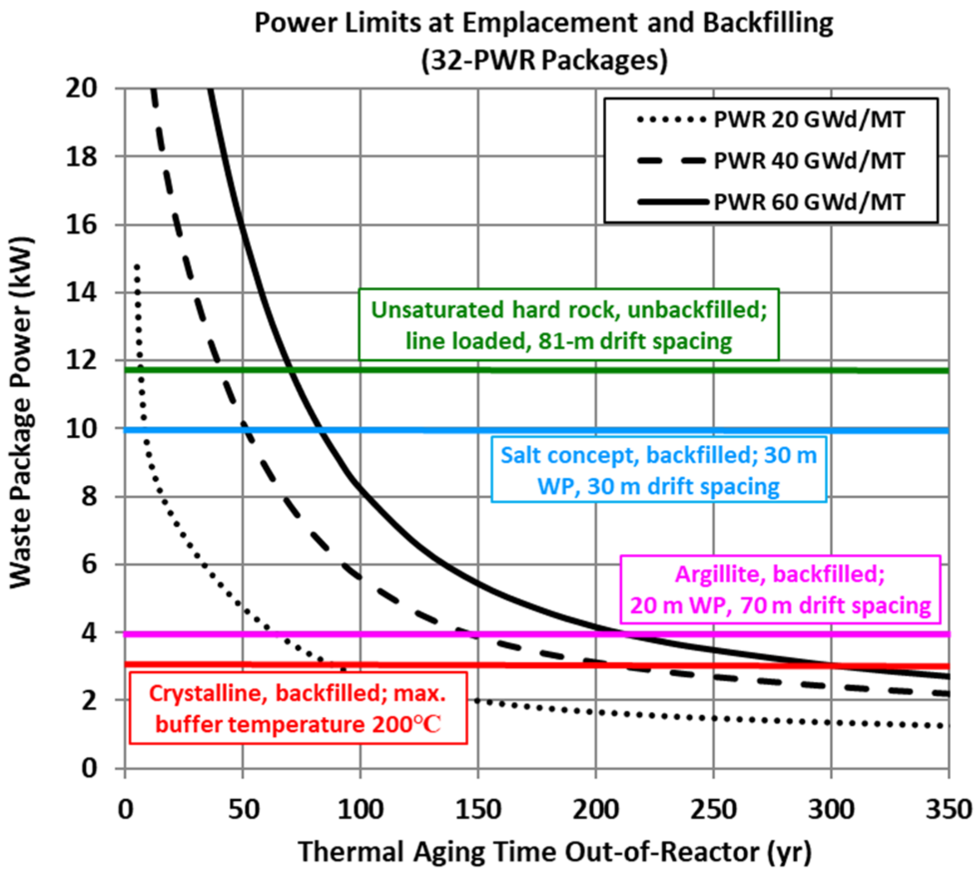
Figure 4-1 can be interpreted as showing that some cask/canister systems can be transported as soon as 6 or 7 years after fuel is removed from the reactor regardless of the fuel's burnup, but other systems may require decades of aging before they can be moved. Note, however, that these are upper bounds specified in the NRC's certificates of compliance for maximally-loaded canisters. In practice, site operators tend to load canisters with lower heat loads, and few, if any, existing canisters will approach the upper limits.

4.3.2 Implications of Current Practice for Disposal

DPCs are substantially larger and heavier than waste packages considered in any current disposal concept design other than the proposed Yucca Mountain repository, and they are incompatible with disposal concepts being considered in nations other than the US. For example, Swedish (SKB 2011), Finnish (Posiva 2012), and French (Andra 2005) concepts call for disposal of spent fuel in waste packages containing four PWR assemblies (or equivalent amounts of other fuel types), rather than the 32 or more PWR assemblies typical of US DPCs. Necessary modifications to existing repository design concepts other than Yucca Mountain to accommodate the size and mass of DPCs could be extensive, including changes in hoist or ramp design, emplacement drift dimensions, and emplacement transporter technology, but they appear to be achievable using current engineering technology (Section 5.1.2.2; Hardin et al. 2015). Modifications to disposal concepts similar to the Yucca Mountain design could be more straightforward, given the larger size of the TAD canister specified for the repository (21 PWR assemblies per waste package) and the use of rails and an inclined ramp for waste emplacement.

The greater thermal power of large waste packages may pose a more significant challenge to most existing disposal concept designs because the difficulty of transferring decay heat away from packages after emplacement may cause temperatures to rise above intended limits (Hardin et al. 2011; Hardin et al. 2015). Figure 4-2 shows representative estimates of aging times required before repository closure for a hypothetical waste package containing 32 PWR assemblies to meet temperature constraints for representative repository designs in unsaturated hard rock (unbackfilled), salt (backfilled with salt), argillite/clay/shale (with a bentonite buffer), and crystalline/granite (also with a bentonite buffer). Decay curves indicate the thermal power of waste packages containing low-, medium-, and high-burnup fuels. Horizontal lines indicate modeled thermal power limits associated with peak waste package temperatures for each repository design concept, and the intersections of the thermal decay curves with the horizontal power limits indicates the aging time required before emplacement for each disposal concept.

The thermal power limits are based on waste package surface temperature limits of either 100°C or 200°C (Hardin 2020). A limit of 100°C is hypothesized for disposal concepts with a bentonite buffer (i.e., argillite and crystalline), which represents a conservative approach to ensuring that properties of the buffer surrounding the waste package in these concepts are not degraded. It is consistent with limits assumed for existing argillite and crystalline disposal concepts in France, Sweden, and Finland. For the crystalline concept in Figure 4-2 the limit is relaxed to 200°C; a waste package surface temperature limit of 100°C would result in a thermal power limit of less than 2 kW for the assumed spacing of 20 m for waste packages and 70 m for drifts. For the argillite concept in Figure 4-2 most regions of the backfill remain below the 100°C temperature limit, but temperatures of up to 250°C occur in some regions of the backfill near the waste package surface (Stein et al. 2020, Section 4.5).



Cooling times are for a hypothetical waste package containing 32 PWR assemblies, shown for low, medium and high burnup fuel in various repository design concepts

(Source: Hardin 2020; SNL 2021a, Figure 7)

Figure 4-2. Cooling Time Needed Before Repository Closure to Meet Waste Package Surface Temperature Constraints

A higher waste package surface temperature limit of 200°C is assumed for the hard rock concept without a buffer (of which Yucca Mountain is the only well-documented example), consistent with design concepts that allow peak temperatures in the wall rock to go above boiling. The higher 200°C limit for salt reflects the absence of a bentonite buffer, the relatively high thermal conductivity of salt, and is safely below the temperature at which crystalline salt decrepitates.

Figure 4-2 should be interpreted with caution because temperature estimates are strongly dependent on the repository layout (i.e., spacing between waste packages within an emplacement drift and spacing between drifts), fuel burnup and waste package loading, and assumptions about the rate and duration of repository ventilation after emplacement and before drifts are sealed. Aging times can be significantly reduced by increasing spacing between waste packages and/or ventilation after emplacement. With these caveats noted, however, qualitative observations from Figure 4-2 are straightforward.

- Higher-burnup fuels will require longer cooling times than low-burnup fuel (either with above ground storage/aging or with ventilation in the repository) before the repository can be sealed.
- Disposal concepts that require bentonite backfill (e.g., argillite and crystalline) are likely to require unrealistically long aging times for disposal of waste packages containing 32 PWR assemblies, regardless of burnup, primarily because heat retention due to the relatively low thermal conductivity of the backfill.
- Disposal of older low-burnup fuel in large waste packages may be feasible with little or no additional aging in either salt (primarily because of its relatively high thermal conductivity) or hard rock designs that do not call for backfill.
- Large waste packages containing high-burnup fuel are likely to require many decades of aging before disposal in any concept.

Disposal of some already-loaded DPCs without repackaging the CSNF may also pose complications for licensing analyses of postclosure criticality control because neutron absorber materials used in existing designs are aluminum-based and will degrade with long-term exposure to groundwater. In all cases, the potential for criticality is negligible as long as waste packages remain intact and water does not reach the spent fuel assemblies. If packages fail following disposal and fill with water, the potential for criticality rises and can become a concern once neutron absorber plates degrade. Options for assuring postclosure criticality control without opening existing DPCs include corrosion-resistant disposal overpacks that limit the potential for waste package failure, analysis of uncredited reactivity margin associated with the specific loading details of each DPC, partial reliance on neutron absorption by chloride in saline groundwaters, and injectable fillers for moderator exclusion and/or neutron absorption (SNL 2021a). Further description of options for disposal of DPCs, including possible design modifications for future loading, is provided in Section 5.1.2.

4.4 Regulatory, Legal, and Stakeholder Considerations

In addition to the technical challenges related to storage, transportation, and disposal there are also regulatory, legal, and stakeholder factors to be considered in developing integrated solutions to nuclear waste management. These include, but are not limited to, evaluations of relevant regulatory considerations (e.g., needs for new or modified regulations or legislation), legal sensitivities (e.g., ongoing litigation between the federal government and nuclear utilities related to non-performance on Standard Contracts), and socio-political influences (government and public opinion towards nuclear waste and potential storage or disposal sites).

4.4.1 Regulatory and Legal Factors

The NWPAA (1987) identified Yucca Mountain as the only site for evaluation as a repository. However, the Yucca Mountain repository licensing process remains suspended with no funding, Congress has made no change to the law, and the US government is considering no other sites for disposal. If Congress chooses to permanently abandon the Yucca Mountain site and pursue as-yet-unidentified alternatives, additional revisions or amendments to the NWPA would be necessary.

In the NWPA, the nuclear utilities have the primary responsibility for providing at-reactor interim storage of CSNF until it is accepted by the federal government, which is responsible for transportation, away-from-reactor storage (if applicable), and disposal. Without a permanent disposal option this separation of responsibilities has led the utilities, driven by economic and worker safety considerations, to load the ever-increasing inventory of CSNF into larger DPCs that are not designed for disposal. The longer-than-planned storage of the larger DPCs will lead to the need for ISFSI license extensions, potentially delay transportation, and complicate disposal.

The DOE has still not begun to take title to CSNF, and taxpayers, via the Judgment Fund, continue to pay utilities for costs of on-site management of CSNF. The trend of the last decade for older and less profitable nuclear power plants to close seems likely to continue, leaving increasing quantities of CSNF stored at sites where there is no operating reactor.

A federal CISF could alleviate some of the logistical and taxpayer liability issues associated with extended storage at the operating and shutdown reactor sites. However, the NWPAA (1987, Section 148(d)) restricts construction of a federal CISF (which would be limited to 15,000 MTHM anyway) under the provisions for MRS until a license for construction of a repository has been issued. This legislative restriction would need to be removed to allow the construction of a federal CISF independent of the progress on a repository site. Private sector CISFs are under consideration (Section 3.1.4) but are not yet NRC-licensed and their economic viability is uncertain.

Finally, developing a uniform set of regulatory requirements for CSNF that spans the storage, transportation, consolidated interim storage (if used), and repository operation phases is an area of useful regulatory integration. For example, as described in Section 3.2, there are challenges associated with the storage-to-transportation-to-storage "72-71-72" sequencing for system component inspections (e.g., canister, fuel, cladding integrity). To include disposal operations, this sequencing could be "72-71-63" or "72-71-72-63", where "63" refers to a disposal regulation (currently 10 CFR Part 63 for Yucca Mountain).

4.4.2 Stakeholder Factors

Public perceptions and preferences about nuclear energy and nuclear waste can and do influence waste management policy, and thereby also influence attempts to integrate the storage, transportation, and disposal of SNF and HLW. In general, for many Americans the risk perception associated with nuclear energy and radiation is heightened as compared to other kinds of risks. Broad public support for a new strategy or project will depend on some shared understanding of the nature and extent of the problem, available options for resolving the problem, and the consequences and risks associated with different actions – including the consequences and risks of further inaction (BRC 2012, Chapter 2.3.5).

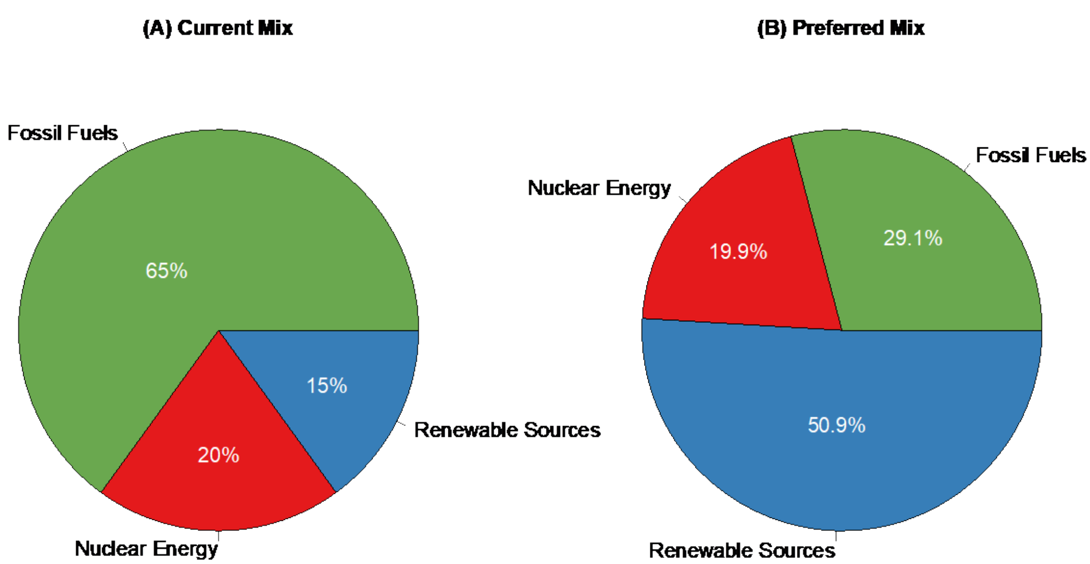
As a result, engagement with all affected units of government, including the host state or tribe, regional and local authorities, and the host community (collectively, the "stakeholders") can enhance the likelihood of success in developing nuclear waste management policy and/or siting a facility.

US public perceptions about nuclear power and nuclear waste management, summarized in Section 4.4.3, suggest that the US public would prefer to be able to continue to rely on nuclear energy to provide ~20% of the electricity for the next several decades. Members of the public generally favor permanent disposal over continued on-site or interim storage and express more support for mined geologic facilities than for surface or deep borehole options. They currently place considerable trust in experts from universities, the National Academy of Sciences (NAS), and the national laboratories to provide technical information about management of nuclear wastes.

4.4.3 Public Perceptions of Nuclear Power and Waste Management

This section provides some insights into the US public's perception of nuclear power and waste management, based on national public opinion surveys. Public perceptions and preferences about nuclear energy and nuclear waste management are likely to influence support for any current and future policies, and therefore merit attention. The following trends and findings are from data collected between 2006 and 2017, and speak to issues of preferences for US energy mix, awareness of current storage and disposal policy, level of support for short and long-term storage/disposal, trust in the information provided by key actors in the waste management domain, and likelihood of participation during transportation and siting efforts.

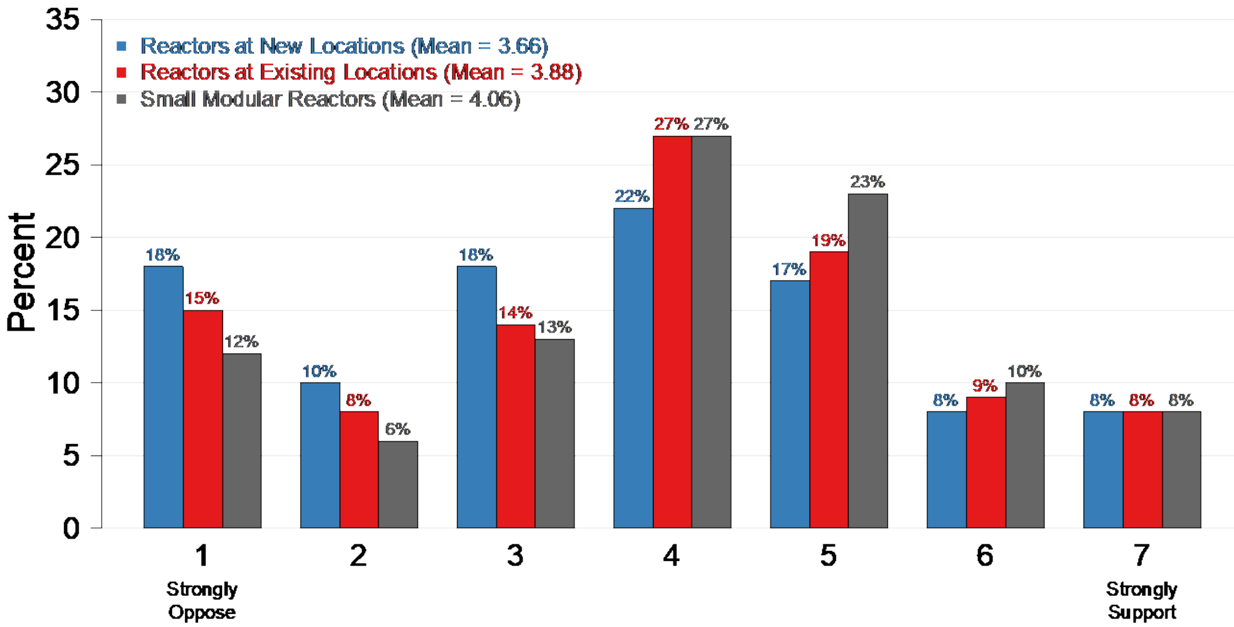
In a 2017 national survey, members of the US public were provided information about the current mix of electricity sources and asked to indicate their preferred mix over the next 20 years. As shown, the US public prefers to increase the share of electrical power coming from renewables (wind, solar, hydropower, geothermal) and to decrease the share coming from fossil fuels; the public prefers that nuclear power continue to provide 20% of electrical power over the next 20 years (Figure 4-3).



(Source: Jenkins-Smith et al. 2018, Figure 2)

Figure 4-3. US Public Preferences for Electrical Power Sources in the Next 20 Years

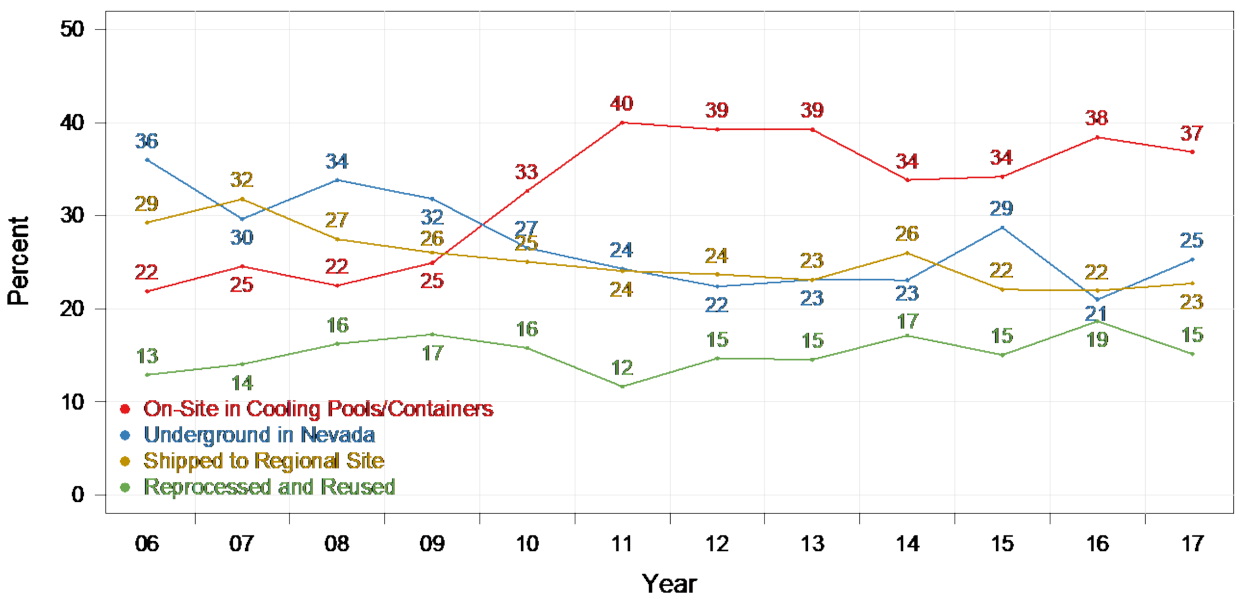
Given the age of current infrastructure, new reactors are necessary to maintain the share of electricity coming from nuclear power over the coming decades. There are a number of options for the types of nuclear reactors that can be constructed, ranging from traditional BWRs and PWRs to new designs like small modular reactors (SMRs). To gauge preferences for different reactor types, members of the US public were asked to indicate their level of support for constructing new traditional reactors and SMRs in the US. The results (shown in Figure 4-4), from a 2017 survey, indicate that, on average, support for SMRs is higher than for constructing traditional reactor types at either new or existing power plant locations.



(Source: Jenkins-Smith et al. 2018, Figure 3)

Figure 4-4. US Public Preferences for Constructing New Nuclear Reactors in the US

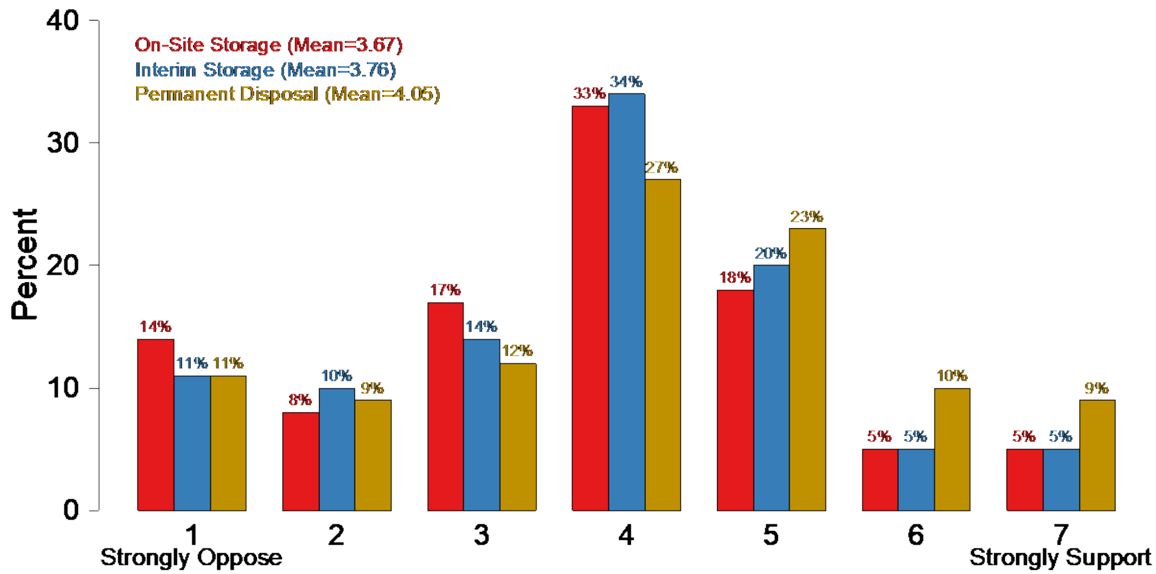
When exploring public support for SNF management options, it is important to consider public awareness of current practices for storage of spent fuel. Figure 4-5 displays awareness of current policy over time (2006-2017), where respondents were asked to indicate what they thought is being done with most of the spent fuel in the US. Findings indicate that a relatively small portion of the US public is aware of current policy for radioactive waste management. Note that focusing events such as the tsunami-initiated reactor meltdowns at the Fukushima nuclear complex in Japan can increase public awareness (as seen by the increased awareness about on-site storage in 2011).



(Source: Jenkins-Smith et al. 2017, Figure E-6)

Figure 4-5. Public Awareness of US Waste Management Policy

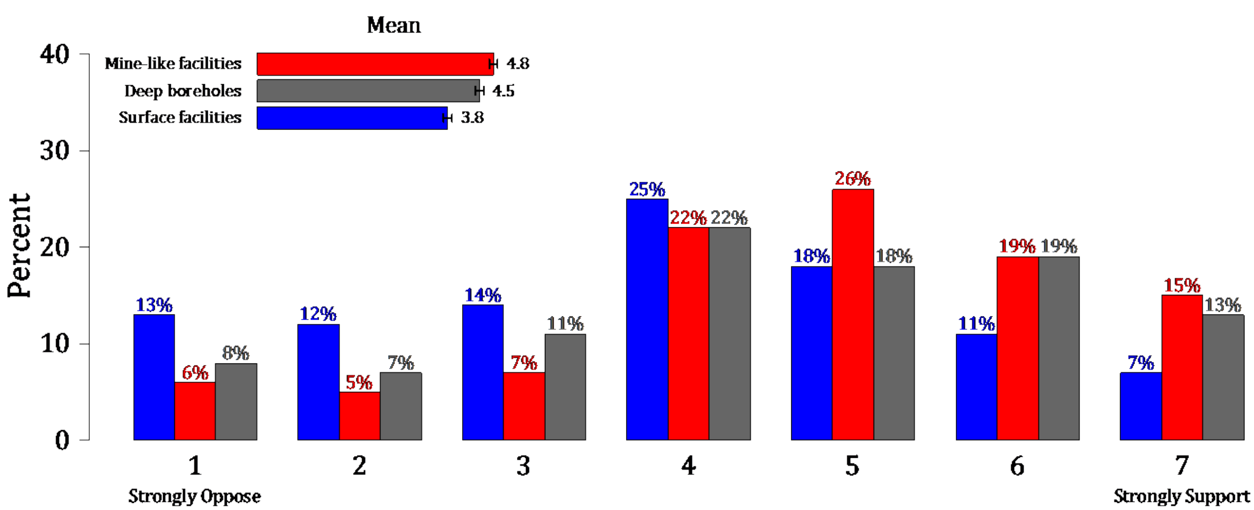
Section 3 of this report outlines the current storage practices for SNF and HLW, proposals for consolidated storage, and plans for disposal of these materials. To gauge public perceptions about these storage and disposal options, a 2017 national survey asked members of the public to indicate their level of support for on-site storage, interim storage (consolidated), and permanent disposal. Findings (shown in Figure 4-6) indicate that members of the public are most supportive of permanent disposal and least supportive of on-site storage.



(Source: Jenkins-Smith et al. 2018)

Figure 4-6. Public Support for Three Different Storage/Disposal Options

Storage and disposal options for SNF and HLW may include a variety of features. One feature relates to the depth of storage/disposal. Options range from surface storage with ready recovery and retrieval, mine-like repositories with recovery difficult but retrieval still possible, and deep borehole disposal with very difficult recovery and retrieval. In 2016, members of the US public were asked for to provide their views about these features; findings suggest that public preference is greatest for geologic mine-like repositories (Figure 4-7).

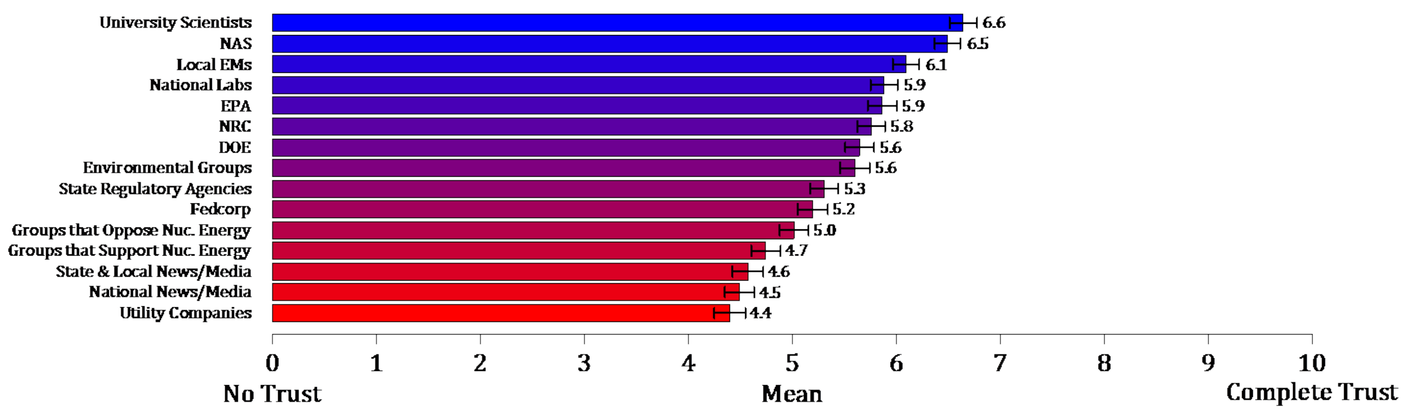


Note: The surface storage option did not indicate if the SNF would be stored on-site or at a consolidated location.

(Source: Jenkins-Smith et al. 2016)

Figure 4-7. Public Support for Storage/Disposal Depth Options

Many actors are involved in the formulation and implementation of SNF management policies, including regulatory agencies, agencies in charge of operating facilities, advocacy groups, scientists, researchers, and media outlets. Trust (or lack thereof) in these players and the information they provide can play a significant role in how members of the public think of nuclear energy and facility siting. When indicating their trust in the information provided by different agencies involved in managing radioactive waste in a 2016 national survey, respondents placed the most trust in university scientists who study nuclear energy and technologies (mean of 6.6) or those from the NAS (mean of 6.5 when rated between 1 for no trust and 10 for complete trust) (Figure 4-8). Personnel from state and local emergency agencies, such as police and fire departments, were also highly trusted (mean of 6.1).

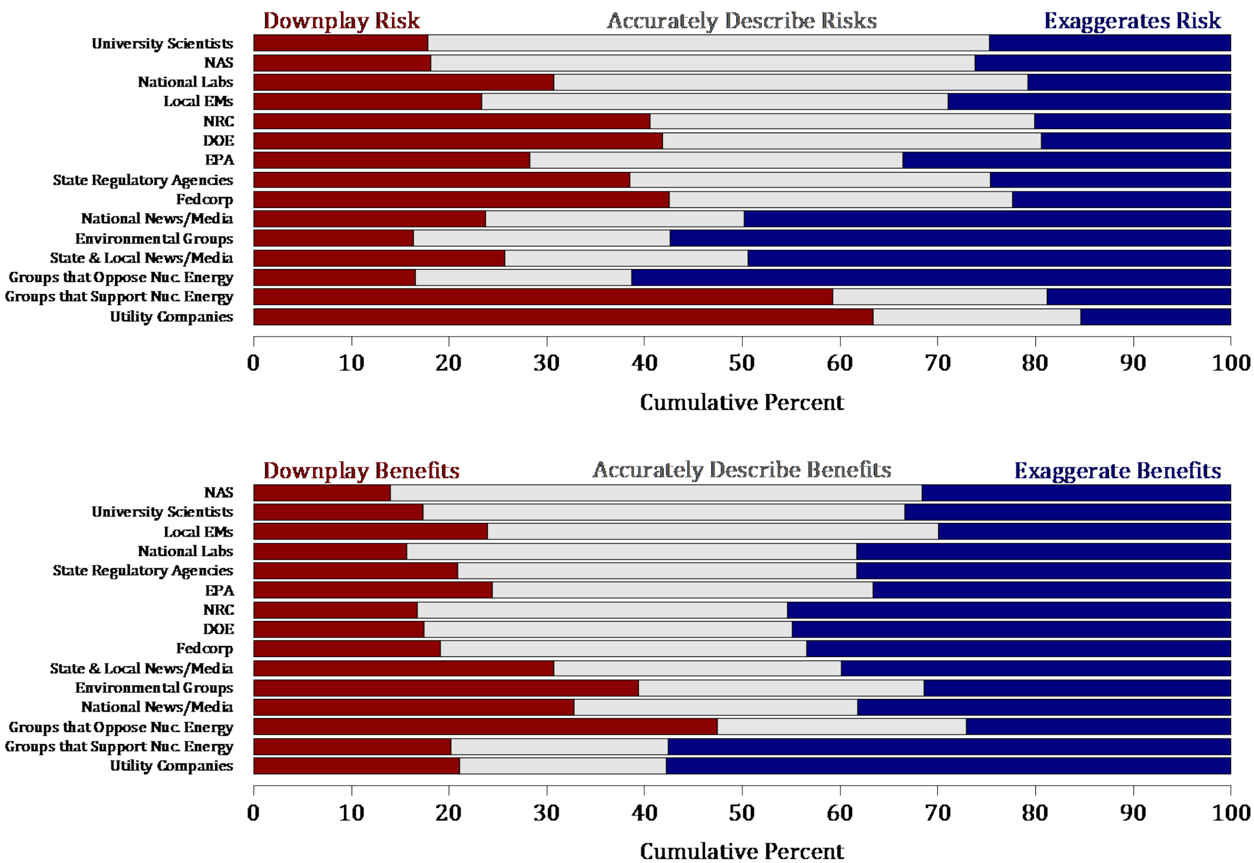


Note: Fedcorp is a hypothetical independent waste management agency.

(Source: Jenkins-Smith et al. 2017, Figure E-10)

Figure 4-8. Public Trust for Organizations Involved in Waste Management

This variation in trust stems (in part) from perceptions that these actors exhibit bias when communicating information about the risk and benefits of SNF facilities. In a 2016 national survey, respondents rated actors and institutions on a scale ranging from 1 to 7; one means they downplay risks/benefits, 7 means they exaggerate risks/benefits, and 4 means they accurately describe the risks/benefits. Findings (shown in Figure 4-9) indicate that university scientists and the NAS are seen as the least likely to either exaggerate or downplay the risks and benefits of siting a SNF facility. National labs, local emergency managers and federal regulators and government agencies (NRC, DOE, EPA) were expected to be modestly accurate in their description of risks and benefits of siting a nuclear waste facility.



Respondents rated institutions between 1 and 7 with 4 being neutral.

(Source: Jenkins-Smith et al. 2017, Figure E-11)

Figure 4-9. Perceived Bias of Various Institutions Describing Risk Associated with Managing SNF

4.5 Total System Life Cycle Costs

A total system life cycle cost (TSLCC) estimate (DOE 2008b, Section 1) for the prospective Yucca Mountain repository life cycle (design, engineering, licensing, construction, surface and subsurface operations, and decommissioning) and transportation activities was prepared in 2008 (DOE 2008b) to provide a basis for assessing the adequacy of the Nuclear Waste Fund Fee. The TSLCC estimate includes historical (sunk) costs starting from 1983 and projected costs through an assumed closure date of 2133 (DOE 2008b, Section 1). Although the project was not fully funded by Congress and was eventually suspended by DOE, the TSLCC estimate was based on the following schedule milestones from the then-current baseline schedule (DOE 2008b, Sections 1.3 and 2.2):

- Repository Construction Authorization by the NRC in 2011
- Utilities stop loading CSNF into DPCs and start loading it into TAD canisters in 2011
- Initial waste receipt and start of repository surface and subsurface operations in 2017
- End of 57-year period of waste emplacement in 2073
- End of 50-year period of monitoring with drift ventilation in 2123
- End of 10-year period of closure operations in 2133.

The TSLCC estimate was based on the acceptance, transport, and permanent disposal in the Yucca Mountain repository, using the project-specific TAD-canister-based system design (see Section 3.3.2 and A.2.5.2), of a projected (in 2008) inventory of 109,300 MTHM CSNF and 12,100 MTHM of DSNF and DHLW (DOE 2008b, Executive Summary). The TSLCC assumption was appropriate as an estimate of the cost for disposal of the full projected US inventory of SNF and HLW because (1) no basis for cost information for a second repository existed, and (2) then-proposed legislation was being considered to remove the 70,000 MTHM limit for the first repository (Yucca Mountain).

The TSLCC estimate for transportation and disposal of the projected US inventory of commercial and DOE-managed (defense) wastes at the Yucca Mountain repository, estimated as of May 2007, was $96.18 billion (in 2007$), consisting of $13.54 billion (2007$) in historical costs from 1983 through 2006 and $82.64 billion (2007$) in future costs from 2007 through 2133 (Table 4-1).

The TSLCC estimate also includes an allocation of the total system costs between commercial (civilian) waste and defense waste. The calculated cost share allocation was 80.4% commercial ($77.38 billion), to be paid from the Nuclear Waste Fund, and 19.6% defense ($18.80 billion), to be paid from annual Congressional funding appropriations.

These TSLCC estimates, specifically the commercial (CSNF) cost share allocation, provide the basis for cost estimates and cost comparisons for integrated spent nuclear fuel management options proposed in Section 5, which consider alternative choices about the timing and combination of storage, transportation, and disposal practices.

Table 4-1. Summary of TSLCC Estimate (Billions of 2007$)

COST ELEMENT	HISTORICAL COSTS (1983-2006)	FUTURE COSTS (2007-2133)	TOTAL COSTS (1983-2133)
Repository Costs (Development, Construction, Operations, Monitoring, Closure)	9.91	54.82	64.73
Transportation Costs (Development, Infrastructure, and Operations)	0.78	19.48	20.25
Balance of Program Costs (Development, Management, Benefits and Outreach)	2.86	8.34	11.20
TOTAL COSTS	**13.54**	**82.64**	**96.180**

Column totals may not add due to rounding

(Source: DOE 2008b, Table 1-1)

5. MANAGEMENT STRATEGIES FOR AN INTEGRATED SYSTEM

An optimal solution for integrated management of SNF would maximize the operational efficiency of nuclear power plants, minimize handling of spent fuel and associated occupational radiation doses at all steps in the process, provide for efficient transportation of spent fuel as it moves from reactors to repositories, and support a range of permanent geologic disposal options. Regulatory, legal, and stakeholder considerations must factor into decision making. Costs should be evaluated at the full system level and should be minimized while meeting all other constraints. Safety and security should not be viewed as adjustable metrics in the debate; all pertinent regulations associated with safety and security of storage, transportation, and disposal operations must be met regardless of cost. All solutions for SNF management and disposal, including the suboptimal system in place today, must be safe and secure.

Current practice for CSNF management in the US includes storage of spent fuel in both pools and dry storage cask systems at nuclear power plants. In the absence of a repository that can accept spent fuel for permanent disposal, projections indicate that the US will have ~136,000 metric tons of spent fuel in dry storage in ~10,000 cask/canister systems (mostly in large DPCs) by mid-century when the last plants in the current reactor fleet are decommissioned. These large DPCs are not designed for disposal. In addition, the DOE has not begun to take title to CSNF, and most nuclear utilities have successfully brought suit against the US government for its failure to honor contractual requirements to begin accepting waste by 1998.

Various options exist in the US for improving integration of these current management practices across the back end of the nuclear fuel cycle. Options include (Bonano et al. 2018):

- Constructing one or more repositories that can accommodate DPCs without repackaging
- Repackaging SNF in the future before disposal
- Storing SNF at surface facilities indefinitely, repackaging as needed

Each of these options is technically feasible, but none is what was originally planned. Nonetheless, improved integration can be achieved through some combination of these three options. A key factor that influences the path forward for integration of nuclear waste management practices is the identification of the timing and location for a repository. Alternatives range from a restart of Yucca Mountain licensing proceedings in the next decade to development of a new repository at a different location several decades or more from now.

Freeze et al. (2019a) developed a set of scenarios representing a range of possible combinations of alternative spent fuel management approaches over the next century. These scenarios, which are summarized in Appendix F, provide a basis for examining future spent fuel management options. The future alternative scenarios each make assumptions about the timing of repository availability and provide choices about integration between storage, transportation, and disposal that combine one or more of the three aforementioned options: direct disposal of CSNF in DPCs without repackaging, repackaging of existing CSNF and/or loading future CSNF into disposal-ready canisters, and extended dry storage of CSNF at utility sites and/or a CISF.

The future alternative scenarios are constructed around three representative dates for the first receipt of spent fuel at the repository: 2031, which corresponds to an early date for the opening of

Yucca Mountain should licensing activities resume immediately (Scenario 2)[17]; 2041, which represents an additional ten-year delay in restarting Yucca Mountain (Scenario 3); and 2117, which represents a 100-year delay in the repository program from the original 2017 date assumed in the Yucca Mountain TSLCC (Scenario 4). Variants (i.e., "one-off" sub-scenarios) within the scenarios examine decisions regarding direct disposal of CSNF in DPCs, repackaging CSNF from DPCs into TAD canisters, and impacts of having a federal CISF available in 2025.

While not necessarily comprehensive, the alternative scenarios span a representative range of spent fuel management options that provide useful information to examine (1) the technical integration needs between direct disposal of DPCs, repackaging/loading CSNF into disposal-ready canisters, and extended dry storage including the possibility of a CISF, and (2) the comparative costs associated with each of the options.

Section 5.1 presents the technical integration needs for a repository to be available by mid-century (i.e., Scenarios 2 and 3) and Section 5.2 presents the additional needs if a repository is not available until the next century (i.e., Scenario 4). Section 5.3 presents the comparative costs for all scenarios and variants. Section 5.4 summarizes regulatory, legal, and stakeholder considerations.

The technical integration needs include relevant ongoing and identified new R&D that is outlined in storage and transportation R&D gap reports (Hanson and Alsaed 2019; Teague et al. 2019), a disposal R&D roadmap (Sevougian et al. 2019c), and five-year plans for storage and transportation R&D (Saltzstein et al. 2020) and for disposal R&D (Sassani et al. 2020). The integration of these ongoing and identified future activities, which are summarized in Appendix C, and continued interaction with industry and the NRC, will help ensure that CSNF can continue to be stored safely and is ready for transportation (to a CISF and/or a repository) and disposal when a repository becomes available.

5.1 Path to a Repository by Mid-Century

The basis for an integrated waste management system culminating in a repository by mid-century is Scenario 2 (Table F-3). This scenario assumes disposal of the TSLCC-based inventory (109,300 MTHM of CSNF) at Yucca Mountain starting in 2031, an optimistic early date for the repository to open following a restart of the licensing process. This effectively delays the TSLCC-based Reference Case (Scenario 1) timeline (Section 4.5) by 14 years, leading to the following assumed schedule milestones:

- Repository Construction Authorization by the NRC in 2025
- Utilities stop loading CSNF into DPCs and start loading it into TAD canisters in 2025[18], which is an optimistic early date that TAD canisters could be available.
- Initial waste receipt and start of repository surface and subsurface operations in 2031
- End of 57-year period of waste emplacement in 2087
- End of 50-year period of monitoring with drift ventilation in 2137
- End of closure operations in 2147

[17] The 2031 opening date for Yucca Mountain corresponds to a restart of licensing activities in 2018-19.
[18] The 2025 date for TAD canister availability corresponds to a decision to start TAD development in 2018-19.

Insights into a mid-century repository are also gained from Scenario 3 from Freeze et al. (2019a, Section 3), which assumes disposal of CSNF at Yucca Mountain starting in 2041; all schedule milestones are delayed a further 10 years as compared to Scenario 2. Variant scenarios 2B and 2C examine options for direct disposal of some or all DPCs without repackaging. Variant scenarios 2D and 3A examine options for a federal CISF that opens in 2025. All of the mid-century repository scenarios are summarized in Table F-3.

Ongoing and needed R&D, special considerations, and necessary integration of activities that would support the opening of a repository by mid-century are discussed as follows: extended storage and subsequent transportation (Section 5.1.1), direct disposal of CSNF in DPCs (Section 5.1.2), and repackaging of CSNF for disposal (Section 5.1.3).

5.1.1 Extended Storage and Subsequent Transportation

Under any of the future spent fuel management scenarios, at-reactor dry storage of CSNF will occur for longer than was initially intended. For example, even if a repository becomes available as early as 2031, some CSNF will have been in dry storage since prior to 2000, including some stranded CSNF at shutdown reactor sites (Section 3.1.3). By 2031, there is projected to be a total inventory of ~109,000 MTHM of CSNF including 60,000 to 75,000 MTHM in dry storage at reactor sites (Figure 2-3). With an assumed waste receipt rate at the repository of 3,000 MTHM/yr (Section 3.3.2), it would take 20 to 25 years just to clear the backlog of CSNF in dry storage. So regardless of repository timing, there will be CSNF subjected to dry storage for 40 years or more before transportation, with additional storage time possible at the repository during the multi-decade waste emplacement period. If a repository opening date is even further delayed, storage durations increase commensurately.

As noted in Section 4.1, NRC findings for “Continued Storage of Spent Nuclear Fuel” (NRC 2014a) suggest that safe storage is possible for times exceeding 100 years, based on an assumption that spent fuel canisters and casks and ISFSI facilities would be replaced approximately once every 100 years. The NRC also notes that high-burnup fuel may be subject to increased degradation and that ongoing research into the extended storage of spent fuel is part of the effort to continuously evaluate and update safety regulations (NRC 2014a, Appendix B.3.2.1). Newer cladding materials such as ZIRLO^{TM} and M5^{TM} were developed to help reduce high-burnup effects (Hanson and Alsaed 2019, Section 1.2.6).

As noted in Section 3.2, the standards, regulations, and safety record for past shipments of SNF, HLW, and other nuclear materials are proven, but the transportation of the projected ~136,000 MTHM of CSNF, including high-burnup fuel and after extended storage, will require a large-scale shipping campaign and infrastructure spread over several decades. For example, the mid-century repository scenarios outlined in Section 5.1 all have an emplacement period of 57 years.

Therefore, an integrated strategy should consider storage for time periods ranging from 40 to 100 years with concurrent and/or subsequent transportation, depending on the repository timeline. The strategy outlined here is based upon long-term integrity of spent fuel and cladding, including thermally hotter and more radioactive high-burnup fuel (Section 5.1.1.1), long-term integrity of canisters, including the potential for chloride-induced SCC (Section 5.1.1.2), and transportability after extended storage (Section 5.1.1.3). Supporting this integrated strategy are considerations of transportation system planning and design (Section 5.1.1.4), security (5.1.1.5), and the possible development of CISFs as augmentation and/or alternatives to at-reactors ISFSIs (Section 5.1.1.6). Together, these strategy elements promote system integration by enabling subsequent disposal in DPCs and/or disposal-ready containers (Section 5.1.2).

5.1.1.1 Spent Fuel and Cladding Integrity

Nearly all degradation mechanisms for storage and transportation system components are dependent on temperature. Peak cladding temperatures (PCT) are limited to 400°C during storage to minimize the potential for degradation of the cladding; high temperature can increase rod internal pressure which in turn can increase the pressure-induced hoop stress, which has a design limit of 90 MPa (NRC 2010, Sections 4.4.2 and 8.4.17).

Thermal modeling and experiments and mechanical testing conducted over the last several years have contributed to a better understanding of thermal and stress profiles, including from high-burnup fuel, and their potential effects on cladding and canister integrity during extended storage and subsequent transportability. Specific R&D activities focused on the long-term integrity of spent fuel and cladding are described in Section C.1. They include:

- **Electric Power Research Institute (EPRI)/DOE High Burnup Dry Storage Research Project –** A bolted lid TN-32B Research Project Cask, instrumented with thermocouples to measure temperature distributions within the cask, was loaded with high-burnup PWR spent fuel with different types of cladding (Zircaloy-4, ZIRLO™, M5™) at the Dominion North Anna Power Station in November 2017.
  - **Research Project Cask** – Provides temperature data from high-burnup PWR spent fuel in an actual storage cask.
  - **Sibling Pin Testing** – Provides mechanical properties of cladding from fuel rods with similar high-burnup irradiation histories as those in the Research Project Cask.
- **BWR Dry Cask Simulator (DCS)** – Provides temperature and air mass flow data associated with passive cooling of a simulated DCSS. Testing at SNL has included representations of vertical (above ground and below ground) and horizontal storage systems.
- **Thermal-Hydraulic Modeling and Analyses** – Provides model validation and temperature estimates for comparison with experimentally measured temperature data. Specific activities at PNNL and SNL include:
  - Thermal Modeling of the High Burnup Research Project Cask
  - Thermal-Hydraulic Model Validation with DCS Data
  - EPRI Thermal Performance Phenomena Identification and Ranking Tables (PIRTs)

Industry typically employs very conservative or bounding assumptions in licensing calculations that result in significantly higher estimated DCSS heat loads and cladding temperatures than are actually experienced. Results from the thermal modeling and experiments to date, focused on obtaining best estimates of actual temperature profiles, have borne this out – model estimated temperatures are found to be consistently higher than measured temperatures, such as from the Research Project Cask (Section C.1.1).

Results from recent mechanical experiments, including the sibling pin testing, have determined that potential degradation mechanisms – such as hydride reorientation, delayed hydride cracking (DHC), creep, annealing, and/or oxidation – would not lead to significant cladding degradation under temperatures and hoop stresses anticipated during extended storage, including from high-burnup fuel (Section C.1.5).

These findings collectively provide the basis for NRC licensing guidance for evaluating the integrity of the spent fuel (including thermally hotter and more radioactive high-burnup fuel), cladding, and canisters during extended storage (NRC 2014a, NRC 2019b, NRC 2020e, NRC 2020f).

While the recent thermal and mechanical testing results and analyses suggest that significant cladding degradation is not expected during extended storage, the following activities will further enhance the understanding of thermal and stress environments of high-burnup spent fuel subjected to extended storage (Sections C.1.1, C.1.3, and C.1.5):

- Continuation of the ongoing EPRI/DOE High Burnup Dry Storage Research Project will provide additional data on temperature profiles within canisters in dry storage that includes high-burnup PWR fuel that will support further characterization of spent fuel, cladding, and canister degradation during the decades of extended storage now expected.
- Scaling up the DCS testing to larger systems (i.e., more than one assembly) could provide additional thermal data and insights.
- Continuation of the thermal-hydraulic modeling efforts will improve the ability to simulate decay heat transfer within canisters and casks to more accurately predict PCTs and temperature profiles without excess conservatism.
- The ongoing Sibling Pin Testing will provide further data on cladding integrity to inform models of high-burnup fuel behavior during both extended storage and transportation. This includes measurement of cladding properties after 10 years of dry storage in the Research Project Cask and comparison with the pre-storage sibling pin cladding properties.
- Examination of mechanical properties for (1) BWR cladding, which is assumed to be bounded by PWR cladding since PWR cladding is at much higher internal pressure, (2) fuel rods with higher internal pressure, such as Integral Fuel Burnable Absorber (IFBA) rods, which may have higher hoop stresses, and (3) newer cladding alloys and designs, such as are planned for ATF, especially designs meant to go to significantly higher burnups.
- Quantification of the types of stresses from external loads during loading/handling and extended dry storage (e.g., postulated drop accidents or design basis seismic events) and transportation, and their potential effects on cladding and canister integrity. This could include drop tests and/or shaker table tests.

Thermal management and mechanical integrity of CSNF also provides a basis for integration across operational boundaries (i.e., storage, transportation, and disposal). For example, PCT limits for storage impacts the timing for loading CSNF from pools to dry storage, and thermal aging requirements for transportation and/or disposal could extend storage times. Accurate evaluation of actual cladding temperatures and thermal profiles in storage and transportation casks can improve integration and efficiency throughout the spent fuel management system.

5.1.1.2 Canister Integrity

Research conducted over the last several years, such as the thermal modeling and experiments and mechanical testing described in Section 5.1.1.1, has contributed to a better understanding of canister integrity during extended storage. An additional focus is on chloride-induced SCC; the NRC (2019b, Section 3.2.2.5) has identified the potential for SCC of austenitic stainless-steel (e.g., Types 304 and 316), in particular around the canister welds, as credible during extended storage. SCC can occur during dry storage where dust accumulates on canister surfaces, and as the CSNF cools, salts within that dust deliquesce to form concentrated brines. If the salts contain aggressive species such as chloride, then the resulting brine can cause localized corrosion, and if sufficient tensile stresses are present in the metal (e.g., on welds), the potential for SCC exists (Schaller et al. 2020). It has been postulated that, over time, SCC could lead to cracks that penetrate the canister wall.

Specific R&D activities are focused on the potential impact of SCC on canister integrity and on inspection and mitigation technologies that could extend canister lifetimes, if the need arises. These activities, described in Section C.1.2, include:

- **Analysis of Canister Surface Environments** – Visual inspection and collection of dust deposition samples from ISFSIs in different geographic areas of the US to establish a more complete understanding of the potential chemical environment formed on the canister.
- **Corrosion Testing and Modeling** – Provides understanding of SCC processes in canister relevant environments. Specific activities at SNL and PNNL include:
  - Large-Scale Pitting Tests – Evaluation of chemical environments, pit growth, and crack initiation
  - Load Frame Tests – Evaluation of crack growth rates
  - SCC Model – Representation of (1) brine evolution, both before and after initiation of corrosion; (2) corrosion processes, and (3) SCC crack initiation and growth.
- **Consequence Assessment of a Canister Through-Wall Crack** – Experimental testing, coupled with modeling and analysis, to estimate the radiological consequences (from gaseous and particulate release) of a potential breach of confinement (i.e., a through-wall crack).
- **Canister Monitoring** – Development of robotic and sensor technologies capable of detecting SCCs on welded canisters in the tight space between the storage overpacks and canisters.
- **SCC Repair and/or Mitigation** – Evaluation of options to repair and/or mitigate the possibility or impact of SCCs on the canister surface under dry storage conditions, such as coatings or cold spray techniques.

The risk of corrosion and SCC is greatest in near-marine settings, where chloride-rich sea-salt aerosols are deposited on the canister surface (Schaller et al. 2020); however, chlorides may also be present in the atmosphere near cooling towers, salted roads, or other locations (NRC 2019b, Section 3.2.2.5). Austenitic stainless steels are more resistant to SCC in sodium chloride (NaCl) solutions but crack readily in magnesium chloride ($MgCl_2$) solutions (NRC 2019b, Section 3.2.2.5).

The canister inspections and sampling to date did not show any indication of SCC. Dust deposition samples and brine stability experiments suggest the possibility of (1) low chloride salt loads and (2) limited conditions for formation and stability of deliquescent magnesium chloride brines (Section C.1.2). The following activities will further enhance the understanding of long-term canister integrity, including the potential impact of SCC and thermal and chemical environments during extended storage (Sections C.1.2 and C.1.4):

- Collection of additional in-service dust deposition samples will provide a better understanding of the diversity of chemical environments in different geographic areas of the country.
- Continued evaluation of the stability of magnesium chloride brines in relevant thermal-chemical environments.
- Continued testing to acquire data for pitting, SCC initiation and growth rates, and through-wall crack consequence.
- SCC model validation against newly acquired data for canister-relevant conditions, which will reduce uncertainty and improve the capability to estimate the timing and occurrence of SCC.
- Continued evaluation of canister inspection, repair, and mitigation techniques.
- Development and implementation of a full-scale canister deposition field demonstration. A full-scale demonstration at various heat loads would provide controlled data on dust deposition and brine stability and serve as a platform for inspection and eventual repair and mitigation techniques.
- Implementing an intermediate-scale spent fuel drying experiment to explore water retention, or the lack thereof, during normal drying procedures. Many degradation mechanisms are dependent on or accelerated by the presence of water (e.g., fuel oxidation, hydrogen buildup). While there is no direct evidence that the amount of water that remains in a cask/canister after a normal drying process is of concern, ongoing R&D will help to confirm estimates of residual water in canisters.

The cross-cutting nature of the thermal profiles (Section 5.1.1.1) is also important; accurate estimation of canister thermal environments is critical to the understanding of canister SCC and canister drying processes. Continuation of these testing and analysis activities will improve the capability and accuracy in evaluating SCC initiation and growth rates as a function of environmental parameters (salt load, temperature, humidity, and salt/brine composition, overpack design and airflow), material properties, and stress state.

5.1.1.3 Transportability After Extended Storage

Transportability of CSNF after extended storage is dependent on the confidence in cladding and canister integrity demonstrated by the thermal and mechanical experiments and analyses summarized in Sections 5.1.1 and 5.1.1.2.

Testing and modeling conducted over the last several years have contributed to a better understanding of the types of external loads (forces, strains, accelerations, etc.), stresses (magnitude, frequency, duration, etc.), and their potential effects on cladding and canister integrity during extended storage and transportation. These activities have primarily focused on stresses encountered under transportation conditions and are described in Section C.2. Specific R&D activities include (Section C.2.2):

- **Multi-Modal Transportation Test** – An ENUN 32P cask was loaded with three surrogate assemblies; the assemblies, basket, cask, cradle, and conveyance systems were all instrumented with strain gauges and accelerometers. Data were collected during the following NCT segments: handling, heavy-haul truck, coastal vessel, ocean-going ship, open rail, and captive-rail (Section 3.2.1).
- **MMTT Data Analysis** – Provides peak strains and accelerations during the MMTT for each mode of transportation.
- **MMTT Modeling** – Provides fatigue analysis using MMTT and other "real" system data.
- **Drop Tests** – Provides acceleration data from instrumented surrogate casks during 30-cm drop tests, which are part of NCT analyses.

Results from these recent tests and analyses has shown that operational loads on cladding under NCT lead to induced stresses that are well below yield levels. These findings collectively provide the basis for NRC licensing guidance for evaluating the integrity of spent fuel (including thermally hotter and more radioactive high-burnup fuel) during transportation (NRC 2014b, NRC 2020e, NRC 2020h). The following activities will further enhance the understanding of thermal and stress environments of high-burnup spent fuel subjected to transportation after extended storage (Sections C.2.1, C.2.2, and C.2.3):

- Continued testing and analysis of external loads that the canister and system components (beyond just the cladding) will experience during their lifetime up to final disposal. This includes:
    - Ongoing Sibling Pin Testing (Section C.1.5) to provide further data on cladding integrity to inform models of high-burnup fuel behavior under both NCT and HAC.
    - Further quantification of the types of stresses on the canister and system components (e.g., assembly hardware loads and rod-to-rod impacts) from external loads (e.g., postulated drop accidents during NCT or HAC).
- Continued development of a cumulative effects model of all external loads over the full life cycle of a CSNF system, including loading/handling, extended dry storage (including design basis seismic events), and NCT. Considered together, there is the possibility that multiple individual low-magnitude events could cause enough incremental damage to the components of a CSNF storage or transportation system to challenge its structural integrity if the time span is long enough or if the life cycle includes a sufficient number of loading events. The cumulative effects model will be used to perform a systems analysis to evaluate the integrity of the cladding, canister, and other components in preparation for transportation following extended storage.
- Development of a thermal model that estimates PCT and temperature profiles for a canister from the beginning of storage, through transportation, to disposal. This "canister thermal lifetime" model could evaluate temperature histories for a range of DPC types, loadings, and burn-ups to inform canister and cask designs, transportation timing, and disposal aging/cooling and emplacement spacing, and would significantly enhance integration across the back end of the fuel cycle.

Together, these data and analyses will inform transportation system planning and design. The 30-cm drop tests (Section C.2.2) complete the regulatory loading requirements associated with NCT. The cumulative effects model and the thermal lifetime model will integrate the mechanical integrity and thermal management of CSNF the across the operational boundaries, from storage to transportation to disposal. For example, the surface temperature of the transportation cask must not exceed 50°C (10 CFR Part 71.43). Therefore, transportation of large DPCs, especially with high-burnup fuel, may not be feasible for decades to come due to the high heat loads generated and/or external dose limits (see Figure 4-1), which could extend storage times. Similarly, canister thermal loads impact disposability following transportation (e.g., geologic-media and backfill-specific thermal constraints, waste package and drift spacing, aging, etc.) (see Figure 4-2).

5.1.1.4 Transportation System Planning and Design

Transportation of the projected ~136,000 MTHM of CSNF, including high-burnup fuel and after extended storage, will require a large-scale shipping campaign and infrastructure spread over several decades. At a nominal annual waste acceptance rate at the repository of 3,000 MTHM (Section 3.3.2), it would take ~45 years to transport the entire projected CSNF inventory. The mid-century repository scenarios outlined in Section 5.1 all have an emplacement period of 57 years. The presence of a CISF would potentially double the number of shipments. Advance planning timeframes on the order of a decade could be required to plan and coordinate a transportation strategy and to establish the institutional and physical infrastructure to conduct a large-scale shipping operation. Specific activities that would support transportation system planning and design include (Sections C.2.4 and C.2.5):

- Identification and planning of rail lines, system support, maintenance, and operations.
- Early implementation and testing of institutional arrangements involving state, tribal and local officials. The NWPAA (1987, Section 180(c)) mandates technical assistance and funding from the Nuclear Waste Fund for training local government and Indian tribes to cover procedures required for safe routine transportation of SNF and HLW, as well as procedures for dealing with emergency response situations.
- Design, fabrication, testing, licensing, and acquisition of the necessary transportation fleet, which includes:
  - Rail rolling stock (cask cars, buffer cars and escort cars), and heavy-haul trucks and barges at the sites without direct rail access. Estimates of rolling stock needed for different DPC transportation scenarios are summarized in Table E-1. A larger fleet would be required if smaller standardized canisters are implemented sometime in the future.
  - Certified transportation casks, which currently exist for about 85% of the current CSNF inventory in dry storage (Section A.2.4), are needed for all CSNF. The existence of storage-only canisters (typically older) complicates the task of eventually transporting the spent fuel in them to a CISF site or to a disposal facility.
  - Continued development of 8-axle and 12-axle rail cask cars.
- Transportation system logistics modeling (Section 3.2.3 and Appendix E). Important logistics considerations are:
  - Identification of transportation routes (e.g., from reactor sites and shutdown sites to a repository, and possibly also to a CISF).

- Development of a transportation schedule.
- Availability of the equipment (transportation fleet) required to move the fuel in accordance with the selected schedule.
- Possibly prioritizing shipments of stranded fuel from shutdown reactor sites, which may be complicated by lack of rail access (Sections 3.1.3 and 3.2.2).
- Regulatory and technical challenges associated with "72-71-72" (storage to transportation to storage) or "72-71-63" (storage to transportation to disposal) sequencing for system component inspections (e.g., canister, fuel, cladding integrity) (Sections 3.2 and 4.4.1).
- Upstream (storage) and downstream (disposal) impacts related to thermal management, criticality, inspections (see previous bullet), and other factors.
- Socio-economic, socio-political, and safety factors.

DOE-NE IWM has ongoing tasks supporting several of these activities.

5.1.1.5 Security

Security concerns during extended storage and transportation have become more pronounced since the terrorist attacks of September 11, 2001. ISFSI licensees must implement a layered defensive security that includes on-site protective forces with appropriate skills, weaponry, and other response equipment, and security systems to defend against physical and cyber attacks and insider threats (Section (C.1.6). NRC has engaged in the process of proposed a rulemaking that would revise the existing security requirements for the storage of SNF at an ISFSI or a CISF (Section C.1.6) and has sponsored studies to assess the vulnerability of transportation packages to certain types of terrorist attacks (Section C.1.7). There are ongoing DOE-NE SFWST R&D activities to identify the needs for site-specific physical security and monitoring during extended storage and during transportation.

5.1.1.6 Consolidated Interim Storage

The development of a CISF could provide flexibility and integration between storage, transportation, and disposal; it could expedite acceptance of CSNF from shutdown reactor sites and a federal CISF could expedite DOE taking title of CSNF to reduce long-term taxpayer liabilities (DOE 2013). DOE-NE IWM has performed a preliminary examination of the design, planning, operational logistics, and costs of a DOE owned and operated CISF (Jarrell et al. 2016; Cumberland et al. 2016). Congressional appropriations for FY21 included funding for federal interim storage.

However, federal CISF sites remain to be licensed and their viability is uncertain without modifications to the portions of the NWPA that link federal interim storage to repository licensing and operation, and private efforts for developing regional interim storage facilities have been stymied by national and state political opposition (Section 3.1.4).

5.1.2 Direct Disposal of CSNF in DPCs

Under any future spent fuel management scenario, the permanent disposal of the projected ~136,000 MTHM of CSNF in ~10,000 DPCs into one or more repositories will be necessary. The basis for repository planning and design assumes the long-term integrity of spent fuel, cladding, and canisters, including high-burnup fuel, during extended storage and subsequent transportation (Section 5.1.1). With the suspension of the Yucca Mountain licensing process, DOE-NE SFWST disposal research is currently focused on generic repositories in several different geologic media (Mariner et al. 2019), including bedded salt, argillaceous material (e.g., clay and shale), crystalline rock, and unsaturated alluvium (Section C.3). A key focus is evaluating the technical feasibility of disposing of CSNF in DPCs in a repository, as an alternative to repackaging the CSNF into purpose-designed or standardized disposal containers. Reference cases describing these generic repositories have been developed to provide a platform for integrating concepts, including conceptual models and simulations that evaluate for impacts associated with direct disposal of DPCs.

As compared to repackaging the CSNF for disposal, the direct disposal of CSNF in DPCs has the potential to simplify disposal operations, minimize the number of transportation shipments, reduce occupational worker doses, and decrease the overall costs associated with geologic disposal. A repository that can accommodate direct disposal of CSNF in DPCs promotes integration across the back end of the nuclear fuel cycle.

DPCs tend to be large, heavy, and have a high thermal and radiological output; the possibility of direct disposal of CSNF in DPCs includes options with or without modification to the DPCs. Studies of technical feasibility for direct disposal of CSNF in DPC-based waste packages have focused on four aspects (Hardin et al. 2015; Liljenfeldt et al. 2017): operational and postclosure radiological safety (Section 5.1.2.1), engineering feasibility (e.g., handling and emplacement) (Section 5.1.2.2), thermal management (Section 5.1.2.3), and postclosure criticality control (Section 5.1.2.4). As discussed below, challenges associated with the first three aspects can be accomplished using currently available technologies and modeling approaches, and R&D investigating approaches for postclosure criticality control is ongoing (SNL 2020; SNL 2021a).

Longer-term plans are focused on continued R&D on direct disposal of CSNF in DPCs with and/or without modification, with consideration of postclosure criticality control, effects of potential modifications on the thermal-chemical evolution of the repository system, and design of DPC disposal overpacks that could enhance postclosure performance (Section C.3).

5.1.2.1 Operational and Postclosure Radiological Safety

Preliminary postclosure performance assessments (PAs) showed that regulatory performance objectives on individual protection and groundwater protection could be met for generic reference disposal concepts in salt, argillite, and crystalline host media (Freeze et al. 2013; Hardin et al. 2015). These preliminary PAs included CSNF in disposal-ready TAD canisters. These results are generally applicable to direct disposal of CSNF in DPCs, because of the similar size and thermal load in DPCs and TAD canisters. In either case, a site-specific disposal overpack would be used (Hardin et al. 2015, Section 2.2.1). Differences between assessments for DPCs and TAD canisters could arise because of the quantity of waste in each package, the duration of elevated temperature, and the internal design of the canisters. Such differences are difficult to represent in PA models, and conservative simplifications are often used (e.g., minimal or no containment credit for waste forms and waste packages of either type).

More recently, reference cases that include CSNF in DPCs in generic repositories in argillite (Mariner et al. 2017; Sevougian et al. 2019a; Sevougian et al. 2019b), salt (Sevougian et al. 2019a), crystalline (Mariner et al. 2016), and unsaturated host rocks (Mariner et al. 2018; Sevougian et al. 2019a; Sevougian et al. 2019b) have been developed to further evaluate the feasibility of direct disposal of CSNF in DPCs. Preliminary results from PA models of these reference cases, supplemented by PA model results from other countries (e.g., Andra 2005; SKB 2011; Posiva 2012; NWMO 2013), suggest that repository designs capable of sufficient postclosure isolation are feasible in all geologic media, although thermal and criticality constraints differ between geologic media and designs. Site-specific repository designs and PA models would be necessary if a location other than Yucca Mountain were pursued.

Preclosure operational safety considerations for direct disposal of CSNF in DPCs, summarized in SNL (2021, Section 3), rely heavily on the current state of industrial practice and on the Yucca Mountain preclosure safety analysis (DOE 2009a, Section 1). While not quantified, the direct disposal of DPCs without repackaging would avoid the potential for additional occupational radiation doses to workers from repackaging operations.

5.1.2.2 Engineering Feasibility

Engineering feasibility of direct disposal in DPCs has been investigated (Hardin et al. 2015, Section 2.2; SNL 2021a, Sections 2 and 3). Waste disposal in DPC-based packages would most closely resemble the TAD canister packaging concept proposed for Yucca Mountain. DPCs fully loaded with fuel are somewhat larger and heavier than TAD canisters would be, so disposal overpacks would be heavier as well. The overall size and weight increase is on the order of 10 to 20%, depending on which of the many DPC designs is considered (current DPCs may hold up to 32 or 37 PWR spent fuel assemblies or up to 68 or 89 BWR spent fuel assemblies, as compared to the 21-PWR / 44-BWR TAD canisters). The near equivalence in weight is partly attributable to a more robust TAD fuel basket for the absorber plate configuration, and to the circular geometry.

Specific aspects and considerations of repository operations for DPC disposal – design and layout, surface facilities, and subsurface facilities – are summarized in SNL (2021, Section 2.1). Handling of DPCs is within the state of safe industry practice; it occurs regularly at ISFSI sites. Modification of the Yucca Mountain disposal concept, which uses ramps for moving waste packages underground, to accommodate DPC-based waste packages rather than TAD-based waste packages would be relatively straightforward. It could be accomplished by redesign or up-rating of the transporter designed for Yucca Mountain, or by other methods commonly used for moving heavy loads in factories, shipyards, etc. (Hardin et al. 2015).

Modifications to existing repository design concepts other than Yucca Mountain to accommodate the size and mass of DPCs could be more extensive but they appear to be achievable using current engineering technology (Hardin et al. 2015). A number of these concepts are described in SNL (2021, Section 2.3). For example, if waste packages are transported underground via a shaft (e.g., for a repository in salt) there are modern hoists available of the friction-winder type, with payload capacity of 175 MT or greater which is enough for a large waste package, personnel shielding, and an undercarriage. Placing waste packages in final disposal position would be done using the in-drift mode of emplacement, using a deposition machine such as that proposed for Yucca Mountain, or other concepts (SNL 2021a). If ramps are used to transport waste packages underground, the transporter designed for Yucca Mountain, with some modifications could be used for shielded transport and emplacement.

An additional consideration is the need for long-lived ground support to keep emplacement drifts and access tunnels open for up to 100 years of operations (SNL 2021a, Section 2.4). Cementitious materials (e.g., shotcrete) are effective for long-term ground support but their effects on in-drift chemistry must be evaluated.

5.1.2.3 Thermal Management

Extensive generic studies have been performed of thermal management for direct disposal of CSNF in DPC-based packages, for repositories in different host media (Hardin et al. 2015; SNL 2021a, Section 2.3). Disposal concepts typically fall into two groups for thermal management: those that can tolerate a maximum temperature of 200°C or greater in the near field, and those for which the limit is much less, on the order of 100°C. Concepts for salt and unbackfilled hard rock fall into the former (hotter) group, while all concepts involving clay-based buffer/backfill around waste packages, or clay/shale host rock, fall into the latter (cooler) group. The 100°C limit is a conservative approach to limited buffer/backfill degradation (Section 4.3.2). These geologic-media and buffer/backfill-specific thermal constraints in turn affect the acceptable waste package power/temperature at emplacement.

Figure 4-2 provides representative estimates of aging durations necessary to meet the thermal constraints for different geologic media. As noted in Section 4.3.2, these temperature estimates and aging durations are strongly dependent on the repository layout (i.e., spacing between waste packages within an emplacement drift and spacing between drifts), fuel burnup and waste package loading, and assumptions about the rate and duration of repository ventilation after emplacement and before drifts are sealed. Required aging times can be significantly reduced by increasing spacing between waste packages and/or ventilation after emplacement. With those caveats, aging times for representative DPCs in salt or unbackfilled hard rock repositories are less than 100 years, even for high-burnup fuel, whereas aging times in backfilled argillite or crystalline repositories may be hundreds of years.

A canister thermal lifetime model that estimates thermal power and temperature (PCT and surface) throughout the life cycle of a DPC, outlined in Sections 5.1.1.3 and C.2.1, would integrate thermal management of CSNF across the operational boundaries, from storage to transportation to disposal. It would provide upstream information to interface with disposal requirements for aging, spacing, etc.

5.1.2.4 Postclosure Criticality Control

As noted in Section 4.3.2, direct disposal of CSNF in DPCs may pose complications for licensing analyses of postclosure criticality because aluminum-based materials used in existing designs for neutron absorption (see Section A.2.3) during storage and transportation are expected to degrade in tens to hundreds of years when exposed to groundwater in a repository (SNL 2020; SNL 2021a).

The potential for postclosure criticality is negligible as long as waste packages remain intact and water does not reach the spent fuel assemblies. However, once a waste package breaches during the postclosure period and floods with groundwater, the fuel assembly and basket will begin to degrade. The fuel assembly and stainless-steel basket materials can retain a substantially intact structural configuration for many thousands of years, but the much faster degradation of the aluminum-based neutron absorbers will increase reactivity and give rise to the possibility of criticality. Over the very long term after waste package breach (e.g., significantly longer than 10,000 years of exposure to ground water) the fuel and basket components will lose structural integrity, allowing the fuel array to collapse and consolidate. The eventual collapse is likely to consolidate the fuel and basket to a point where criticality is no longer possible even without neutron absorbers (SNL 2021a, Section 1.2.1).

As a result, current research on direct disposal of DPCs is focused on postclosure criticality control. Specific research areas include direct disposal without modification (Section 5.1.2.4.1), modification of already-loaded DPCs (Section 5.1.2.4.2), and modification of DPCs to be loaded in the future (Section 5.1.2.4.3).

R&D investigating the fuel and basket structural evolution and longevity will also benefit R&D investigating the long-term evolution of the repository system. For example, the postclosure PA reference cases (See Section 5.1.2.1) make a bounding assumption (for the purposes of radionuclide release) that the fuel cladding degrades instantaneously and does not provide any performance credit. However, the criticality consequence PA models (see Section 5.1.2.4.1) make a bounding assumption (for the purposes of the occurrence of criticality) that fuel cladding stays intact. Integration of these PA modeling activities will provide consistency in the fuel cladding evolution assumptions.

5.1.2.4.1 DPC Disposal Without Modification

Options to facilitate disposal of CSNF in DPCs of current designs without modification include: (1) analysis of as-loaded reactivity margin, generally from loading of less reactive fuel than originally analyzed, but also as influenced by favorable dry and/or saline conditions; (2) analysis of criticality consequences should one or more criticality events be predicted to occur in a repository; and (3) analysis of the mitigation of criticality events with a high-performance disposal overpack that could be relied on to maintain containment for an extended period beyond 10,000 years (SNL 2021a, Section 1.1).

Clarity et al. (2020) describe as-loaded criticality analyses of more than 500 existing DPCs (both BWR and PWR) for stylized degradation cases under long-term disposal conditions, which included reactivity margin from actual DPC loading details, burnup credit, and radionuclide decay. The analyses also considered the effects of fresh water versus saline water because chlorine in groundwater can provide reactivity reduction. The detailed canister-specific analyses showed that subcriticality can be demonstrated for more than 50% of the DPCs in fresh water and for 100% of the DPCs in saline (>32,500 mg/L Cl) groundwater (Clarity et al. 2020). However, better understanding of the corrosion process of basket structural materials and their physical degradation mechanisms, as well as the probability of flooding or partial flooding, could further and significantly influence the projected likelihood of criticality events.

These results suggest that a portion of the existing fleet of DPCs could be disposed of directly without modification, with low probability of a postclosure criticality event based on reactivity margin analysis. It should be noted that the existing as-loaded DPCs that were analyzed were dominated by early DPCs design and loadings, which have the greatest built-in reactivity margin. For more recent DPCs, which generally do not have sufficient uncredited reactivity margin, it may be more difficult to demonstrate subcriticality under disposal conditions that include waste package breach and flooding with fresh water (SNL 2021a, Section 1.1.1). This is particularly true for PWR DPCs, which are generally more reactive than BWR DPCs; ongoing R&D is investigating improvements in BWR burnup credit analysis that could result in projected subcritical conditions for many BWR DPCs under degraded conditions (SNL 2021a, Section 1.1.1).

Another consideration for reactivity margin is from unsaturated and/or saline conditions. It may be possible to demonstrate long-term subcriticality for repositories in which insufficient water is available to flood breached waste packages, or where high-salinity brines are present to limit reactivity.

Consideration of criticality in a postclosure PA analysis is commonly based on a feature, event, and process (FEP) screening. For the Yucca Mountain Repository License Application, postclosure criticality was screened out based on low probability (less than 1 chance in 10,000 of occurrence within 10,000 years of disposal or ~10^{-8} per year) (10 CFR Part 63.342) due to the long-term performance of the criticality controls in the TAD canister design (DOE 2009a, Section 2.2.1.4.1). For disposal of CSNF in existing DPCs, the probability of postclosure criticality will not meet the low probability FEP screening threshold under certain conditions (i.e., no DPC modifications, fresh or low-salinity groundwater). As a result, ongoing R&D with the SNL GDSA Framework, using the open-source PFLOTRAN code for system-level modeling of coupled thermal-hydrologic-chemical processes in a high-performance computing (HPC) environment, is evaluating the potential consequences from postclosure criticality events using PA models of generic repository systems (Section C.3). The estimated consequences from postclosure criticality may be insignificant in accordance with 10 CFR Part 63.342 (i.e., a low consequence FEP screening) and/or they may be included in a full PA analysis.

The criticality consequence approach derives from a methodology developed in the 1990s (DOE 2003) and has been revisited for PAs of generic disposal concepts in saturated clay and in unsaturated alluvium (SNL 2021a, Section 1.1.2). The approach considers two different types of criticality events (DOE 2003; SNL 2021a, Section 1.1.2):

- **Steady-State Criticality Event** – Produces energy at a relatively low but fairly constant (or static) rate.
- **Transient Criticality Event** – Potentially produces energy at a high power level but generates significantly less total energy than a steady-state event because of its much shorter duration. Significant kinetic energy releases will not likely occur during transient criticality events since the events that can occur in the repository will be sufficiently slow to preclude such consequences.

Steady-state and transient criticality events, including the coupling between thermal-hydrologic and neutronics processes, have been characterized and bounded in the PA models where possible. Consequences from a criticality event include: (1) an incremental change in the radionuclide inventory over the duration of the criticality event, and (2) thermal and mechanical effects on engineered barriers that could change the release and transport of radionuclides from the waste package. Preliminary assessments of a 10,000-year long steady-state criticality event in a 37-PWR DPC indicated that, for a hypothetical repository in a saturated shale host rock, the steady-state criticality power output is ~4 kW, the maximum temperature in the waste package is about 200°C, the maximum temperature increase at an adjacent waste package is 5 to 15 C°, and the increase in inventory of radionuclides that tend to be important to repository performance is on the order of 4% (Price et al. 2020, Section 3).

For a hypothetical repository in unsaturated alluvium, the power output of a steady-state postclosure criticality event depends on the deep percolation rate of water into the repository and waste package (Price et al. 2020, Section 3). The criticality event starts when the waste package fills with water and terminates when decay-heat driven evaporation without boiling dries out the waste package. For an assumed percolation rate of 10 mm/yr, a power output of 300 W can be sustained for about 1000 years, whereas for an assumed percolation rate of 2 mm/yr, a power output of 100 W can be sustained for a few hundred years. Maximum waste package temperatures associated with these power levels are 78°C and 57°C, respectively (Price et al. 2020, Section 3). Because the changes in inventory are proportional to the power level and duration, the inventory changes in these two unsaturated alluvium cases are about 7.5% and 2.5%, respectively, of those calculated for the saturated shale case (assuming the same duration).

Additionally, specific processes that affect reactivity were analyzed to determine which ones were likely to cause permanent criticality termination (Alsaed 2020). This analysis concluded that changes in geometry due to grid spacer corrosion or failure that results in uniform pin pitch reduction by ~3 mm would lead to permanent termination of criticality for most DPCs.

The potential for criticality in DPCs during long-term disposal could be reduced with a high-performance disposal overpack that could extend the waste package containment lifetime to tens of thousands to hundreds of thousands of years. This type of disposal overpack was included in the Yucca Mountain repository design, where waste package containment performance was credited to the overpack and not the TAD canister (SNL 2021a, Section 1.1.3). A corrosion-resistant disposal overpack for DPCs could be of similar design to Yucca Mountain CSNF waste package (Section 3.3.2). Ongoing R&D is exploring candidate corrosion-resistant materials and the possibility of updating methods and models used to control manufacturing defects that can lead to early waste package failures (SNL 2021a, Section 1.1.3). The use of a robust disposal overpack to has the potential to limit and/or delay waste package breach, decrease the incidence of criticality in DPC-based waste packages, thereby reducing both the probability and consequence of criticality.

5.1.2.4.2 Modification of Already-Loaded DPCs

Ongoing R&D for modification of already-loaded DPCs to facilitate disposal is focused on injectable filler materials for criticality control (SNL 2021a, Section 1.4). Fillers would be injected into canisters as liquids that solidify, and then later exclude/displace some or all of the groundwater that could flood into a breached waste package. Without sufficient moderator (groundwater) the CSNF in a DPC would remain subcritical. Injectable fillers, if successfully demonstrated, could be implemented with existing DPCs and those to be loaded in the future. In addition to moderator exclusion, fillers could also be designed for entrainment of neutron absorbers, such as fine particulate boron carbide (B_4C). The use of injectable fillers has low technical maturity, but it is the only approach under study that could be used to modify weld-sealed DPCs already loaded with fuel without cutting the lid off (SNL 2021a, Section 1.4). Sealed canisters would be accessed by cutting off the welded covers over the access ports used for dewatering, using those access ports to inject the filler material, and re-welding the port covers after filling.

Filler studies are currently focused on cement slurries (aluminum phosphate, calcium phosphate, and wollastonite phosphate) and molten liquids (metals, alloys, and glasses) (SNL 2021a, Section 1.4). Desirable filler properties include chemical inertness, thermosetting behavior, near-neutral pH, self-bonding, low solubility in groundwater, non-toxicity, sorptive of radionuclides, mechanical strength, and thermal conductivity. Additional considerations include mass/weight added to waste package, longevity in the presence of groundwater, quality control during filling operations (to verify that DPC void space has been filled), and water-filled porosity (for cement slurries).

Cement slurry filler R&D has focused on the optimization of compositions and subsequent processing of the candidate materials to achieve dense and well-consolidated monolithic samples with 30 to 40% porosity and permeabilities of 1 millidarcy (SNL 2021b, Section 8). The aluminum phosphate cements (APCs) and the wollastonite phosphate cements (WPCs) appear to show the most promise for continued development. Less progress has been made with the calcium phosphate cements (CPCs); their slurry viscosities are high (and difficult to measure) and they exhibit relatively short cure times of 2 to 3 hours with excessive volatile generation.

Molten liquid filler R&D has focused on computational fluid dynamics simulation and experimental testing to determine whether a prototype DPC can be filled with low-temperature melting metals/alloys using the access port and drain pipe (Cetiner et al. 2020). Simulation and prototype testing will also be used to screen molten filler materials.

5.1.2.4.3 Modification of Future-Loaded DPCs

Ongoing R&D for modification of future-loaded DPCs to facilitate disposal is focused on fuel assembly modifications (disposal criticality control features, optimized zone loading) and/or basket modifications (added neutron absorbing features) (SNL 2021a, Sections 1.2 and 1.3).

Fuel assembly modifications under consideration would try to avoid changing existing DPC basket designs. Ongoing R&D includes evaluation of (SNL 2021a, Section 1.2):

- **PWR disposal control rods** – The feasibility of adding disposal criticality control rods to PWR spent fuel assemblies prior to disposal and their long-term behavior in a post-waste-package-breach disposal environment are being studied (SNL and ICG 2019; SNL and ICG 2021). Among the modifications discussed, this could be closest to being realized. Analysis of the effective neutron multiplication factor (k_{eff}) of as-loaded canisters and availability of basket locations (e.g., guide tubes) needs further development (SNL 2021a, Section 4.2).
- **BWR fuel channel replacement** – The re-channeling approach for disposal criticality control would replace the fuel channels on selected BWR assemblies with channels fabricated from corrosion-resistant advanced neutron absorber (ANA) material. The approach is potentially applicable to all BWR fuel designs, and all DPC basket designs with sufficient clearance for the required disposal channel thickness. As noted in Section 5.1.2.4.1, improvements in BWR burnup credit analysis could result in projected subcritical conditions for many BWR DPCs under degraded conditions, in which case BWR modifications for postclosure criticality control may not be necessary.
- **Zone loading to limit postclosure reactivity** – Zone loading of future DPCs for optimization of postclosure criticality control is a refinement of the reactivity margin approach described in Section 5.1.2.4.1. Previous work suggests that 10% to 15% improvement in k_{eff} might be realized, although it is not clear how many DPCs could be loaded to achieve this reduction. Implementation would require re-licensing of loading protocols so that reactivity is considered in addition to worker dose and fuel temperature (SNL 2021a, Section 4.2).

Basket modifications under consideration include the addition or replacement of built-in DPC neutron absorbing features with longer-term corrosion-resistant materials such as borated stainless steel or Ni-Cr-Mo-Gd ANA. Ongoing R&D includes evaluation of (SNL 2021a, Section 1.3):

- **PWR or BWR chevron inserts** – Chevron-shaped insert plates, similar to those used in a spent fuel pool storage racks, could be constructed with ANA materials for insertion into DPCs with baskets made of aluminum-based materials, and could also be used to retrofit (prior to loading) any DPC of an existing design that uses aluminum-based absorber plates. Design of inserts depends the clearance available in DPC basket cells.
- **PWR absorber plate replacement** - Future DPCs could feasibly be redesigned with corrosion resistant neutron absorbing plates or other basket features. For example, long-lived absorber plates could be substituted for aluminum-based absorber plates in many DPC basket designs during basket fabrication (SNL 2021a, Section 1.3.3).

Evaluation of these potential fuel and basket modifications is supported by (1) estimation of added weight from modifications in relation to spent fuel pool overhead crane hook load limits, and (2) modeling of fuel/basket degradation, which will not necessarily occur uniformly, to generate collapsed and partly collapsed configurations for neutronic analysis (SNL 2021a, Section 1).

The ultimate efficacy of fuel/basket modifications for future DPCs depends on timely implementation, and would need to be supported by the DPC vendors and by the nuclear utilities who use them. At the end of 2019, there were already an estimated 39,000 MTHM of CSNF in dry storage; by 2031 there is projected to be 60,000 to 75,000 MTHM of CSNF in dry storage, almost all in sealed DPCs (Figure 2-3).

5.1.3 Repackaging of CSNF for Disposal

Complementary to the direct disposal of CSNF in DPCs (Section 5.1.2) are options for disposal of CSNF in canisters that are specifically designed (e.g., with appropriate overpacks and long-term criticality control features) for safety in a postclosure environment. These canisters could be purpose-designed for a specific disposal site and/or have a standardized design for disposal in a range of geologic media. Depending on where in the back end of the fuel cycle these purpose-designed or standardized canisters are introduced, their design may also need to be licensable for CSNF storage and/or transportation.

The idea of designing and developing a standardized canister that could be used for all functions of the back end of the nuclear fuel cycle is not new. The federal government proposed standardized canisters for storage (aging), transportation, and disposal in 1994 (the multi-purpose canister (MPC)), in 2006 (the TAD canister for Yucca Mountain), and again starting in 2013 (a standardized transportation, aging, and disposal (STAD) canister). All of these concepts are described in more detail in Section A.2.5.

The implementation of standardized canister systems can be used to support two broad purposes: repackaging CSNF that is already in dry storage (Section 5.1.3.1); and loading CSNF that is not yet in dry storage (Section 5.1.3.2).

A related technology is rod consolidation, which could be used to reduce the total number of waste packages needed and improve postclosure criticality performance. However, interest in rod consolidation has waned since the 1990s. Considerations for rod consolidation are summarized in Section A.2.5.4.

5.1.3.1 Repackaging CSNF Already in Dry Storage

At the end of 2019, there was about 39,000 MTHM of CSNF in dry storage in the US in about 3,100 DPCs (Table 2-1), and the numbers continue to increase. For the CSNF that is already in dry storage, repackaging involves cutting open DPCs and transferring the CSNF into purpose-designed or standardized disposal canisters. Repackaging could be done wet in a fuel pool or wet handling facility, or dry in a large hot cell with a high bay (SNL 2021a, Section 1.5). Repackaging could occur at a reactor site, at a CISF, or at a repository, with appropriate wet or dry handling facilities. An as example, the Yucca Mountain repository design included a wet handling facility for repackaging uncanistered CSNF and CSNF from what was expected at the time to be a small number of DPCs into TAD canisters (DOE 2009a, Section 1.2.5).

Repackaging would be done for one or more of the following reasons: (1) to reduce canister size, and thus thermal output, for disposal concepts that use clay-based backfill or buffer materials contacting the waste package; (2) to reduce reactivity by limiting the amount of CSNF or by installing fillers for moderator exclusion; (3) to limit the size and weight of waste packages, for easier disposal handling and emplacement operations, (4) to provide for continued storage (e.g., after 100 years), and subsequent transportation and disposal, for cases where there are concerns about canister integrity during extended storage, and/or (5) to enable transportation, and disposal, of CSNF currently stored in storage-only systems (SNL 2021a, Section 1.5).

In general, repackaging CSNF into disposal-specific standardized canisters can reduce criticality potential during long-term disposal and help to integrate some or all of the storage, transportation, and disposal functions. The drawbacks are (1) the potential for additional occupational radiation doses to workers from repackaging operations, (2) the need for wet or dry handling facilities at locations that do not have spent fuel pools, (3) the need for added transfer capabilities and reactor spent fuel pool facilities, (4) the generation of low-level radioactive waste (LLW) from used DPCs, and (5) the added cost of the canisters and the handling facilities. In some cases, it is possible that repackaging needs could be met through a new overpack (e.g., "can-in-can") rather than requiring reopening a DPC and transferring the CSNF to a new canister.

Future R&D that could support repackaging, especially if needed at shutdown sites, would be to examine the possibility of developing a mobile repackaging system. A mobile system could be designed for repackaging CSNF from its current storage system into a standardized system, and to be disassembled, transported, and reassembled as needed to be used at multiple sites.

5.1.3.2 Loading CSNF Not Yet in Dry Storage

CSNF that has not yet been placed into dry storage or that has yet to be produced could be placed directly into a standardized canister when it is ready to be removed from the pool. This option offers the same advantages of repackaging, but without many of the drawbacks (Section 5.1.3.1).

However, similar to fuel and basket modifications for future DPCs (Section 5.1.2.4.3), the advantages of using standardized canisters for future-loaded CSNF depends on timely implementation. There is no realistic prospect for implementing a standardized canister system in the US before roughly 2030 at the earliest, by which time at least 60,000 MTHM of spent fuel will already be in dry storage in DPCs (Figure 2-3). Furthermore, there is currently no financial incentive for utilities to switch from their current dry storage canisters to a standardized canister.

5.2 Path to a Repository Next Century

The basis for an integrated waste management system that defers a repository until the next century is Scenario 4 (Table F-3). This scenario assumes disposal of 109,3000 MTHM of CSNF at a repository with characteristics equivalent to Yucca Mountain starting in 2117, which represents a 100-year delay from the TSLCC-based Reference Case (Scenario 1) timeline (Appendix F). This scenario has the following assumed schedule milestones:

- Repository Construction Authorization by the NRC in 2111
- Initial waste receipt and start of repository surface and subsurface operations in 2117
- End of 57-year period of waste emplacement in 2173
- End of closure operations in 2233
- With this timeline, all of the CSNF will be loaded into DPCs at the utility sites before a repository is available; those DPCs will all be subsequently transported to the repository where the CSNF will be repackaged into TAD canisters for disposal.

Variant scenario 4A examines options for direct disposal of DPCs without repackaging. Variant scenario 4B examines options for a federal CISF that opens in 2025. These variant scenarios are summarized in Table F-3.

Necessary R&D to support a next-century repository is similar to R&D needed for a mid-century repository (as described in Section 5.1), with a focus on extended storage and subsequent transportation, direct disposal of CSNF in DPCs, and/or repackaging of CSNF for disposal. Due to the long delay before a repository becomes available, additional R&D may be needed to address the extended storage for greater than 100 years in some cases. CSNF and cladding that has been stored for extended periods may require additional handling and preparation, or even repackaging (Section 5.1.4) for continued storage and/or before it can be transported. For example, the NRC suggestion that safe storage is possible for times exceeding 100 years is based on an assumption that spent fuel canisters and casks and ISFSI facilities would be replaced approximately once every 100 years (NRC 2014a).

5.3 Comparative Costs

The selected future spent fuel management scenarios and variants are summarized in Table F-3. As described in Appendix F, total life cycle costs for each scenario are based on the disposal of 109,300 MTHM of CSNF (the total projected US inventory at the time of the TSLCC (DOE 2008b) in 2008) in single repository, specifically Yucca Mountain. The estimated scenario costs reflect the commercial (CSNF) cost allocation only, costs associated with treating, storing, transporting, and disposing of defense wastes (DSNF and DHLW) are omitted to allow analysis to focus exclusively on the impacts of decisions related to the management of CSNF (Section 4.5). Additional key assumptions are identified in Appendix F. Comparison of the estimated costs between the alternative scenarios provides an indication of the economic effects and tradeoffs, which are driven by decisions about repository timing, canister selection (e.g., direct disposal of DPCs and/or repackaging), and a CISF.

The "baseline" mid-century scenarios (Scenarios 1, 2, and 3), summarized in Table F-3, all include the disposal of CSNF in TAD canisters at Yucca Mountain; any CSNF loaded into DPCs prior to the availability of TAD canisters will be transported in the DPCs and then repackaged into TAD canisters at the repository. The primary activities affected by the delays between the assumed repository opening dates of 2017 (Scenario 1), 2031 (Scenario 2), and 2041 (Scenario 3) are: more CSNF will be loaded into DPCs and will subsequently need to be repackaged; fewer rail shipments (DPCs are larger than TADs so more DPCs correlates to fewer overall canisters for the same mass of waste); continuing taxpayer liability for ongoing ISFSI costs. Variant scenarios 2B and 2C introduce activities related to direct disposal of DPCs at the repository, which reduces repackaging efforts. Variant scenarios 2D and 3A introduce activities associated with the opening of a CISF in 2025, which also increases (doubles) transportation needs.

The next-century baseline (Scenario 4) is similar to the mid-century scenarios, with disposal of CSNF in TAD canisters (Table F-3). However, the 100-year delay in availability of a repository is assumed to (1) cause the need the re-incur repository development and evaluation costs, and (2) delay the availability of TAD canisters such that all CSNF is loaded and transported in DPCs, where it will then be repackaged into TAD canisters. Variant scenario 4A includes direct disposal of DPCs at the repository instead of repackaging and variant scenario 4B includes the opening of a CISF in 2025.

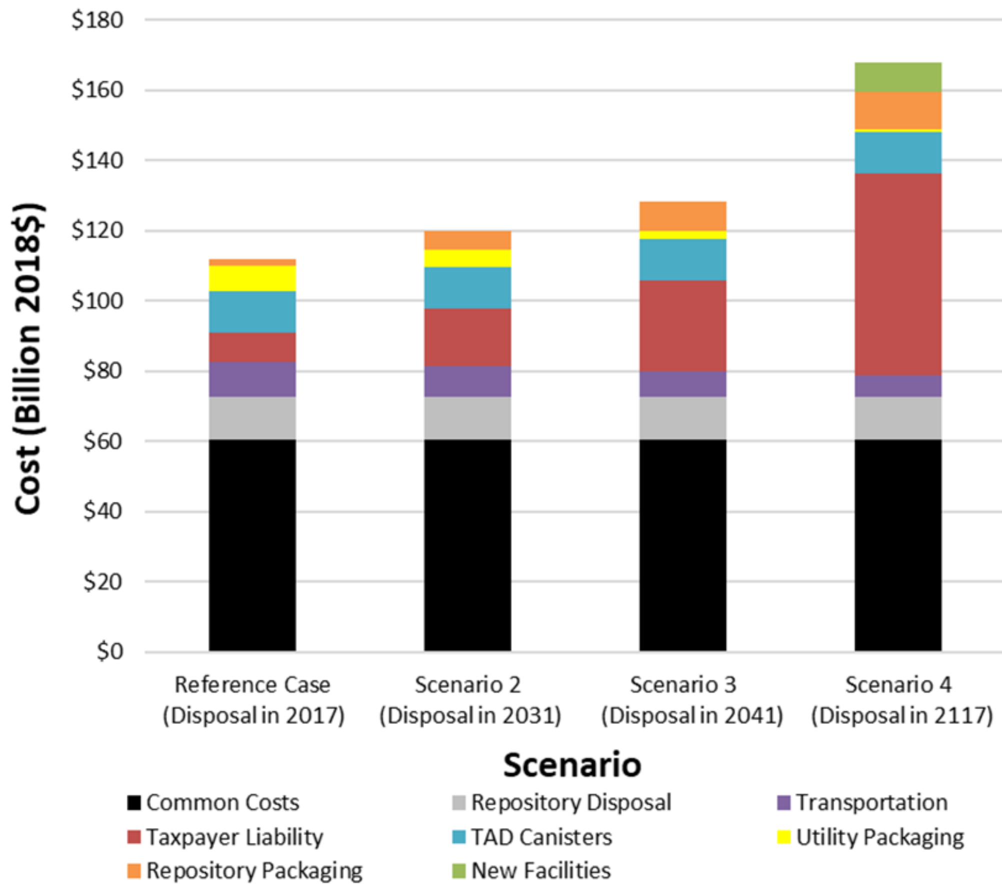
The estimated total life cycle costs for each of the scenarios and variants are shown graphically in Figure F-2 and detailed in Table F-4. The life cycle cost estimates include (1) common costs, which are primarily related to program management and repository development, construction, operation, and closure, and are assumed to be the same for each scenario, and (2) discriminating costs, which are primarily related to the relative number of and type of canisters (DPCs and/or TADs) and the associated storage, transportation, and disposal packaging operations, and are different between scenarios (Appendix F). Most costs would be paid from the Nuclear Waste Fund created by the NWPA, but some costs would be borne by taxpayers in the form of payments from the U.S. government "Judgment Fund" to nuclear utilities due to non-performance on Standard Contracts mandated by the NWPA (Section 1.1.). Subsets of the results shown in Figure F-2 are compared in the following subsections to emphasize the possible cost impacts of specific future spent fuel management decisions.

These estimates of life cycle costs are considered representative for the purposes of comparative analysis between scenarios, but they should not be taken as formal projections of the life cycle cost for any specific future scenario. For example, the effects of inflation beyond 2018 (e.g., the future value of money and/or different effects across scenarios with different time horizons) are not explicitly considered. Also, for simplicity and consistency in the cost comparisons, Yucca Mountain is assumed to be the repository for all of the alternative scenarios. Although there are likely to be cost differences between repository systems in different geologic media (e.g., salt, argillite, crystalline) (e.g., Hardin 2016; Hardin and Kalinina 2016), they were not considered in the cost comparisons. While certain geologies favor certain repository characteristics (e.g., hard rock unbackfilled or salt for thermal constraints (see Figure 4-2)), similar observations about comparative costs would likely arise from a repository in other geologic media.

5.3.1 Impacts of Delay in Repository Opening Date

Figure 5-1 (and Table F-4) compares the estimated total life cycle costs from Scenarios 1, 2, 3, and 4 to illustrate the possible impacts of a delay in repository opening, if all other factors are held constant. The estimated total costs are (Freeze et al. 2019a, Section 4.1.4.1):

- $112.1 billion (2018$) if disposal begins in 2017 (Scenario 1)
- $120.0 billion (2018$) if disposal begins in 2031 (Scenario 2)
- $128.4 billion (2018$) if disposal begins in 2041 (Scenario 3)
- $167.7 billion (2018$) if disposal is delayed to 2117 (Scenario 4)



(Source: Freeze et al. 2019a, Figure 4-5)

Figure 5-1. Comparison of Estimated Costs for Different Repository Opening Dates

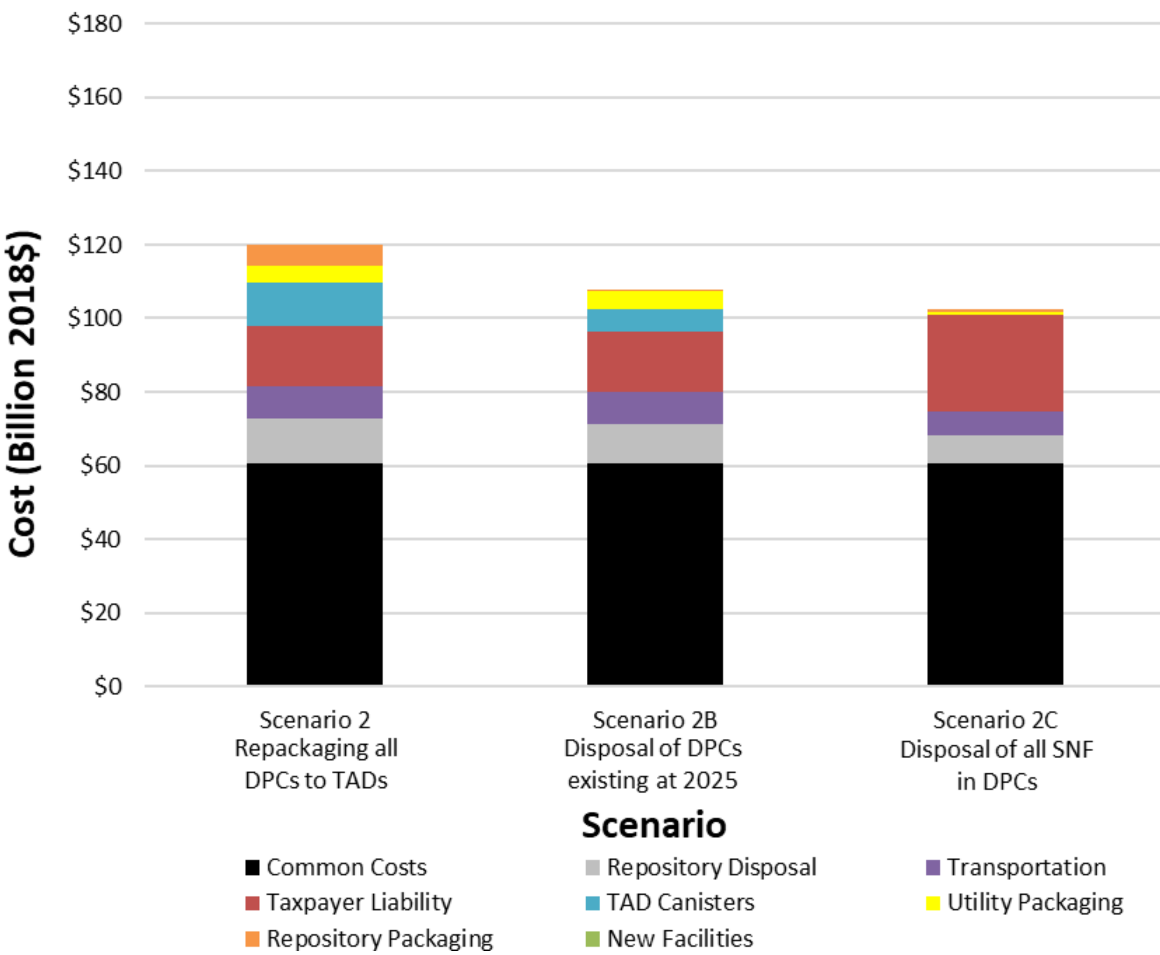
These scenarios all include the ultimate disposal of CSNF in TAD canisters. CSNF already loaded into DPCs will be repackaged into TAD canisters at the repository. The primary effect of the delay in repository opening is that (1) taxpayer liability continues longer because CSNF remains longer at the utility sites, (2) more DPCs will be loaded at the utility sites (and then need to be repackaged at the repository), and (3) transportation needs are reduced because DPCs are bigger than TAD canisters, resulting in fewer trips. The overall effect of a delay in the beginning of disposal operations is a steady increase in total program cost. The primary contributor to the cost increase is the ongoing taxpayer liability cost, paid through the Judgment Fund, driven by the continued loading of DPCs and by the annual operating costs associated with an increasing number of shutdown sites. The other effects on costs, from changes in DPC loading, transportation, and repackaging between scenarios, are comparatively minor.

The effects of the ongoing taxpayer liability are especially pronounced if repository opening is delayed to 2117 (Scenario 4), resulting in nearly $50 billion (2018$) in additional payments from the Judgment Fund as compared to the Reference Case. This increase includes about $15 billion (2018$) for loading more DPCs and about $35 billion (2018$) for continued operation of ISFSIs at shutdown sites. In addition, extra costs of $8 billion (2018$) for "redevelopment" of the Yucca Mountain design and licensing basis are assumed to be necessary following a 100-year delay.

5.3.2 Impacts from Disposal of DPCs Without Repackaging

Figure 5-2 (and Table F-4) compares the estimated total life cycle costs for disposal operations that begin in 2031 from Scenarios 2, 2B, and 2C to illustrate the possible impacts from disposal of DPCs without repackaging. The estimated costs are (Freeze et al. 2019a, Section 4.1.4.2):

- $120.0 billion (2018$) if no disposal of DPCs, full repackaging to TAD canisters (Scenario 2)
- $107.8 billion (2018$) if disposal of DPCs that exist as of 2025, TAD canisters loaded thereafter (Scenario 2B)
- $102.3 billion (2018$) if disposal of all CSNF in DPCs (Scenario 2C)



(Source: Freeze et al. 2019a, Figure 4-6)

Figure 5-2. Comparison of Estimated Costs for Different DPC Disposal Options (Repository Opens in 2031)

Figure 5-2 shows that, for a repository that opens in 2031, total life cycle costs can be reduced if some or all of the CSNF can be disposed of in DPCs without repackaging. Directly disposing of DPCs that exist as of 2025 (Scenario 2B) has the potential to reduce costs by about $12 billion (2018$) as compared to full repackaging (Scenario 2); directly disposing of all spent fuel in DPCs (Scenario 2C) has the potential to reduce costs by about $18 billion (2018$) as compared to full repackaging. The relative impacts would be similar for other mid-century repository opening dates. It should be noted that for Scenario 2C, the decreased costs from avoiding repackaging more than offset the increased taxpayer liability costs from continued loading of DPCs.

A similar cost reduction for direct disposal of DPCs is projected for a repository that opens in 2117, based on a comparison of the estimated total life cycle costs Scenarios 4 and 4A (Table F-4). The estimated costs are (Freeze et al. 2019a, Section 4.1.4.2):

- $167.7 billion (2018$) if no disposal of DPCs, full repackaging to TAD canisters (Scenario 4)
- $141.3 billion (2018$) if disposal of all CSNF in DPCs (Scenario 4A)

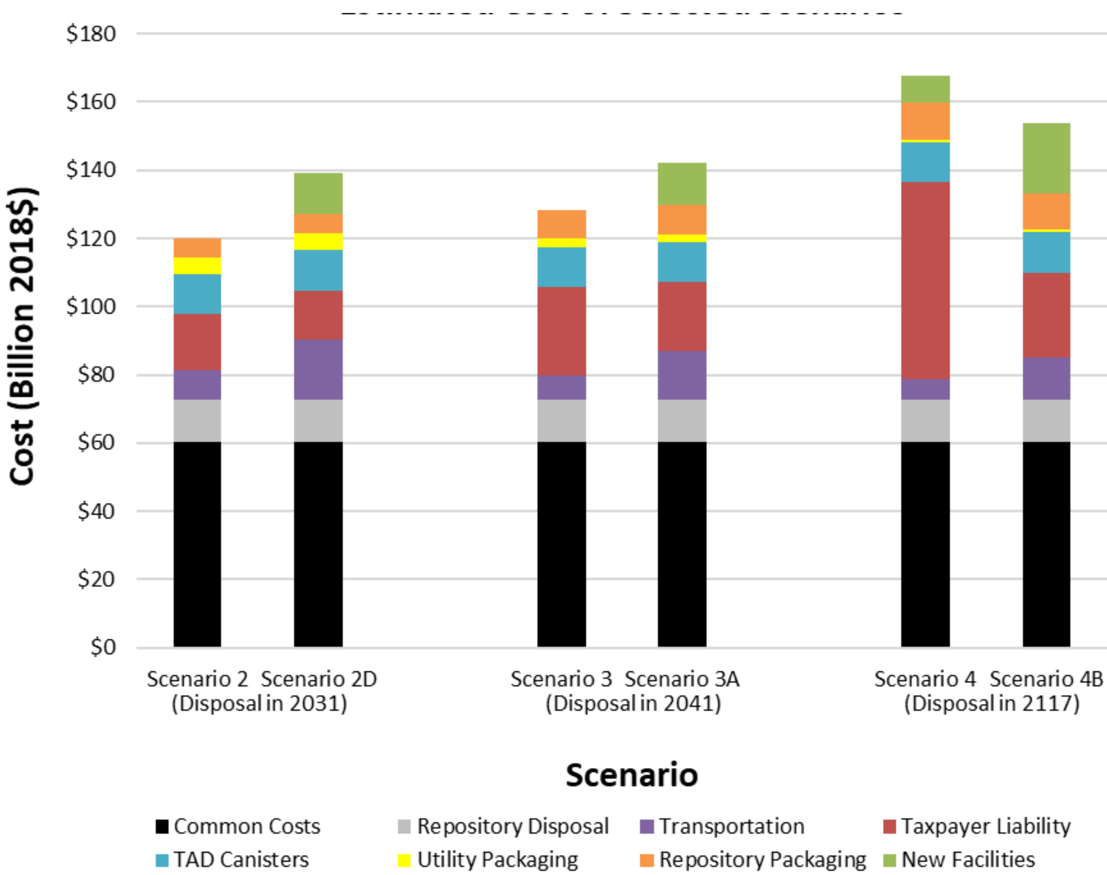
In this case, similar to the comparison of Scenarios 2 and 2C, total life cycle costs can be reduced by disposing of CSNF in DPCs without repackaging; directly disposing of all spent fuel in DPCs (Scenario 4A) has the potential to reduce costs by about $26 billion (2018$) as compared to full repackaging (Scenario 4).

In addition to the cost savings, the direct disposal of DPCs avoids the potential for additional occupational radiation doses to workers from repackaging operations.

5.3.3 Impacts from Federal Consolidated Interim Storage

Figure 5-3 (and Table F-4) compares the estimated total life cycle costs from Scenarios 2 and 2D, Scenarios 3 and 3A, and Scenarios 4 and 4B, to illustrate the possible impacts from a federal CISF. The estimated costs are (Freeze et al. 2019a, Section 4.1.4.3):

- $120.0 billion (2018$) if disposal begins in 2031 with no CISF (Scenario 2)
- $139.3 billion (2018$) if disposal begins in 2031 with a CISF open in 2025 (Scenario 2D)
- $128.4 billion (2018$) if disposal begins in 2041 with no CISF (Scenario 3)
- $142.0 billion (2018$) if disposal begins in 2041 with a CISF open in 2025 (Scenario 3A)
- $167.7 billion (2018$) if disposal begins in 2117 with no CISF (Scenario 4)
- $153.6 billion (2018$) if disposal begins in 2117 with a CISF open in 2025 (Scenario 4B)



(Source: Freeze et al. 2019a, Figure 4-8)

Figure 5-3. Comparison of Estimated Costs for Scenarios with CIS and Different Repository Opening Dates

The primary conclusion drawn from this comparison is that the relative cost impact of implementing a CISF depends on the date at which the repository begins disposal operations.

- If the repository is available relatively soon after the CIS facility begins operations (e.g., Scenarios 2D and 3A), then the increased costs associated with construction, operation, and transportation for the CIS facility are greater than the savings associated with earlier termination of the Judgment Fund liabilities, resulting in an overall increase in scenario cost of as much as $20 billion (2018$).
- If disposal operations are delayed by 100 years (e.g., Scenario 4B), cost savings from the CIS facility, due primarily to early termination of the Judgment Fund liabilities, are estimated to be about $14 billion (2018$).

5.4 Regulatory, Legal, and Stakeholder Considerations

The future scenarios and cost estimates in Section 5.3 included the quantitative effects of technical and programmatic considerations. However, additional factors for consideration in decisions about future spent fuel management options include (Freeze et al. 2019a, Section 4):

- Regulatory considerations – needs for new or modified regulations or legislation
- Legal sensitivities – ongoing litigation between the U.S. government and utilities
- Socio-political influences – government and public opinion towards nuclear waste and potential storage or disposal sites, stakeholder perceptions, political and social costs of delays
- Siting and licensing of future storage or disposal facilities
- Structure and funding for a nuclear waste management implementing organization
- Knowledge management

These additional considerations, while important to decision-makers, are subject to intangible factors that make quantification of their effects impractical and were therefore not included in the cost estimates.

5.4.1 Regulatory and Legal Sensitivities

The future scenarios identified in Appendix F and used to examine cost tradeoffs in Section 5.3 are generally based on the NWPA – disposal at Yucca Mountain in TAD canisters, with taxpayer liability for the ongoing use of DPCs. Other alternatives such as direct disposal of DPCs, a federal CISF (prior to a license for construction of a repository), and/or a repository at a location other than Yucca Mountain, would require amendments to the NWPA. Modifications to the NWPA, regulations, legislation, and/or the Standard Contract to promote back end integration and progress toward a repository could be motivated by the projected cost savings.

Specific regulatory changes that might be necessary include (1) updates to the disposal requirements at 10 CFR Part 63 and 40 CFR Part 197 if a site other than Yucca Mountain is ultimately selected, and (2) developing a uniform set of regulatory requirements for CSNF that spans the storage, transportation, consolidated interim storage (if used), and repository operation phases to integrate system component inspections (e.g., canister, fuel, cladding integrity), as described in Sections 3.2 and 4.4.1 (i.e., "72-71-72-63" sequencing).

5.4.2 Siting and Socio-Political Influences

As noted in Section 4.4, public perceptions and preferences influence waste management policy, and thereby also influence attempts to integrate the storage, transportation, and disposal of SNF and HLW. Unexpected nuclear-related events, such as the 2011 accident at Fukushima (Section C.1.7), can lead to shifts in public perception. SNL partners with the University of Oklahoma to analyze public opinion about specific aspects of the commercial nuclear fuel cycle; insights from some national public opinion surveys are presented in Section 4.4.3.

If a location other than Yucca Mountain becomes an option for a repository, the siting process will need to be well thought out. Finding sites where all affected units of government, including the host state or tribe, regional and local authorities, and the host community, are willing to support or at least accept a facility has proved exceptionally difficult. Experience in the US and in other nations suggests that any attempt to force a top-down, federally mandated solution over the objections of a state or community—far from being more efficient—will take longer, cost more, and have lower odds of ultimate success (BRC 2012, Executive Summary). The erosion of trust in the federal government's nuclear waste management program has only made this challenge more difficult (Section 4.4).

Several other countries have restarted their repository siting process with input, communication, and negotiation with interested host communities (e.g., Canada (NWMO 2010; NWMO 2021), Germany (BWMi 2021), Switzerland (SFOE 2008; Nagra 2021)). In the US, technical guidelines for siting were published in 10 CFR Part 960 in accordance with the NWPA, prior to the identification of Yucca Mountain as the only site for evaluation as a repository in the NWPAA (Section 1.1). However, any new US siting process, for a repository or for a federal CISF, should encourage expressions of interest from a large variety of communities that have potentially suitable sites (BRC 2012, Chapter 6) and involve early and effective communication and information sharing between the implementing organization and potentially affected stakeholders (state, tribal, regional, and local governments and members of the host community).

Associated with the siting process is the need for early preparation for a large-scale transportation program. This includes (1) early implementation of institutional arrangements, technical assistance and funding for state, tribal and local government to cover procedures required for safe routine transportation of SNF and HLW, as well as procedures for dealing with emergency response situations, as mandated by the NWPAA (see Sections 5.1.1.4 and C.2.4) and (2) possibly amending the NWPA to allow states to recover the full costs of planning and operations for transportation across their borders and ensuring an independent regulator has authority over the transportation regime to strengthen public confidence in the program (Bowen 2021, Executive Summary).

5.4.3 Implementing Organization and Funding

The Yucca Mountain licensing process was managed by OCRWM within DOE, as prescribed by the NWPA. With the suspension of the Yucca Mountain licensing process, funding for OCRWM was terminated (Section 1.2). Resumption of the Yucca Mountain licensing process and/or initiation of a process to select a different repository site could benefit from a dedicated implementing organization; it could be the re-establishment of OCRWM or a new management model.

BRC (2012, Chapter 7) recommended management by an independent federal corporation chartered by Congress with a well-defined mission, access to adequate resources, ability to make binding contractual commitments, and subject to rigorous external oversight.

Davis et al. (2012) reviewed different organizational models, including federal government corporations, federally chartered private corporations, and independent government agencies. The review included a look at the major problems in nuclear waste management in the past decades in five areas: (1) governance and leadership, (2) funding and budget control, (3) the siting process, (4) procurement and personnel rules, and (5) the public trust. Davis et al. (2012, Summary) concluded that the organizational design of OCRWM within DOE contributed less to the problems than the NWPA itself and subsequent congressional and executive branch actions, most notably the 1987 NWPAA, which designated Yucca Mountain as the only candidate repository site, over the objections of Nevada, and changes in budgeting that severely restricted the Secretary of Energy's access to the Nuclear Waste Fund. As flawed as OCRWM's program implementation may have been at times, it is difficult to imagine that any organization could have successfully executed the program in the absence of both public support in the affected state and sustained funding from Congress, itself an indicator of public support (Davis et al. 2012, Summary). Davis et al. (2012, Chapter 5) identified the critical attributes of a management and disposition organization to include, for example, political credibility and influence to accomplish consent-based siting of the facilities; accountability; transparent decision-making; ability to commit to providing incentives; insulation from political control; a public interest mission; and organizational stability (durability and consistency in policy and management) and noted that these attributes exist or could be built into a federal government corporation or an independent government agency.

To succeed, a new waste management organization must have the resources needed to implement an effective program, which is dependent on making the revenues generated by the nuclear waste fee and the balance in the Nuclear Waste Fund available when needed and in the amounts needed to implement the program (BRC 2012, Chapter 8). The US nuclear waste program was supposed to be self-financing, with the Nuclear Waste Fund covering the costs of management and disposal. However, due in part to budget laws enacted in the 1980s and 1990s, a lack of access to needed funding has arisen (Bowen 2021). Federal appropriations are supposed to be able to access the Nuclear Waste Fund when needed, and in the amounts needed, to implement the waste program without facing competition from other funding priorities. However, Congressional oversight through the annual appropriations process, meant to ensure that expenditures from the Nuclear Waste Fund would be made prudently and for their intended purposes, has resulted in waste management needs competing with other priorities in DOE's annual budget request and in the Congressional appropriations process, subjecting the program to exactly the sort of "budgetary perturbations" that the funding mechanism was intended to avoid (BRC 2012, Chapter 8).

5.4.4 Knowledge Management

The BRC (2012, Chapter 11.4) notes that "over the next five years, half of the nation's nuclear utility workforce will need to be replaced." As the Yucca Mountain licensing process remains suspended and no other sites are under consideration, the institutional nuclear waste management knowledge base is undergoing similar aging and attrition. Section C.3 outlines plans for a knowledge management approach to archive and transfer institutional nuclear waste management knowledge, with a focus on relevance/benefit to the DOE-NE SFWST R&D program.

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6. CONCLUSIONS

The NWPAA (1987) identified Yucca Mountain as the only site for evaluation as a repository. However, the Yucca Mountain repository licensing process remains suspended with no funding, Congress has made no change to the law, and the US government is considering no other sites for disposal. If Congress chooses to abandon the Yucca Mountain site permanently and pursue as-yet-unidentified alternatives, additional revisions or amendments to the NWPA would be necessary.

In the NWPA, the nuclear utilities have the primary responsibility for providing at-reactor interim storage of CSNF until it is accepted by the federal government, which is responsible for transportation, away-from-reactor storage (if applicable), and disposal. Without a permanent disposal option this separation of responsibilities has led the utilities, driven by economic and worker safety considerations, to load the ever-increasing inventory of CSNF into larger DPCs that are not designed for disposal. The longer-than-planned storage of the larger DPCs will lead to the need for ISFSI license extensions, potentially delay transportation, and complicate disposal.

The DOE has still not begun to take title to CSNF, and taxpayers, via the Judgment Fund, continue to pay utilities for costs of on-site management of CSNF. The trend of the last decade for older and less profitable nuclear power plants to close seems likely to continue, leaving increasing quantities of CSNF stored at sites where there is no operating reactor.

As a result, the current practice for spent fuel management in the US lacks integration between storage, transportation, and disposal. Options for improving integration of these current management practices across the back end of the nuclear fuel cycle include:

- Constructing one or more repositories that can accommodate DPCs without repackaging
- Repackaging SNF in the future before disposal
- Storing SNF at surface facilities indefinitely, repackaging as needed

Each of these options is technically feasible, but none is what was originally planned, and each comes with significant drawbacks. Constructing one or more repositories that can accommodate DPCs is technically feasible, but it will surely complicate the task of siting and licensing a disposal facility, and could add additional delay to a process that is already decades behind schedule. Repackaging SNF that has already been loaded into DPCs will be expensive and will unavoidably result in additional occupational radiation dose to workers that would be avoided if the canisters could be sent directly to disposal. Storing spent fuel at surface facilities indefinitely can be done safely given a commitment to indefinite security and maintenance with repackaging as needed (NRC 2014a), but it commits future generations to the financial costs of spent fuel management and fails to achieve the long-recognized goal of permanent geologic disposal.

Nonetheless, improved integration can be achieved through some combination of these three options, enabled by regulatory and legislative actions, as needed, and by supporting R&D. A key factor that influences the path forward for integration of nuclear waste management practices is the identification of the timing and location for a repository. Alternatives range from a restart of Yucca Mountain licensing proceedings in the next decade to development of a new repository at a different location several decades or more from now.

A set of possible future spent fuel management options, described in Section 5, were developed based on assumptions about the timing of repository availability (2031, 2041, or 2117) and choices about integration between storage, transportation, and disposal that combine one or more of the three aforementioned options: direct disposal of CSNF in DPCs without repackaging, repackaging of existing CSNF and/or loading future CSNF into disposal-ready canisters, and extended dry storage of CSNF at utility sites and/or a CISF.

Based on comparisons of estimated system life cycle costs for these future scenarios, the most cost-effective path forward would be to open a repository by mid-century with the capability to directly dispose of DPCs without repackaging. Options involving repackaging of CSNF from DPCs into disposal-ready (e.g., TAD) canisters are also feasible, but could add anywhere from $6 to 18 billion (2018$) to the total system life cycle cost. Delaying repository opening until 2117 is the most expensive option, adding nearly $50 billion (2018$) in additional taxpayer liability paid from the Judgment Fund.

The feasibility of these integrated spent fuel management options is supported by experiments, modeling, and analyses, described in Section 5 and Appendix C, which:

- suggest that system components (specifically, spent fuel, cladding, and canisters) will maintain adequate thermal, mechanical, and chemical behavior under extended dry storage conditions such that subsequent transportation to a CISF and/or repository is practical.
- suggest that, in addition to the Yucca Mountain design, repository designs capable of sufficient postclosure isolation are feasible in a range of other geologic media including salt, argillite, and crystalline rock.
- make recommendations for additional R&D to more fully explore the technical bases for the future scenarios.

The implementation of any of these integrated spent fuel management options would also need to address regulatory, legal, and stakeholder considerations, as outlined in Section 5.4.

In summary, the sooner a repository location and design can be identified, the more flexibility there will be in future spent fuel management options. The longer it takes to reach a decision, the more expensive the solution will be. Current efforts should focus on identifying barriers to implementing integrated SNF management practices that can be addressed within existing policy constraints. Even small actions now can have large impacts over the decades during which the current reactor fleet will continue to operate, and potential benefits to the US could be significant.

7. REFERENCES

10 CFR Part 50. Domestic Licensing of Production and Utilization Facilities.

10 CFR Part 51. Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions.

10 CFR Part 52. Licenses, Certifications, and Approvals for Nuclear Power Plants.

10 CFR Part 54. Requirements for Renewal of Operating Licenses for Nuclear Power Plants.

10 CFR Part 60. Disposal of High-Level Radioactive Wastes in Geologic Repositories.

10 CFR Part 63. Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada.

10 CFR Part 71. Packaging and Transportation of Radioactive Material.

10 CFR Part 72. Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.

10 CFR Part 960. General Guidelines for the Preliminary Screening of Potential Sites for a Nuclear Waste Repository.

10 CFR Part 961. Standard Contract for Disposal of Spent Nuclear Fuel and/or High-Level Radioactive Waste.

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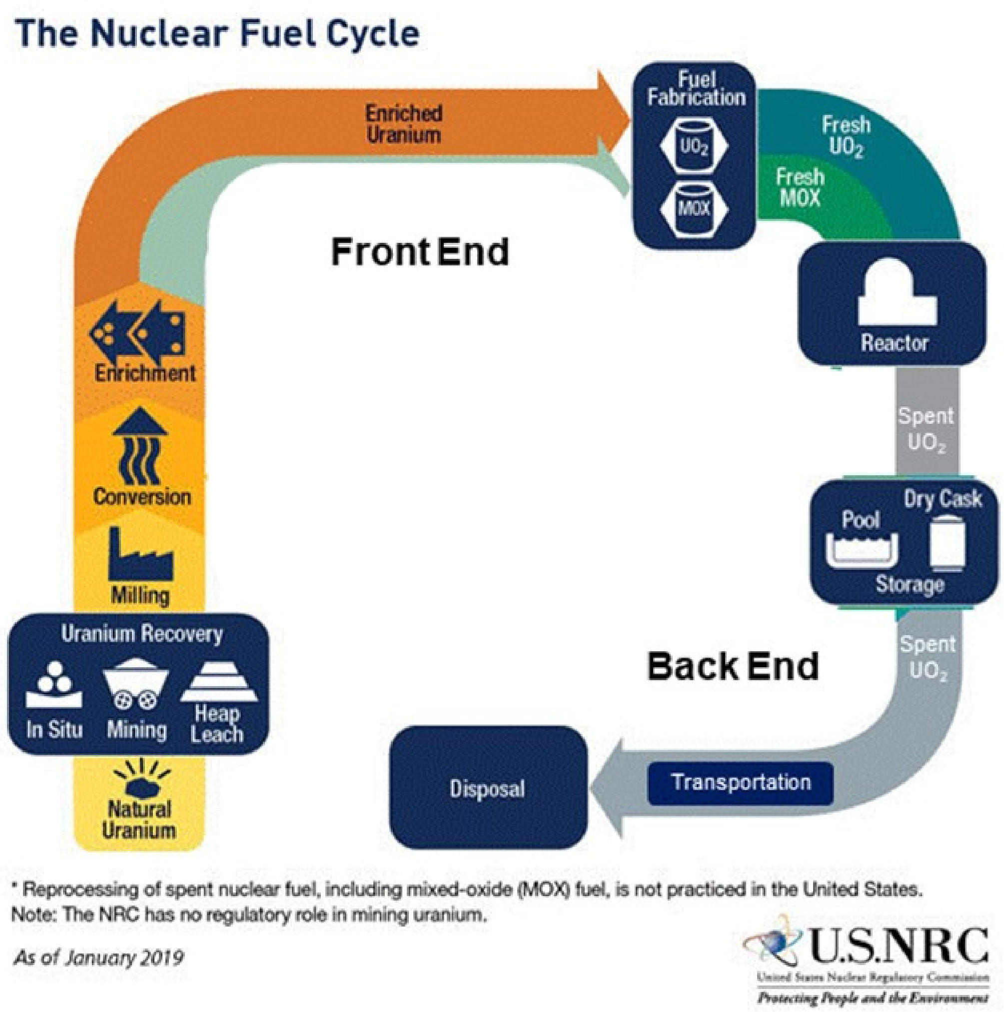
APPENDIX A. ONCE-THROUGH NUCLEAR FUEL CYCLE

The nuclear fuel cycle can be described as having three major parts: the "front end" where uranium is mined and processed into fuel for use in a nuclear reactor; the use of that fuel in a reactor; and the "back end" where the spent fuel is first stored and ultimately sent for disposal or reprocessing (if the spent fuel is reprocessed, remaining wastes would still require disposal) (BRC 2012, Chapter 3.1).

All 94 of the operating commercial power reactors in the US are light-water reactors (LWRs); 63 are pressurized water reactors (PWRs) and 31 are boiling water reactors (BWRs). These LWRs use a "once-through" or open fuel cycle (i.e., there is no reprocessing of spent fuel) (Figure A-1).



Note: Reprocessing of SNF, including mixed-oxide (MOX) fuel, is not practiced in the US.

(Source: adapted from NRC 2020i)

Figure A-1. Stages of the Once-Through Nuclear Fuel Cycle

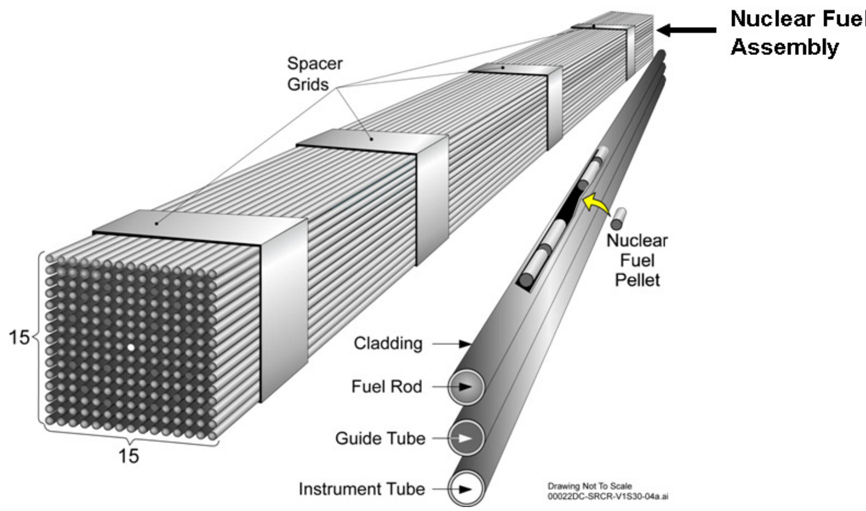
A.1. Stages of the Nuclear Fuel Cycle

The LWR once-through nuclear fuel cycle (Figure A-1) typically includes the following stages (MIT 2011, Chapter 2; BRC 2012, Chapter 3.1; NRC 2020i):

- *Uranium recovery (mining and milling)*. Uranium provides the basic fissile material or "fuel" for nearly all nuclear reactors. Mining extracts uranium ore; extracted uranium consists almost entirely of two isotopes, mostly ^{238}U (99.3%) together with a much smaller fraction (0.7%) of the fissionable isotope ^{235}U. In its natural state, uranium ore is only weakly radioactive – meaning that it can be handled without the need for radiation shielding. Milling produces a uranium ore concentrate, U_3O_8 or "yellowcake". About 200 tons of natural uranium is mined to fuel a 1000-MW(e) LWR for one year (MIT 2011, Chapter 2).
- *Uranium conversion*. The uranium ore concentrate is chemically purified and converted into uranium hexafluoride (UF_6).
- *Uranium enrichment*. Before it can be used in a commercial reactor, natural uranium must be purified and enriched to boost the amount of fissionable ^{235}U present in the fuel. In the uranium enrichment process, natural uranium is converted into an enriched uranium product stream containing 3 to 5% ^{235}U (required for today's LWRs), and depleted uranium that contains ~0.3% ^{235}U. Techniques for enriching uranium are well developed, with the most prominent methods involving gaseous diffusion or centrifuge technology.
- *Fuel fabrication*. The enriched uranium is converted into uranium dioxide (UO_2), which is cast into hard pellets and stacked inside long metal tubes or "cladding" to form fuel rods. The fuel rods are bundled into fuel assemblies (each assembly is about 12 to 14 feet long). Figure A-2 shows a typical PWR fuel assembly.
- *Light-water reactor*. When fresh fuel is loaded into a reactor, the fissioning of ^{235}U produces heat. The ^{238}U, upon absorption of neutrons, produces ^{239}Pu, which also fissions to produce heat. Just before the fuel is discharged from the reactor as SNF, about half the energy being generated is from the fissioning of ^{239}Pu that was created in the reactor. The heat is converted into electricity.
- *Storage of SNF*. A typical LWR fuel assembly remains in the reactor for three to four years. Upon discharge, the SNF is stored in wet and/or dry storage systems to reduce radioactivity and radioactive decay heat before disposal.
- *Transportation of SNF*. Waste must be transported from storage sites to a geologic disposal facility. Additional transportation would be necessary if an interim storage facility is used.
- *Waste disposal*. In the once-through fuel cycle, where the fissile ^{238}U and ^{239}Pu are not reprocessed, the SNF is considered waste that ultimately must be sent to a geologic repository for disposal.

In the reactor, the enriched uranium fuel sustains a series of controlled nuclear fission reactions that produce (1) energy (heat) that is converted to steam to drive turbines that generate electricity, and (2) SNF, including uranium, plutonium, minor actinides, fission products, and neutron activation products.

Nuclear fission is a process in which the nucleus (protons and neutrons) of a particle splits into smaller parts (lighter nuclei). The fission process often produces free neutrons and photons (in the form of gamma rays) and releases a very large amount of energy. Much of the fission yield is concentrated in two peaks, with atomic masses centering near 95 (e.g., Se, Sr, Zr, Mo, Tc, Ru, Rh, Pd, Ag) and 135 (e.g., Te, I, Xe, Cs, Ba, La, Ce, Pr, Nd, Sm, Eu, Gd) (Bruno and Ewing 2006, Table 1). Many of the fission products are either non-radioactive or very short-lived (half-life on the order of days or less) radioisotopes. But a considerable number are longer-lived (half-lives from tens to millions of years).



(Source: Carter et al. 2016b)

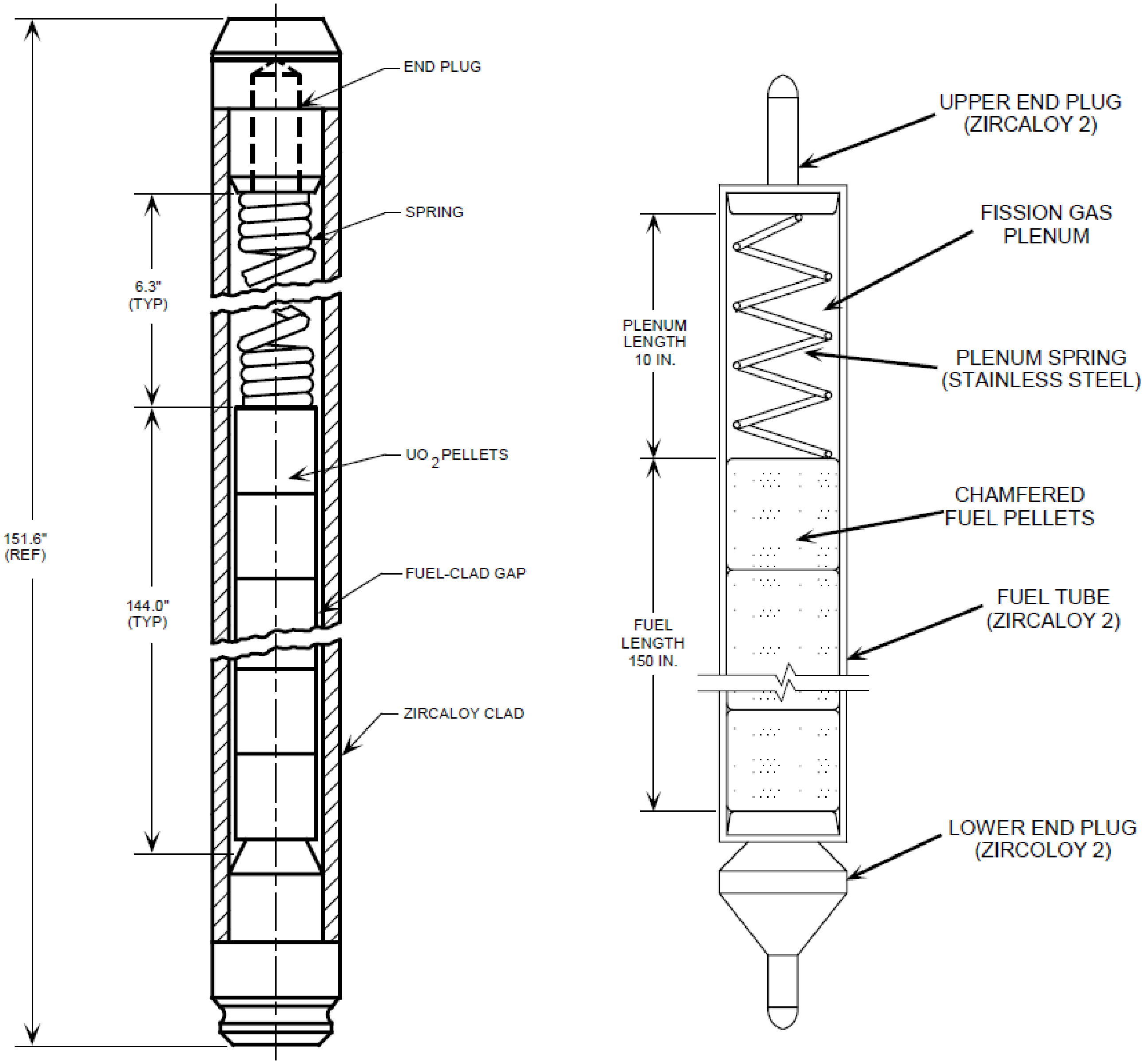
Figure A-2. PWR Nuclear Fuel Assembly

A.2. Components of the Back End of the Nuclear Fuel Cycle

A.2.1. Fuel Rods

Nuclear fuel for an LWR consists primarily of cylindrical pellets of compacted enriched UO_2 stacked inside long corrosion-resistant metallic tubes or "cladding" to form fuel rods. The cladding material is most commonly a zirconium alloy, although a small number of early designs used stainless steel. Differences between PWR and BWR operating environments led to differences in cladding materials; Zircaloy-2 is most commonly used in BWRs whereas Zircaloy-4, and more recently ZIRLO^{TM} and M5^{TM}, are most commonly used in PWRs (Hanson et al. 2012, Section 5). Schematics of typical nuclear fuel rods (PWR and BWR) containing general dimensions and parts are shown in Figure A-3.

The cylindrical UO2 pellets, which can range from 0.3 in (8 mm) to 0.4 in (10 mm) in diameter and 0.35 (9 mm) to 0.6 in (15 mm) in thickness (Bruno and Ewing 2006; NRC HRTD 2009, Table 3.1-3), are stacked in the cladding and are prevented from shifting by a stainless-steel spring in the plenum region at the top of the rod. The top and bottom ends of each pellet are dished slightly to allow greater axial expansion at the center of the pellets. The fuel rod is then plugged and seal-welded at both ends to encapsulate the fuel. Fuel rods are internally pressurized with helium during welding to minimize compressive clad stresses and to prevent cladding flattening due to coolant operating pressures (NRC HRTD 2009, Section 3.1.3.3). The outside diameter of a typical fuel rod ranges from 0.4 in (10 mm) to 0.5 in (13 mm) (NRC HRTD 2009, Table 3.1-3; NRC HRTD 2011, Section 2.2.2.2).



Specific dimensions depend on design variables such as pre-pressurization, power history, and discharge burnup

(Source: NRC HRTD 2009, Figure 3.1-20) (Source: NRC HRTD 2011, Figure 2.2-3)

a) PWR Fuel Rod **b) BWR Fuel Rod**

Figure A-3. Fuel Rod Schematics

A.2.2. Fuel Assembly

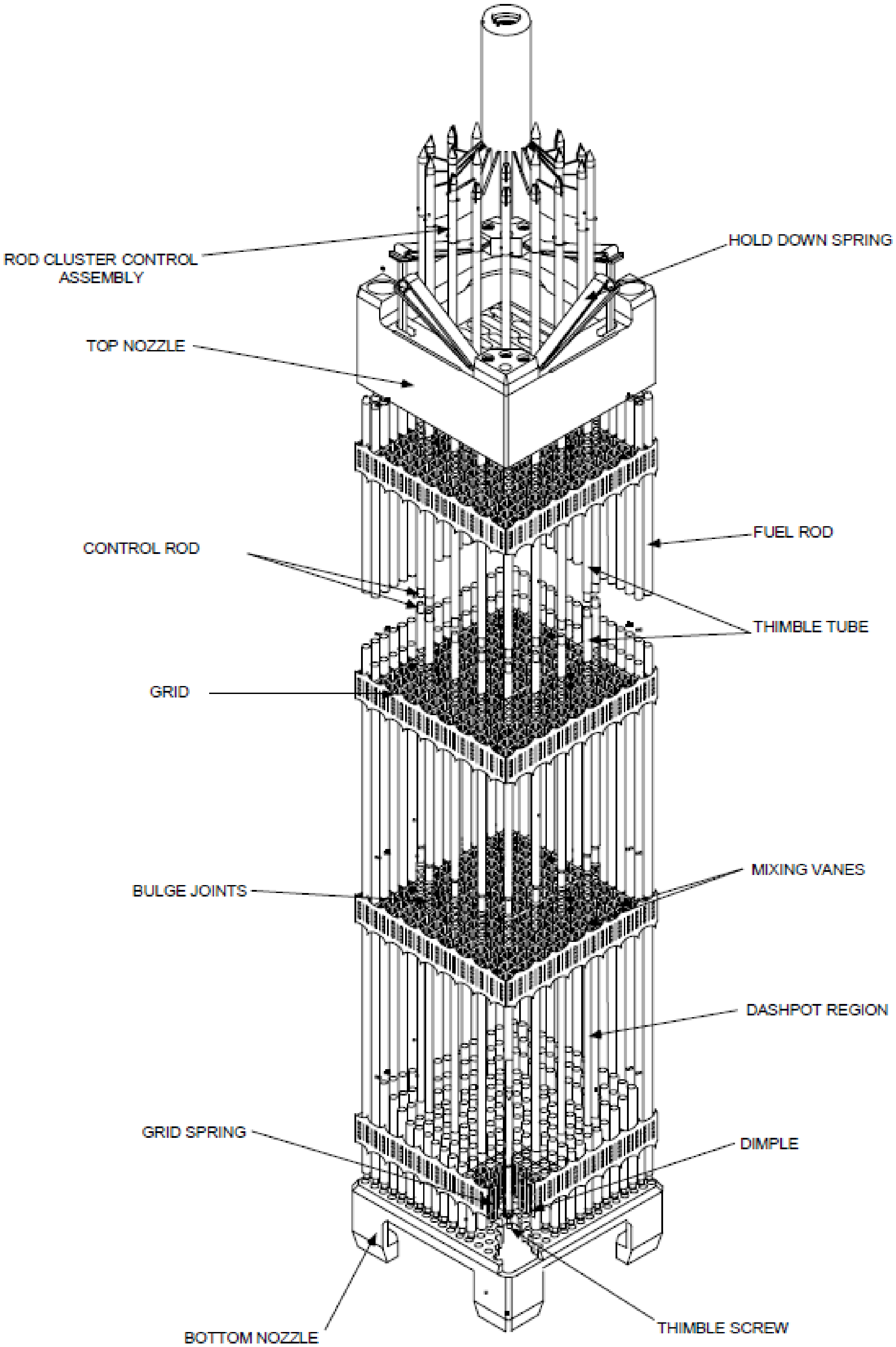
To obtain more efficient and effective reactor operation, fuel rods are bundled together with steel hardware into fuel assemblies. Fuel assemblies vary in physical configuration, depending on reactor type and manufacturer. LWR assemblies are generally configured in square arrays to accommodate the fuel rods as well as a smaller number of reactivity control rods. PWR assemblies range from 14x14 to 17x17 arrays, with lengths most commonly ranging from about 137.1 in (3.5 m) to 165.7 in (4.2 m) and widths most commonly ranging from about 7.76 in (0.20 m) to 8.54 in (0.22 m) (Peters et al. 2020, Table A-1). BWR assemblies range from 7x7 to 11x11 arrays, with lengths most commonly ranging from about 134.4 in (3.4 m) to 176.2 in (4.5 m) and widths most commonly ranging from about 4.28 in (0.11 m) to 5.44 in (0.14 m) (Peters et al. 2020, Table A-2).

A typical PWR fuel assembly is shown in Figure A-4. The PWR assembly shown is 17x17, consisting of 264 fuel rods, 24 guide thimble tubes, and 1 instrumentation guide thimble; structural support is provided by the top nozzle, the bottom nozzle, the spacer grids, and the guide thimble tubes (NRC HRTD 2009, Section 3.1.3.3).

The fuel rods are loaded into the fuel assembly structure so that there is clearance between the fuel rod ends and the top and bottom nozzles to allow for expansion without causing bowing of the rods (NRC HRTD 2009, Section 3.1.3.3). The spacer grids (Figure A-5) provide an "egg-crate" type structure to separate the individual fuel rods. Coolant water flow is directed upward by the bottom nozzle, around the fuel rods and through the assembly. The coolant flow exits the assembly through the top nozzle. The PWR top nozzle also houses a spider and hub that is used to connect and drive the control rods. The neutron absorbing control rods can be raised or lowered within the guide thimble tubes to control the reactivity (fission) in, or shutdown, the reactor. In a Westinghouse PWR design, the spider, hub, and control rod grouping is called a rod cluster control assembly (RCCA). Depending on the position of a particular assembly within the reactor core, the guide thimble tubes may also provide channels for insertion of a neutron source assembly, a burnable poison assembly, or a thimble plug assembly ((NRC HRTD 2009, Section 3.1.3.3). Soluble boron (boric acid) is added to the cooling water for additional neutron absorption control.

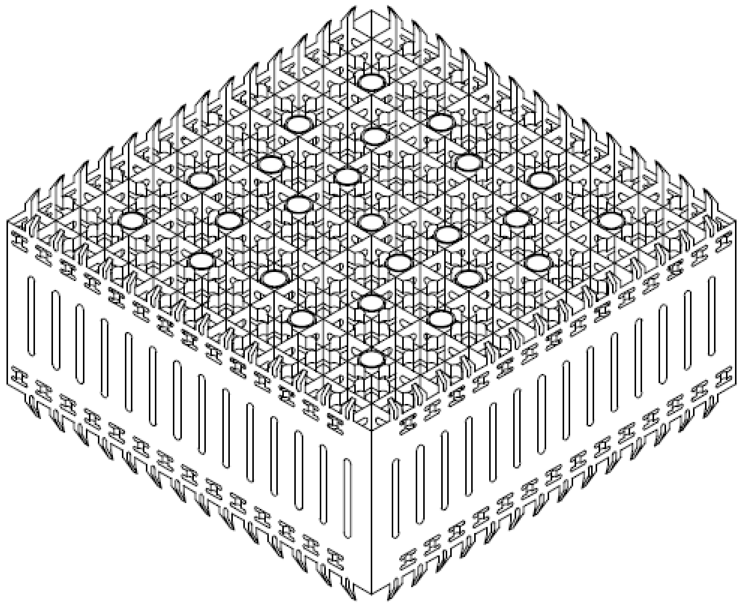
A typical BWR fuel assembly is shown in Figure A-6. The BWR assembly shown is 10x10, consisting of 70 fuel rods, 14 partial length fuel rods, 8 fuel tie rods, and 2 water rods (NRC HRTD 2011, Section 2.2.2.3). It has similar components and function to a PWR assembly, but there are a few differences (NRC HRTD 2011, Section 2.2.2):

- There are fewer fuel rods per BWR assembly.
- Each BWR assembly is enclosed by a metal (Zircaloy) fuel channel.
- the water rods increase coolant flow to the center of the assembly, providing for more even fuel burn across the assembly.
- Control rods are typically located between four BWR assemblies, rather than within the assemblies, as in a PWR.



(Source: NRC HRTD 2009, Figure 3.1-16)

Figure A-4. PWR Fuel Assembly Schematic (17x17)



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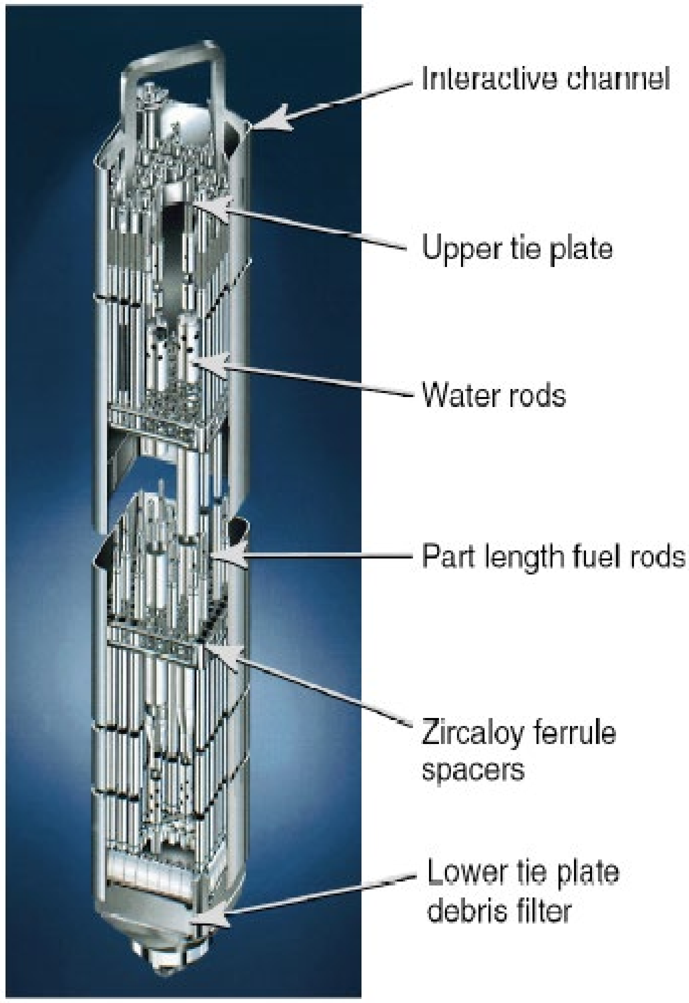
(Source: NRC HRTD 2009, Figure 3.1-22)

Figure A-5. PWR Spacer Grid Schematic (17x17)



(Source: NRC HRTD 2011, Figure 2.2-6)

Figure A-6. BWR Fuel Assembly Schematic (10x10)

A typical PWR assembly weighs approximately 700 kg (Wagner et al. 2012, Figure A-1), including, on average, 431 kg of uranium (i.e., 0.43 MTU) (SNL 2014b, Table A-2). A typical BWR assembly weighs approximately 300 kg (Wagner et al. 2012, Figure A-2), including, on average, 179 kg of uranium (0.18 MTU) (SNL 2014b, Table A-2). The weight of the cladding and hardware (e.g., top and bottom nozzles, spacer grids) accounts for the non-uranium mass of the assembly.

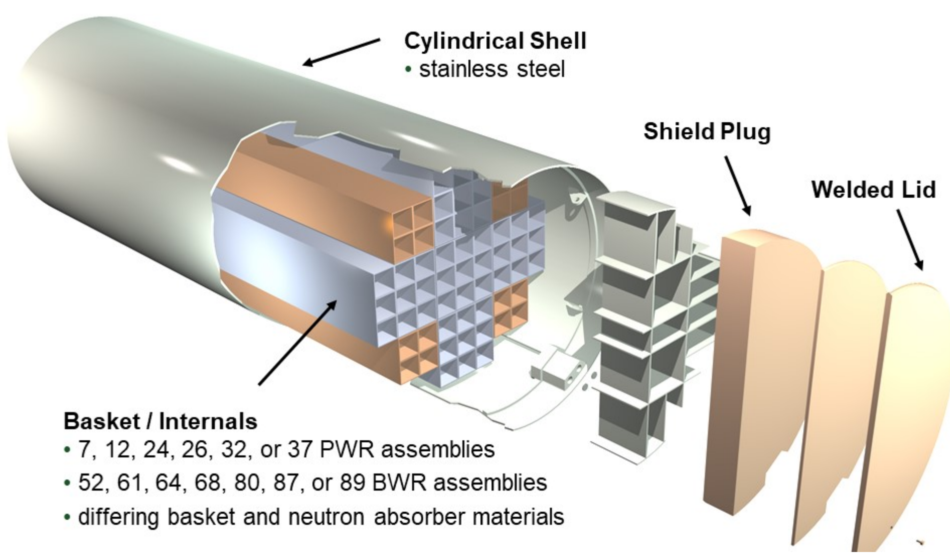
When spent fuel is removed from a reactor, it is removed as an entire assembly (including fuel rods, top and bottom nozzles, and spacer grids) and transported to the storage pool. Control rods/RCCAs are typically left in the reactor to be used with another assembly.

A.2.3. Storage Canisters and Casks

Multiple dry cask storage systems (DCSSs) are in use in the US today. The most common are canister-based systems where a single sealed (welded) canister (e.g., a DPC) containing CSNF is placed inside a single concrete (or concrete and steel) storage module (i.e., a cask or overpack) for physical protection, radiation shielding, and thermal management. Some canisters are licensed to take damaged fuel and/or fuel debris in addition to intact spent fuel assemblies.

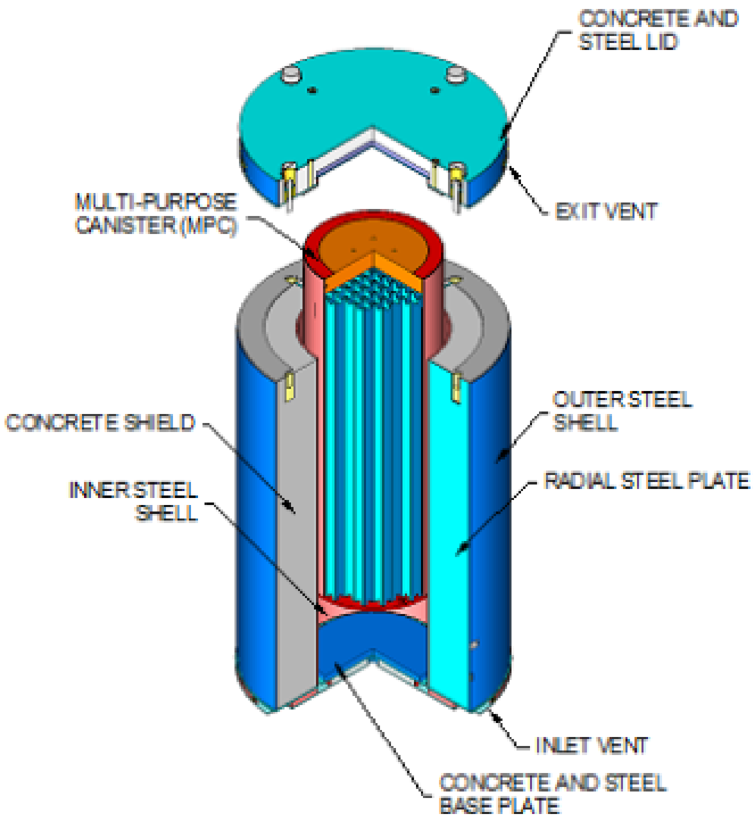
A typical DPC includes a cylindrical outer shell around a fuel basket that has compartments/cells for spent fuel assemblies (or, in some cases, damaged fuel) (Figure A-7). Among the common DPC designs, all use stainless steel for the canister shell and many (but not all) use stainless steel for the shield plugs, and top and bottom containment and structural lids (SNL 2020, Section 1). DPC baskets are constructed from various materials including stainless steels, carbon steel, and aluminum. The baskets inside the DPCs are designed to promote heat transfer from the assemblies to the canister outer shell and to control nuclear reactivity in accordance with regulatory requirements for storage and transportation. Some older systems used flux traps, but these take up space and these baskets are no longer produced. To achieve greater loading density, neutron absorbing materials (typically aluminum-boron carbide composites) are now built into the basket cells that hold the fuel assemblies (e.g., Boral™), or the entire basket is made from a material that is both structural and neutron absorbing (e.g., Metamic™) (SNL 2020, Sections 1 and 3.2).

An example of a storage cask/overpack for a DPC is shown in Figure A-8. Inlet vents at the bottom of the cask and outlet vents at the top facilitate the natural upward circulation of air through the annular space between the cask and the canister, provide passive cooling during the storage period.



(Source: Carter et al. 2016b)

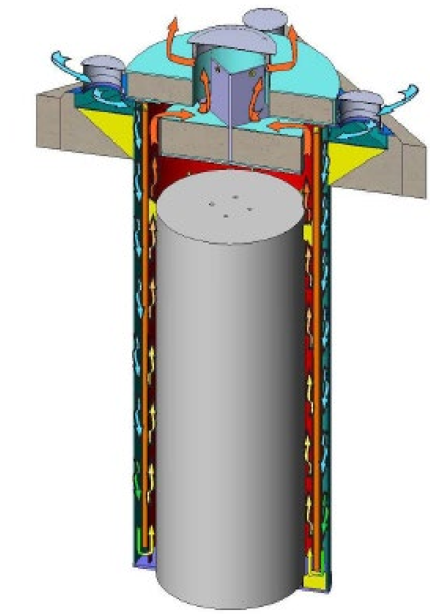
Figure A-7. Schematic of Storage Canister / DPC



(Source: Greene et al. 2013)

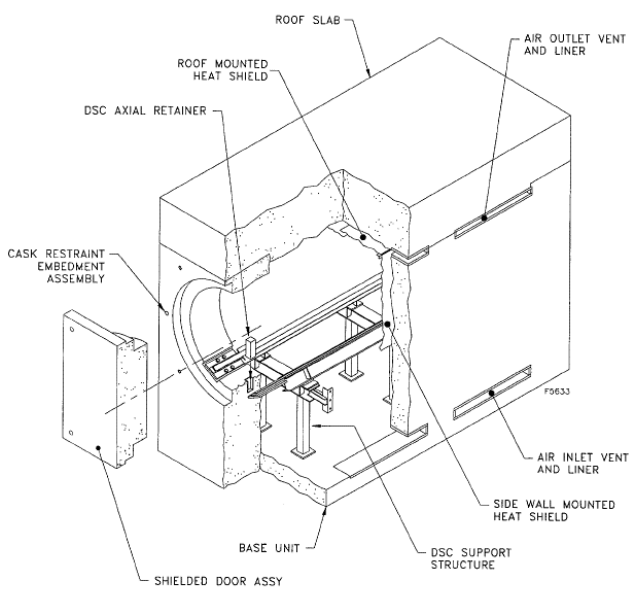
Figure A-8. HI-STORM 100 Above Ground Storage Cask System

Storage cask designs for canister-based systems include vertical above ground cask/canister systems on a concrete pad (Figure A-8), vertical cask/canister systems emplaced below grade (Figure A-9), and horizontal systems with canisters emplaced in modular concrete storage units or "vaults" (Figure A-10). A dry cask storage facility includes a number of cask/canister units (see Figure 3-3).



(Source: Greene et al. 2013)

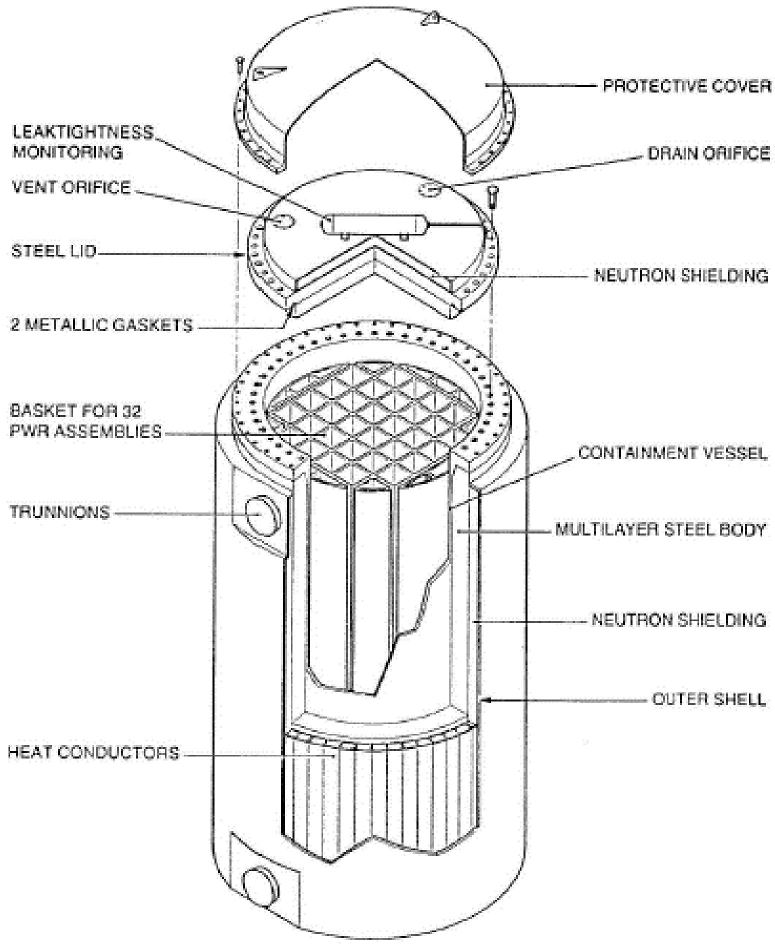
Figure A-9. HI-STORM UMAX Below Ground Storage Cask System



(Source: Greene et al. 2013)

Figure A-10. NUHOMS Horizontal Storage Module

In addition to canister-based systems, there are also bare fuel, or direct load, casks, in which individual spent fuel assemblies are loaded directly into a basket that is integrated into the cask (NRC 2014a, Section 2.1.2.1). Bare fuel casks, which tend to be all metal construction, are self-shielded and bolted closed (Figure A-11).



(Source: Greene et al. 2013)

Figure A-11. TN-32 Bare Fuel Cask

Loading of spent fuel assemblies into a DPC (or bare fuel cask) occurs inside a fuel pool, where the DPC has first been placed in a transfer cask. Once loaded, the canister and transfer cask are removed from the pool and the water is drained enough to weld the top onto the canister (Hanson and Alsaed 2019, Section 2.1). The system is then drained, decontaminated, and dried. Most systems use vacuum drying in which the decay heat of the fuel is used to help drive off water; other systems use a flow of dry helium to remove residual water (Hanson and Alsaed 2019, Section 2.1; NRC 2010). When vacuum drying is complete, the canister is backfilled with helium and is moved in the transfer cask to the storage location, where it is transferred to a stationary storage overpack.

The vacuum drying process often produces the highest cladding temperatures experienced during the dry storage process, and NRC guidance limits the peak clad temperature to 400°C under normal conditions (NRC 2010, Sections 4.4.2 and 8.4.17) to meet the regulations in 10 CFR Part 72.122 (Hanson and Alsaed 2019, Section 2.1).

Loading of bare fuel casks also occurs in a pool, but a transfer cask is not required. After cask sealing, water is removed through a drain tube, the outer surfaces are decontaminated, and the cask is transferred to the dry storage location (Hanson and Alsaed 2019, Section 2.1).

Multiple vendors provide NRC-certified DCSSs to utilities. DCSSs in use at the end of 2019 (as of January 1, 2020) are listed in Table A-1 (canistered systems) and Table A-2 (bare fuel cask systems). There are 37 unique canister types (canister families) in 16 unique canister-based cask systems and 7 unique bare fuel cask systems. At the end of 2019 there were a total of 3,203 DCSSs in dry storage, including 2,974 (93%) in canistered systems (2,958 with CSNF and 16 with GTCC waste) and 229 (7%) in bare fuel casks.

Table A-1. Canistered Dry Storage Systems

Storage Cask System	Cask Type [a]	Canister Family	Fuel Type	Canisters Loaded as of 1/1/2020		Transportation Cask
				CSNF	GTCC	
Holtec						
HI-STAR 100	Above	MPC-68 (HI-STAR)	BWR	7		HI-STAR 100
HI-STAR 100HB	Above	MPC-80HB	BWR	5	1	HI-STAR 100HB
HI-STORM	Above	MPC-24	PWR	36		HI-STAR 100
		MPC-32	PWR	487		HI-STAR 100
		MPC-68 (HI-STORM)	BWR	458		HI-STAR 100
		MPC-68M	BWR	134		Not Available
HI-STORM FW	Above	MPC-37[b]	PWR	57		HI-STAR 190
		MPC-89	BWR	44		HI-STAR 190
HI-STORM UMAX	Below	MPC-37[a]	PWR	62		HI-STAR 190
HI-STORM TranStor	Above	MPC-24E	PWR	29		HI-STAR 100
		MPC-24EF	PWR	5		HI-STAR 100
6		**10**		**1324**	**1**	**3 (for 9 canisters)**
NAC						
NAC-MPC	Above	MPC-26	PWR	40	3	NAC-STC
		MPC-36	PWR	15	1	NAC-STC
		LACBWR	BWR	5		NAC-STC
NAC-UMS	Above	UMS-24	PWR	264	4	UTC
MAGNASTOR	Above	TSC-37	PWR	131	4	MAGNATRAN
3		**5**		**455**	**12**	**3 (for 5 canisters)**
Orano TN						
Standard NUHOMS	Horiz.	24P	PWR	135		Not Available
		24PHB	PWR	64		Not Available
		24PTH	PWR	47		MP197HB
		32P	PWR	30		Not Available
		32PT	PWR	128		MP197HB
		37PTH[b]	PWR	10		MP197HB
		52B	BWR	27		Not Available

Storage Cask System	Cask Type [a]	Canister Family	Fuel Type	Canisters Loaded as of 1/1/2020		Transportation Cask
				CSNF	GTCC	
		61BT	BWR	129		MP197
		61BTH	BWR	200		MP197HB
		FC-DSC	PWR	18	1	MP187
		FF-DSC	PWR	1		MP187
		FO-DSC	PWR	2		MP187
Advanced NUHOMS	Horiz.	24PT1	PWR	17	1	MP187
		24PT4	PWR	33		MP197HB
NUHOMS HD	Horiz.	32PHB	PWR	16		Not Available
		32PTH / PTH1	PWR	202		MP197HB
		37PTH[b]	PWR	8		MP197HB
NUHOMS 0708		07P	PWR	8		Not Available
NUHOMS 12T		12T	ISFSI[c]	29		Not Available
5		**18**		**1114**	**2**	**3 (for 11 canisters)**
Westinghouse (EnergySolutions)						
FuelSolutions	Above	W74	BWR	7	1	TS125
VSC-24	Above	MSB-Long	PWR	24		Not Available
		MSB-Short	PWR	16		Not Available
		MSB-Standard	PWR	18		Not Available
2		**4**		**65**	**1**	**1 (for 1 canister)**
16		**37**		**2958**	**16**	**10 (for 26 canisters)**

[a] Above = vertical above ground
Below = vertical below ground
Horiz. = horizontal above ground
[b] Same canister is used in two different storage cask systems
[c] 29 canisters of TMI-2 core debris that was transferred to DOE and is now stored at INL (Table 2-3)

(Source: StoreFUEL 2020, Table 15; Jones 2016, Tables 4.3-1 and 4.6.-1)

Table A-2. Non-Canistered Dry Storage Systems

Vendor	Cask System	Cask Type [a]	Fuel Type	Canisters Loaded as of 1/1/2020		Transportation Cask
				CSNF	GTCC	
NAC	NAC-I28	Above	PWR	2		Not Available
Orano TN	TN-32	Above	PWR	64		Not Available
Orano TN	TN-40	Above	PWR	44		TN-40 [b]
Orano TN	TN-68	Above	BWR	92		TN-68
GNS [c]	CASTOR V/21	Above	PWR	25		Not Available
GNS [c]	CASTOR X/33	Above	PWR	1		Not Available
Westinghouse	MC-10	Above	PWR	1		Not Available
	7			**229**	**0**	**2**

[a] Above = vertical above ground
[b] There are two versions of the TN-40 cask (TN-40 and TN-40HT). Only the TN-40 is licensed for transport (Greene et al. 2013).
[c] Gesellschaft für Nuklear-Service mbH

(Source: StoreFUEL 2020, Table 15; Jones 2016, Tables 4.1-1)

Of the 16 unique canister-based cask systems, 3 account for almost 75% of the 2,974 canisters in storage:

- Holtec HI-STORM 100 (1115 canisters) – vertical, above-ground cask system for 24 and 32 PWR canisters and 68 BWR canisters
- Orano Transnuclear (TN) Standardized NUHOMS (792 canisters) – horizontal, above-ground cask system for 24, 32, and 37 PWR canisters and 52 and 61 BWR canisters
- NAC International NAC-UMS (264 canisters) – vertical, above-ground cask system for 24 PWR canisters

As noted previously, the term DPC is used in this report to generally refer to all multi-assembly canisters. However, the first generation dry-storage canisters were single-purpose canisters and were not licensed for CSNF transportation (Rechard et al. 2015, Section 2.1.2.2). Most of the canisters in use today are "true" DPCs that can be used for both storage (inside a storage cask) and transportation (inside a transportation cask or overpack) of CSNF. As shown in Table A-1, 26 of the 37 unique canister types are NRC-approved (i.e., have a certificate of compliance (CoC)) for storage (10 CFR Part 72.212) and transportation (10 CFR Part 71.17), which can be renewed in 40-yr increments). This accounts for 2,531 (85%) of the 2,974 canisters currently (at the end of 2019) in dry storage. Of the 7 unique bare fuel cask systems, 2 are certified for storage and transportation, accounting for 136 (59%) of the 229 bare fuel casks (Table A-2).

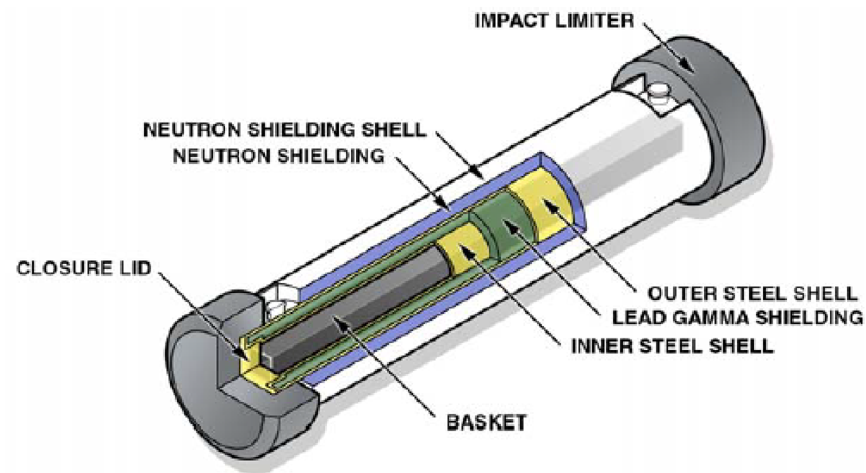
Individual DPC specifications vary, but the most commonly used ones range from about 181.0 in (4.6 m) to 196.3 in (5.0 m) in length, about 67.2 in (1.7 m) to 75.5 in (1.9 m) in cross section (diameter), and have a loaded weight of about 70,800 lbs (32.1 metric tons) to 116,400 lbs (52.8 metric tons) (Greene et al. 2013, Table 2). Design capacities of DPCs have increased over time, made possible by advanced cask/canister system designs that are very efficient in removing heat from the canister. Currently the largest DPCs hold 37 PWR assemblies or 89 BWR assemblies (Table A-1).

The most commonly used storage casks (exclusive of the horizontal cask systems) range from about 225.3 in (5.7 m) to 231.25 in (5.9 m) in length (height) and about 132.5 (3.4 m) to 136.0 in (3.5 m) in cross section (diameter). They weigh from about 219,000 lbs (99.3 metric tons) to 270,000 lbs (122.5 metric tons) empty and 310,400 lbs (140.8 metric tons) to 360,000 lbs (163.3 metric tons) loaded (Greene et al. 2013).

A.2.4. Transportation Casks

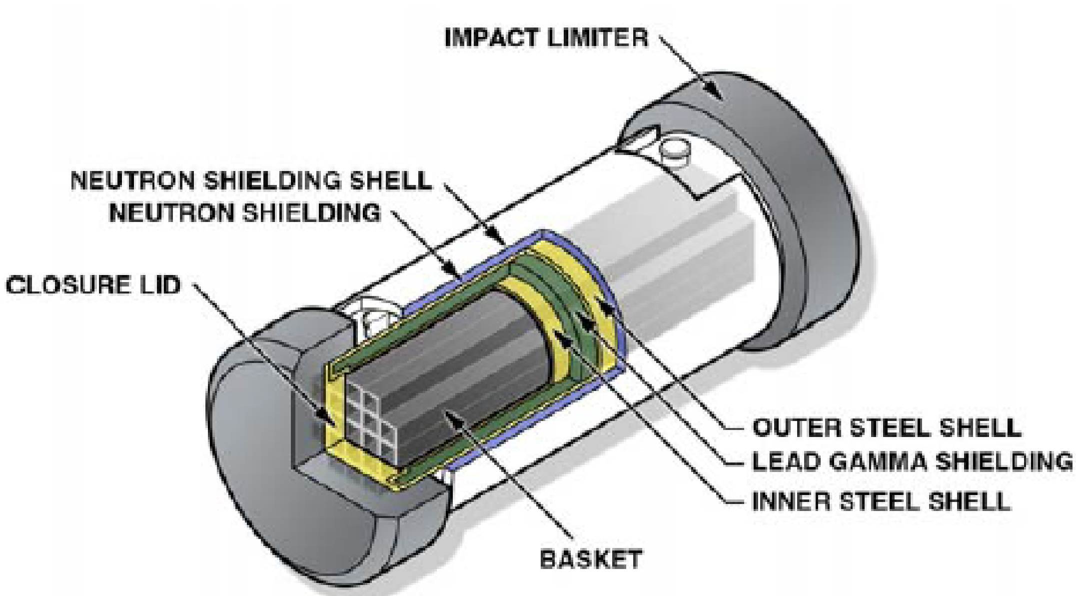
Canister-based systems that are considered dual-purpose usually have different overpacks for loading, storage, and transportation (Table A-1). Transportation of SNF requires removal of the canister (e.g., DPC) from the storage cask using a transfer cask and emplacement in a reusable transportation cask. Transportation casks provide structural protection and containment, radiation shielding, and thermal management.

As noted in Section A.2.3, 26 of the 37 unique canister types are NRC-certified for transportation (per 10 CFR Part 71), with 10 different transportation casks available (Table A-1). Two of the 7 unique bare fuel cask systems are certified for transportation (Table A-2). These transportation casks can accommodate about 85% of the current CSNF inventory in dry storage. Generic transportation casks are shown in Figure A-12 for truck transport and Figure A-13 for rail transport.



(Source: NAS 2006, Figure 2.1)

Figure A-12. Generic Truck Transportation Cask for CSNF



(Source: NAS 2006, Figure 2.1)

Figure A-13. Generic Rail Transportation Cask for CSNF

Truck casks dimensions range from 40 to 48 in (1.0 to 1.2 m) for cask diameter, from 65 to 90 in (1.7 to 2.3 m) for overall diameter including impact limiters, and from 232 to 247 in (5.9 to 6.3 m) for overall length including impact limiters (Greene et al. 2013). Truck casks can accommodate smaller canisters (from single assemblies up to 4 PWR assemblies) and have a loaded weight of about 50,000 lbs (~23 metric tons) (Greene et al. 2013, Table 1 and Table 6).

Rail cask dimensions range from 92 to 100 in (2.3 to 2.5 m) for cask diameter, from 122 to 128 in (3.1 to 3.3 m) for overall diameter including impact limiters, and from 272 to 376 in (6.9 to 9.6 m) for overall length including impact limiters (Greene et al. 2013). The largest rail casks can accommodate canisters with of up to 37 PWR assemblies or 89 BWR assemblies (Greene et al. 2013, Table 5). With loaded DPCs (the heaviest of which may weigh more than 50 metric tons), the most commonly used rail cask systems may weigh from about 250,000 lbs (113.4 metric tons) to 350,000 lbs (158.8 metric tons) (Greene et al. 2013, Table 1).

Due the large diameter and weight, transportation casks for DPCs are most likely to be transported by rail. For example, DOE selected a "mostly rail" scenario as the transportation mode to the proposed Yucca Mountain repository (DOE 2009b, Section I.C). However, the US does not currently have the capability to ship a DPC by rail because (1) for most designs, either the transportation casks or the impact limiters have not been fabricated, and (2) the US must design and certify a rail car according to the Association of American Railroads (AAR) Standard S-2043 (Rechard et al. 2015, Section 2.2.1). DOE has begun preliminary work on rail car design and testing, but this effort will take several years to complete.

While a transportation campaign could begin using trucks and the small current fleet of licensed legal-weight truck (LWT) casks, a sizeable fleet of rail rolling stock will be needed to move larger quantities of CSNF (such as those currently loaded in DPCs in dry storage at reactor sites) (BRC 2012, Section 9.3).

A.2.5. Disposal Waste Packages/Overpacks

Industry began developing DPCs in the late 1980s; the federal government proposed standardized canisters for storage (aging), transportation, and disposal in 1994 (the (MPC) (see Section A.2.5.1 for details) and then again in 2006 (the TAD canister) (see Section A.2.5.2 for details). The Yucca Mountain Repository License Application design (DOE 2009a) was based on the assumption that CSNF would be loaded into TAD canisters that hold 21 PWR or 44 BWR spent fuel assemblies. More recently, DOE revisited the issue, conducting feasibility studies on standardized transportation, aging, and disposal (STAD) canisters that included small (4 PWR/9 BWR), medium (12 PWR/32 BWR, and large (21 PWR/44 BWR) designs (see Section A.2.5.3 for details). None of these proposed standardized canister designs (MPC, TAD, or STAD) have been implemented.

The DSNF inventory is much more heterogeneous than CSNF, consisting of various assemblies, pieces, and debris from many different reactor designs. A large portion of DSNF (from Hanford N Reactor) is already loaded into small-diameter stainless-steel MCOs (Section 2.3.1), the remainder is planned to be loaded into small-diameter stainless-steel standard DSNF canisters, which would have four different sizes (two different lengths and two different diameters (Table A-3).

Naval SNF is being loaded into large-diameter canisters; there are two versions, a naval short SNF canister and a naval long SNF canister (Table A-3). Approximately 400 naval SNF canisters (310 long and 90 short) are currently planned to be packaged to accommodate ~65 MTHM (Peters et al. 2020, Section 3.2.1).

Most of the DHLW has been, or is projected to be, vitrified. Vitrified waste is typically poured directly into small diameter stainless-steel canisters where it cools into a glass "log" inside the canister. Hanford HLW canisters are planned to have a length of ~4.6 m, other HLW canisters have a length of ~3.0 m (Table A-3). Processed calcine HLW and processed sodium-bonded SNF are also planned to be loaded into standard small-diameter HLW canisters.

Table A-3. Relative Sizes of Proposed Storage, Transportation, and Disposal Canisters

Canister Type	Canister Capacity	Canister Dimensions				Canister Loaded Weight (MT)	Thermal Capacity (kW) Storage/Transport
		Diameter		Length			
		(in)	(m)	(in)	(m)		
DPC [a]	32-PWR / 68-BWR		~1.7-1.9		~4.5-5.0	25-53	~23-47 / ~16-40
TAD [b]	21-PWR / 44-BWR	66.5	1.69	212	5.38	49.3	18.0 / 11.8 [g]
STAD [c]	21-PWR / 44-BWR	66.25	1.68	198	5.03	TBD	TBD
	12-PWR / 32-BWR	52	1.32	194	4.93	21.1 / 22.7	TBD
	4-PWR / 9-BWR	29	0.74	196	4.98	6.0 / 6.3	8 / 6
Naval SNF (Long) [d]		66	1.68	210.5	5.35		
Naval SNF (Short) [d]		66	1.68	185.5	4.71		
Hanford N-Reactor MCO [e]		24	0.61	166.4	4.23		
Standard DSNF Canister (Short) [e]		18, 24	0.46, 0.61	120	3.05		
Standard DSNF Canister (Long) [e]		18, 24	0.46, 0.61	180	4.57		
Hanford HLW (Long) Canister [f]		24	0.61	180	4.57		
SRS HLW (Short) Canister [f]		24	0.61	118	3.00		
INL HLW (Short) Canister [f]		24	0.61	120	3.05		
West Valley HLW (Short) Canister [f]		24	0.61	118	3.00		

Sources:
[a] DPCs come in many capacities and dimensions. The values listed here represent some of the more common designs (Section A.2.3; Rechard et al. 2015, Section 2.1.2.2; Greene et al. 2013, Table 2)
[b] DOE (2009a, Section 1.5.1.1.1.2 and Figure 1.5.1-5)
[c] Carter et al. (2016b); Energy Solutions et al. (2015a) – small; Energy Solutions et al. (2013) – medium; Areva (2013) – large.
[d] DOE (2009a, Section 1.5.1.4.1.2 and Figure 1.5.1-29)
[e] DOE (2009a, Section 1.5.1.3.1.2, Figure 1.5.1-9 and Figure 1.5.1-18)
[f] DOE (2009a, Section 1.5.1.2.1.2, Figure 1.5.1-8, and Table 1.5.1-16)
[g] for disposal (at emplacement / at closure)

Disposal of each of these waste forms and canisters requires a disposal overpack (waste package). For the Yucca Mountain Repository License Application, three waste package types, with six configurations, were designed for the commercial and DOE-managed waste (DOE 2009a, Section 1.5.2.1). The different waste package types (CSNF, naval, and codisposal) and configurations would have had multiple internal structures and different external dimensions to allow acceptance of the various waste forms and canisters, specifically CSNF in TAD waste packages, naval SNF in naval waste packages, and standard DSNF canisters, HLW canisters, and MCOs in codisposal waste packages (Figure 3-6). The three waste package types and six configurations are summarized in Table A-4.

Table A-4. Yucca Mountain Waste Packages

Waste Package Type	Waste Package Configuration	Canister(s)	Waste Package Dimensions				Waste Package Loaded Weight (MT)	Number of Waste Packages [a]	Number of Waste Packages [b]
			Diameter		Length				
			(in.)	(m)	(in.)	(m)			
CSNF	21-PWR/44-BWR TAD	1 CSNF TAD	77.28	1.96	230.32	5.85	73.6	7,483	7,796 [c]
Co-disposal	5-DHLW / DOE Short	5 DHLW Short 1 Std DSNF Short	83.70	2.13	145.57	3.70	40.9	1,207	1,257
	5-DHLW / DOE Long	5 DHLW Long 1 Std DSNF Long	83.70	2.13	208.82	5.30	58.1	1,862	1,940
	2-MCO / 2-DHLW	2 DHLW Long 2 MCO (DSNF)	72.07	1.83	207.82	5.28	51.1	210	219
Naval	Naval Short	1 Naval Short	77.28	1.96	205.32	5.22	71.4	90	94
	Naval Long	1 Naval Long	77.28	1.96	230.32	5.85	73.6	310	323
Total								11,162	11,629

[a] Estimated for the design basis inventory (DOE 2009a, Table 2.3.4-27) supporting the Yucca Mountain Repository License Application.
[b] Estimated for the postclosure TSPA calculations (SNL 2010, Table 6.3.7-1) supporting the Yucca Mountain Repository License Application. These numbers of waste packages are ~4% higher than would be necessary for 70,000 MTHM. This is to completely fill all of the available drifts with waste. For example, this assumption implies a total of 417 naval waste packages instead of the design basis estimate of 400.
[c] Includes 4,586 21-PWR TADs, 3,037 44-BWR TADs, and 173 12-PWR TADs (SNL 2010, Table 6.3.7-1).

(Source: DOE 2009a, Tables 1.5.2-1, 1.5.2-3, 1.5.2-5, and 2.3.4-27; SNL 2010, Table 6.3.7-1)

Each type of waste package has a similar design, consisting of two concentric cylinders in which the waste canisters are placed (DOE 2009a, General Information, Section 1.1.3.1). The inner cylinder is stainless steel and includes welded top and bottom inner lids. The outer cylinder is an Alloy 22 (a corrosion resistant, nickel-based alloy) corrosion barrier and likewise includes welded top and bottom outer lids. The CSNF and naval waste packages contain a single canister. The codisposal waste packages contain combinations of DSNF and DHLW canisters. For example, a 5-DHLW/DOE Short waste package contains 1 standard DSNF (short) canister surrounded by 5 standard DHLW (short) canisters.

The Yucca Mountain Repository License Application design did not include disposal of CSNF in DPCs. Rather, CSNF would be repackaged into TAD canisters at the repository for disposal (DOE 2009a, Section 1.5.1.1). Ongoing R&D is examining the feasibility of disposing CSNF in DPCs (Sections 5.1.2 and C.3), with a corrosion-resistant disposal overpack that could be of similar design to Yucca Mountain CSNF waste package, as an alternative to repackaging the CSNF into TAD or other disposal-specific canisters (SNL 2021a). It is also possible that specialized waste packages could be needed for spent fuel from alternative/advanced fuel cycles.

A.2.5.1. Multi-Purpose Canister (MPC)

In 1994, DOE formally adopted in-drift emplacement (rather than vertical floor emplacement) for waste disposal (Rechard and Voegele 2014, Section 2.4.1). This repository design decision allowed for larger disposal packages that reduced construction costs, reduced handling operations, and facilitated a design goal of processing and disposing of CSNF at a rate of 3,000 MTHM/yr (Rechard et al. 2015, Section 3.4.3). The decision also enabled consideration of a standardized MPC for storage, transportation, and disposal.

MPCs were designed for both CSNF and HLW. The CSNF MPC was designed to hold either 21 PWR assemblies or 40 BWR assemblies, with a diameter of 1.8 m, a length of 5.7 m, and a loaded weight of ~60 metric tons; the DHLW MPC was designed to hold 4 HLW pour canisters (diameter = 0.61 m, length = 3.0 m), with an overall diameter of 1.7 m, length of 3.7 m, and loaded weight of ~35 metric tons (Rechard and Voegele 2014, Section 2.4.2 and Table 3). An MPC was to be loaded and welded shut at the reactors and placed in an appropriate overpack for storage, in another overpack for transportation, and in a corrosion-resistant overpack (waste package) for disposal (Rechard et al. 2015, Section 3.4.4). However, Congress eliminated funding for MPCs in 1996.

A.2.5.2. Transportation, Aging, and Disposal (TAD) Canister

In 2006, DOE again decided to plan for the use of a standardized canister as part of the Yucca Mountain Repository License Application. Similar to the MPC concept, the stainless-steel TAD canister was designed to be loaded with CSNF and sealed at the reactor and be used for transportation, aging (storage), and disposal. The TAD canister was designed to hold either 21 PWR assemblies or 44 BWR assemblies, with a nominal diameter of 66.5 in. (1.69 m), a maximum length of 212 in. (5.38 m), and a maximum loaded weight of 54.25 tons (49.3 metric tons) (DOE 2009a, Section 1.5.1.1.1). The TAD canister was designed to be certified for transportation under 10 CFR Part 71 (with a transportation cask/overpack), to be certified for storage under 10 CFR Part 72 (with a storage cask/overpack), and to meet the preclosure and postclosure disposal requirements of 10 CFR Part 63 (in a waste package overpack) (DOE, 2009, Section 1.5.1.1.1). TAD canisters were designed for storage, transportation, and disposal of CSNF only; codisposal waste packages for DSNF and HLW canisters, of similar size to TAD-based waste packages, were also planned (Table A-4).

The large TAD canister was consistent with the desire of utilities to use large containers for storage and transportation of CSNF. Because the Yucca Mountain repository was halted, the TAD canister was not built. Although large at 21 PWR assemblies, the TAD canister would have been smaller than the common 32-PWR and newer 37-PWR assembly DPCs currently used at utilities and, thus, not as cost-effective for the utilities (Rechard et al. 2015, Section 3.4.5).

A.2.5.3. Standardized Transportation, Aging, and Disposal (STAD) Canister

Most recently, starting in 2013, DOE-NE NFST (now IWM) conducted feasibility studies on STAD canisters for CSNF (Areva 2013; Energy Solutions et al. 2013; Energy Solutions et al. 2015a; Energy Solutions et al. 2015b; Jarrell et al. 2015a; Jarrell et al. 2015b; ORNL 2015a; ORNL 2015b).

Proposed STAD canister designs included small (4 PWR/9 BWR with diameter = 0.74 m and length = 4.98 m), medium (12 PWR/32 BWR with diameter = 1.32 m and length = 4.93 m), and large (21 PWR/44 BWR with diameter = 1.68 m and length = 5.03 m) (Carter et al. 2016b). STAD canisters were never implemented.

A.2.5.4. Rod Consolidation

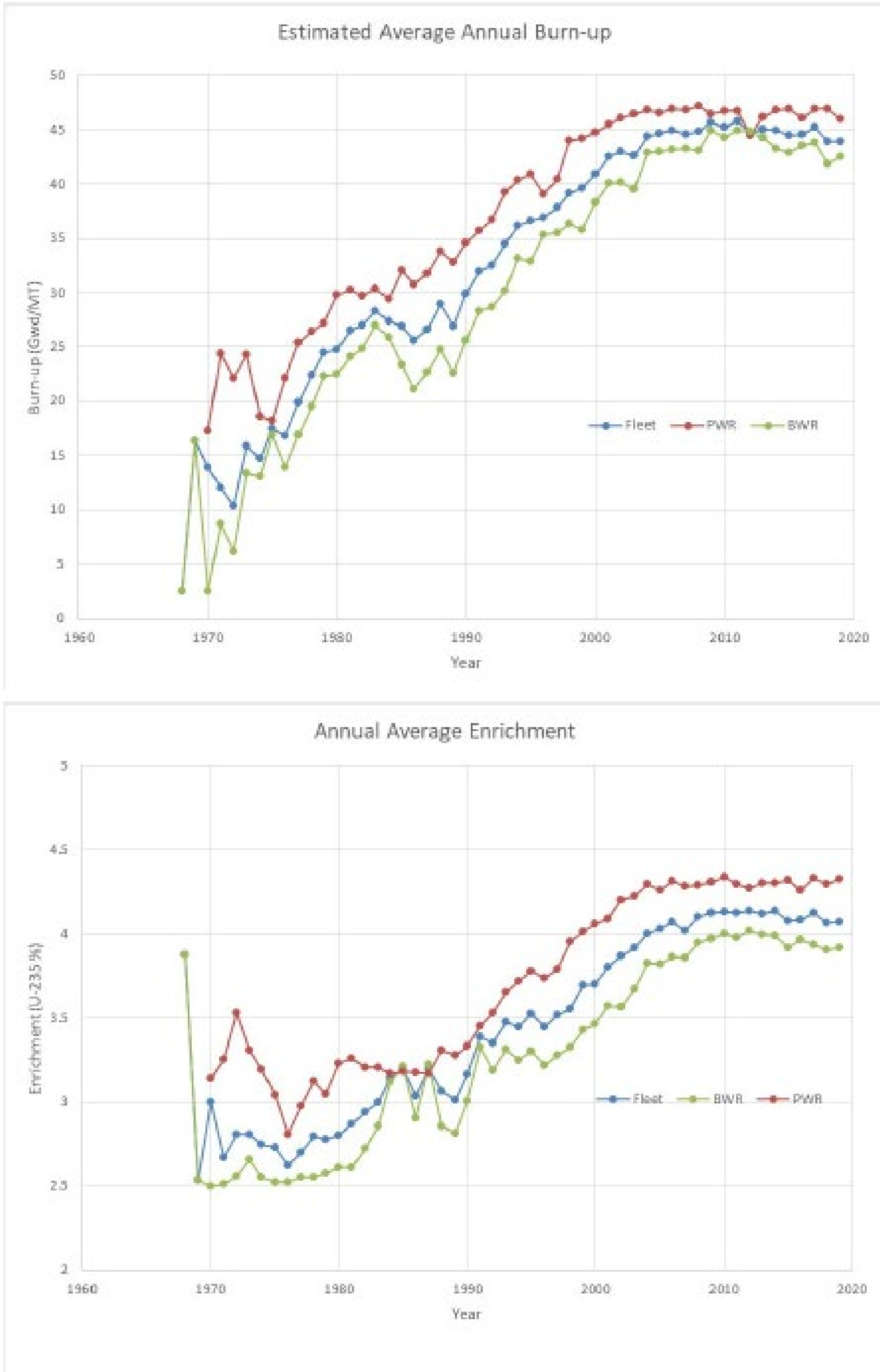
Rod consolidation involves reducing the volume of spent fuel assemblies by mechanically removing the spent fuel rods from the fuel assembly hardware and placing them in either another grid with closer spacing or in a close-packed array (Klein et al. 1986; Sattler 1993). It was practiced at several different sites in the 1980's to utilize existing space in spent fuel storage pools more efficiently, and the process achieved rod consolidation ratios ranging from 1.6:1 to 2:1. (Sattler 1993, Table 7). However, because of the delay in opening a repository, rod consolidation and other space-saving techniques (e.g., re-racking) only postponed the time at which spent fuel storage pools became full. To address the need for additional storage of spent fuel, licensed dry storage systems (which do not involve rod consolidation) were developed and implemented, and continue to be used today (see Section A.2.3).

If rod consolidation were to be pursued moving forward, it would most likely be in the context of improving dry storage or disposal efficiencies. Although most of the previous rod consolidation efforts were "wet" (i.e., took place in pools), some were dry (Bailey 1989, Table 2.2). In addition, the American Nuclear Society (ANS) has a standard that provides design criteria for the process of consolidating LWR spent fuel in either a wet or a dry environment (ANS 1996).

A.3. Characteristics of Commercial SNF

At any one time, the core of a typical LWR in the US contains approximately 100 MTU (BRC 2012, Chapter 3.1). In currently operating reactors (end of 2019), PWRs most commonly contain between 157 and 241 fuel assemblies and BWRs most commonly contain between 560 and 800 fuel assemblies (StoreFUEL 2021, Table 13). Commercial LWR cycle lengths range from 18 to 24 months (StoreFUEL 2021, Table 12); outages between cycles are used to remove some of the spent fuel assemblies and replace them with fresh fuel. A specific fuel assembly typically remains in a LWR for about four to six years. Fuel is removed not because fissile material is fully used-up, but because the neutron-absorbing fission products have built up and the fuel becomes significantly less able to sustain a nuclear reaction.

The amount of SNF accumulated at a reactor over its licensed life depends on factors such as how long the reactor operates each year, the duration of outages, spent fuel burnup, and operating lifetime. Average fuel burnups have increased from around 35 GWd/MTU two decades ago to over 45 GWd/MTU today (Figure A-14) and are expected to eventually exceed 60 GWd/MTU (NRC 2014a, Section 2.8 and Appendix I).



(Source: Peters et al. 2020, Figure 2-5)

Figure A-14. PWR Nuclear Fuel Assembly

A.3.1. Burnup

Spent fuel burnup describes the extent to which energy has been extracted from nuclear fuel and is one factor in how often a reactor's fuel needs to be replaced. Burnup is typically expressed in units of gigawatt-days per metric ton of the initial amount of the uranium in the fuel (GWd/MTU).

It is important to note that the gross amount of fission products generated to produce a given amount of electricity is independent of fuel burnup because the energy yield of fission is always 1000 $GWd/MTHM_{fissioned}$. Therefore, in the case of higher-burnup fuel, the same amount of material must be fissioned and the same quantity of fission products is generated, but the fission products are simply concentrated in a smaller mass of fuel (MIT 2003, Appendix Chapter 4).

As a result, high-burnup fuel, typically defined as fuel with a burnup greater than 45 GWd/MTU, is thermally hotter and more radioactive than low-burnup fuel for a given cooling time (NRC 2014a, Section 2.8 and Appendix I). Therefore, high-burnup fuel may need to be cooled longer than low-burnup fuel before it can be placed into a dry cask storage system. How much longer depends on the difference in burnup, the specific dry cask storage system design, and the decay heat loading pattern of the fuel being used. For example, for a 5% 235U enriched fuel assembly in one particular storage system, the required cooling time goes from 4.5, to 7, to 12 years, for fuel burnups of 35, 45, and 55 GWd/MTU, respectively (NRC 2014a, Appendix I).

High-burnup fuel also results in a reduction in SNF volume (same quantity of fission products, but less uranium) and fissile ^{239}Pu per unit of electricity generated as compared to low-burnup fuel. However, the reduction in SNF volume does not necessarily translate to a smaller repository for disposal; the individual spent fuel assemblies, although there would be fewer of them, would generate more decay heat and would therefore have to be spaced farther apart in the repository (MIT 2003, Chapter 7). The reduction in fissile ^{239}Pu makes high-burnup fuel less suitable for weapons, supporting non-proliferation objectives (MIT 2003, Chapter 7).

At low burnups (less than 45 GWd/MTU), about one-fourth to one-third of the spent fuel assemblies are removed from the reactor and replaced every 12 to 18 months (NRC 2014a, Section 2.8). Therefore, the amount of spent fuel discharged from a typical LWR operating at low burnups is about 20 MTU per year (NRC 2014a, Section 2.8). After 80 years of reactor operation at low burnups, this amounts to about 1,600 MTHM of spent fuel. A reactor could operate for 80 years if the licensee requested, and the NRC granted, two 20-year renewals of its initial 40-year operating license.

For plants at which higher fuel burnups are authorized, the period between outages may be extended to 24 months and the annual discharge of spent fuel reduced to about 15 MTU per year (NRC 2014a, Section 2.8). Should a nuclear power plant operate for up to 80 years with high-burnup fuel, it would generate about 1,200 MTHM of spent fuel.

A.3.2. Alteration and Degradation

Nuclear fuels undergo significant changes during reactor operations. The fission process generates actinides and fission products (see Section A.3.3), the UO_2 fuel pellets undergo thermal expansion and swelling, and reactions between the cladding and cooling water leads to outer surface corrosion of the cladding and releases hydrogen (Hanson et al. 2012, Section 5). Specific considerations include the formation of a zirconium oxide layer on the cladding outer surface, formation of zirconium hydrides which can embrittle the cladding and reduce ductility, cladding creep from hoop stress caused by rod internal pressure, fission gas release (e.g., Xe, Kr), pellet-clad interactions, and the formation of a high-burnup structure at the surface of the fuel pellets (Hanson et al. 2012, Section 5; Hanson and Alsaed 2019, Section 1.2.6). These changes tend to increase as fuel burnup increases.

Fission gases are accommodated in the fuel-clad gap and in the plenum region of the fuel rods. The presence of fission gases outside a fuel rod is indicative of a failed rod.

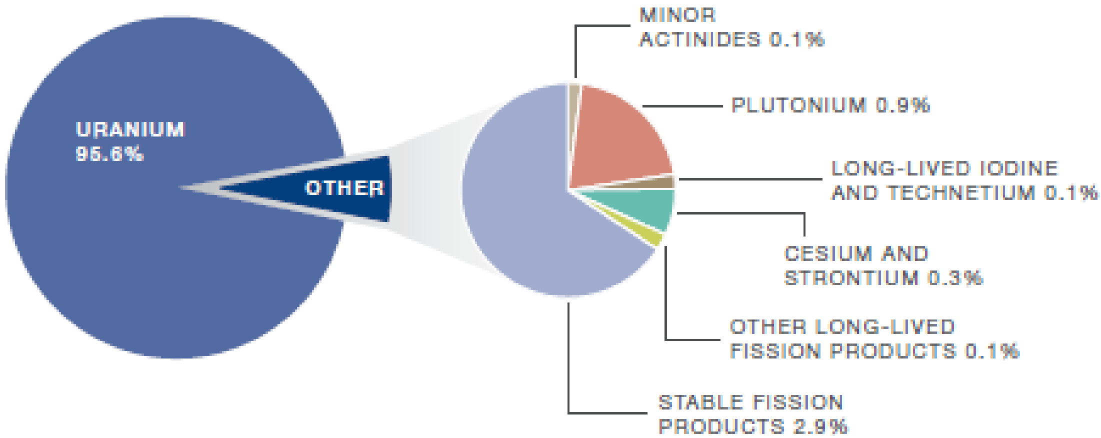
These alteration and degradation processes may affect the performance of the fuel, cladding, and assembly hardware during storage, transportation, and disposal and can be further exacerbated by temperature changes and/or the presence of water during dry storage. As noted in Section A.2.3, cladding temperatures during the storage period may be highest during the drying process associated with transferring spent fuel from pools to storage canisters. Drying procedures are designed to minimize the amount of water remaining in a canister (Hanson and Alsaed 2019, Section 5.3)

Additional details about these processes can be found in Hanson et al. (2012) and NRC (2020e).

A.3.3. Radionuclide Composition

Upon discharge from a reactor, SNF is still predominantly ^{238}U. Figure A-15 shows that SNF, 10 years out-of-reactor from a typical LWR, contains approximately (BRC 2012, Figure 6):

- 95.6% uranium (most of the original ^{238}U ~94.8% and a little remaining fissile ^{235}U ~0.8%; ratio depends on initial enrichment and burnup)
- 0.9% plutonium (^{239}Pu, ^{240}Pu, ^{241}Pu)
- 0.1% minor actinides (e.g., ^{237}Np, ^{241}Am, ^{243}Am, and various isotopes of Cm)
- 0.3% short-lived fission products (e.g., ^{90}Sr, ^{137}Cs)
- 0.1% long-lived fission products (e.g., ^{79}Se, ^{99}Tc, ^{129}I, ^{135}Cs)
- 3.0% stable fission products (e.g., gases - such as Xe and Kr, metals, oxide precipitates, solid solutions, transuranics (Bruno and Ewing 2006))
- trace amounts of neutron activation products (e.g., ^{14}C, ^{36}Cl, ^{60}Co, ^{63}Ni)



(Source: BRC 2012, Figure 6)

Figure A-15. Composition of Spent Nuclear Fuel by Mass After 10 Years of Cooling

One year after discharge from a reactor, the dose rate measured one meter from the fuel assembly is significantly higher than the natural background dose. A person exposed to this level of radioactivity at a distance of one meter would receive a lethal dose in less than a minute; hence spent fuel must be handled remotely (Bruno and Ewing 2006). The very penetrating ionizing radiation (beta (β) and gamma (γ)) comes mainly from short-lived fission products, such as ^{137}Cs and ^{90}Sr, with half-lives of about 30 years (Bruno and Ewing 2006). These fission products are mainly responsible for the thermal decay heat from the fuel. The less penetrating radiation from α decay events comes mainly from the very long-lived actinides, such as ^{239}Pu and ^{237}Np (Bruno and Ewing 2006). Most of these α decays originate from one of four main decay chains, starting from ^{232}Th, ^{238}U, ^{235}U, or ^{237}Np.

The radionuclides of greatest environmental impact in a geologic repository will be those that have some combination of high dose conversion factor, geochemical mobility, and a long half-life. Examples are ^{99}Tc (half-life = 213,000 years), ^{135}Cs (2.3 million years), ^{239}Pu (24,100 years), ^{237}Np (2.14 million years), ^{235}U (704 million years) (Clayton et al. 2011, Table 3.1-1), ^{129}I (15.7 million years), and ^{79}Se (290,000 years) (Freeze et al. 2013, Appendix E-1).

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APPENDIX B. COMMERCIAL REACTORS (OPERATING AND SHUTDOWN) AND INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS

As of December 2020, 94 commercial nuclear reactors were licensed and operating in the US at 56 sites[19] in 28 states (Table B-1). These "counts" include the recent shutdowns of Pilgrim 1 in May 2019, Three Mile Island 1 in September 2019, Indian Point 2 in April 2020, and Duane Arnold in August 2020. These 94 NPPs all are LWRs; 63 are PWRs and 31 are BWRs. Of the 56 sites, 21 sites (11 PWR and 10 BWR) have one operating reactor, 32 sites (23 PWR and 9 BWR) have two operating reactors, and 3 sites (2 PWR and 1 BWR) have three operating reactors.

In addition, two new PWRs, Vogtle 3 and 4, are under construction at an existing site (NRC 2020a, Appendix B); construction of two other new reactors at an existing site, V.C. Summer 2 and 3, was initiated, but was terminated in July 2017 (NRC 2018). Six potential new reactors (Fermi 3, North Anna 3, and Turkey Point 6 and 7 at existing sites, and W.S. Lee 1 and 2 at a new site) have active licenses for construction by NRC (NRC 2020a, Appendix B). However, decisions on proceeding with the new construction are currently constrained by economics (e.g., lower costs for natural gas and coal plants, the lack of carbon emission incentives) (MIT 2011, Chapter 1).

Commercial nuclear reactor (class 103) operating licenses can be issued by NRC for a fixed period of time not to exceed 40 years (10 CFR Part 50.51(a) and 50.57(a)). Renewals (extensions) of operating licenses may be issued by NRC for a fixed period of time not to exceed 20 years; the total term of any renewed license (i.e., the time remaining in the current license plus the up-to-20-year extension) may not exceed 40 years (10 CFR Part 54.31(b)). A renewed license may be subsequently renewed (10 CFR Part 54.31(c)). As of December 2020, of the 94 operating reactors, 82 have received a first 20-year license renewal (all of which expire between 2025 and 2050), 4 (Peach Bottom 2 and 3, Turkey Point 3 and 4) have received a second 20-year license renewal (all expiring between 2052 and 2054), 4 (Comanche Peak 1 and 2, Perry, and Clinton) have announced intentions to submit renewal applications between 2022 and 2024 (NRC 2020j), 2 (Diablo Canyon 1 and 2) have plans to shut down before a renewal is needed, and the 2 newest reactors (Watts Bar 1 and 2) have yet to announce an intention (Table B-2). Also, 3 (Oconee 1, 2, and 3) have announced intention to submit a second 20-year license renewal (NRC 2020k).

As described in Section 3.1, significant storage of CSNF occurs at reactor sites, in the form of both wet (pool) storage and dry storage. CSNF in at-reactor wet storage is licensed by the NRC under 10 CFR Part 50 as part of the reactor license. CSNF in dry storage may be licensed under either a general license at a reactor site (10 CFR Part 72, Subpart K, specifically Sections 72.210 and 72.212) or a site-specific license at or away from a reactor site (10 CFR Part 72.16). Storage under 10 CFR Part 72 may be either in an Independent Spent Fuel Storage Installation (ISFSI) or a Monitored Retrievable Storage (MRS) facility, but all at-reactor dry storage facilities are currently ISFSIs. As summarized in NRC (2014a, Appendix G):

[19] Some publications count Indian Point 2 and Indian Point 3 as separate sites; in this report, they are considered to be two reactors at a single site. Other publications count Fitzpatrick and Nine Mile Point as a single site and Hope Creek and Salem as a single site, due to proximity; in this report, all are considered to be independent sites.

- "The 10 CFR Part 72 general license authorizes a nuclear power plant licensee to store spent fuel in casks approved by the NRC at a site licensed to operate a power reactor under 10 CFR Part 50 or Part 52. An NRC-approved cask is one that has undergone a technical review of its safety aspects and has been found to be adequate to store spent fuel at a site that meets all of the NRC's requirements in 10 CFR Part 72. A licensee is required to perform an evaluation of its site to demonstrate that the site is adequate for storing spent fuel in dry casks. This evaluation must show that the cask certificate of compliance conditions and technical specifications can be met and must include an analysis of earthquake events and tornado missiles. In addition, the licensee must review its security program, emergency plan, quality assurance program, training program, and radiation protection program, and make any changes necessary to incorporate the ISFSI at the reactor site. Requirements for the general license are described in Subpart K of 10 CFR Part 72."
- "Under a 10 CFR Part 72 site-specific license, an applicant submits a license application to the NRC. The NRC performs a technical review of all the safety aspects of the proposed ISFSI and an environmental review in compliance with the National Environmental Policy Act. If the application is approved, the NRC issues a license that is valid for up to 40 years. A spent fuel storage license contains technical requirements and operating conditions (i.e., fuel specifications, cask leak testing, surveillance, and other requirements) for the ISFSI and specifies what the licensee is authorized to store at the site. Requirements for the site-specific license are described in Subparts A through I of 10 CFR Part 72."

The Nuclear Energy Institute (NEI) estimates ~3 years for NRC approval of a general license because the ISFSI application can rely on documentation required for the reactor operating license in 10 CFR Part 50; a site-specific license can take ~6 years for NRC approval (Rechard et al. 2015, Section 2.1.2). An ISFSI may be licensed for up to 40 years with options to renew in up to 40-yr increments (10 CFR Part 72.42 and 72.240).

At the 56 sites with operating reactors, 47 have general licensed ISFSIs, 3 (Calvert Cliffs, Diablo Canyon, and Prairie Island) have site-specific licensed ISFSIs, 4 (H.B. Robinson, North Anna, Oconee, and Surry) have both general and site-specific licenses, and 2 (Shearon Harris and Wolf Creek) have not yet made a decision on licensed storage (Table B-2).

In addition to the 94 operating commercial reactors as of December 2020, there are also 37 shutdown reactors that were used for civilian power generation, 11 of which were for demonstration or prototype purposes (Table B-3). There are also 4 shutdown reactors from early experimental programs that were not used for central-station power generation (Table B-3). The shutdown commercial reactors are in various stages of decommissioning, in accordance with 10 CFR Part 50.82(a)(3), which also states that "decommissioning will be completed within 60 years of permanent cessation of operations". Decommissioning options include (NRC 2014a, Section 2.2.1.1):

- DECON: The equipment, structures, and portions of the facility and site that contain radioactive contaminants are removed or decontaminated to a level that permits termination of the license shortly after cessation of operations.
- SAFSTOR: The facility is placed in a safe, stable condition and maintained in that state until it is subsequently decontaminated and dismantled to levels that permit license termination.

During SAFSTOR, a facility is left intact, but the fuel has been removed from the reactor vessel and radioactive liquids have been drained from systems and components and then processed. Radioactive decay occurs during the SAFSTOR period, which reduces the levels of radioactivity in and on the material and, potentially, the quantity of material that must be disposed of during decontamination and dismantlement.

- ENTOMB: ENTOMB involves encasing radioactive structures, systems, and components within a structurally long-lived substance, such as concrete. The entombed structure is appropriately maintained, and continued surveillance is carried out until the radioactivity decays to a level that permits termination of the license. Because most power reactors will have radionuclides in concentrations exceeding the limits for unrestricted use even after 100 years, this option will generally not be feasible.

In general, these decommissioning options apply to the radioactive reactor and facility components; the SNF is typically removed to an on-site or off-site storage facility (e.g., an ISFSI). NRC (2014a) addresses decommissioning in the rulemaking for "Continued Storage of Spent Nuclear Fuel" (codified in 10 CFR Part 51.23), formerly known as the "Waste Confidence Decision". The new rule (NRC 2014c, p. 56241) finds "reasonable assurance that, if necessary, spent fuel generated in any reactor can be stored safely and without significant environmental impacts for at least 60 years beyond the licensed life for operation (which may include the term of a revised or renewed license) of that reactor in a combination of storage in its spent fuel storage basin and either onsite or offsite ISFSIs."

Of the 37 shutdown commercial non-experimental reactors (Table B-3), 12 have had some or all of the SNF removed from the site:

- 9 were early demonstration or prototype reactors (4 have DECON complete, 2 have reactors in SAFSTOR, and 3 have reactors in ENTOMB; some SNF is now DOE-managed) (NEI 2020),
- Shoreham was never operated,
- Three Mile Island 2 (TMI-2) has been decontaminated and DOE has taken title to the CSNF and core debris and moved it to the DOE TMI-2 ISFSI at INL, and,
- the Fort St. Vrain demonstration reactor has DECON complete; the SNF is now DOE-managed and stored partly in an on-site ISFSI and partly at INL.

The remaining 25 shutdown commercial reactors are in various stages of decommissioning (DECON or SAFSTOR) and have utility-owned CSNF on-site. Of these, 4 are on sites with actively operating reactors (Indian Point 1 and 2, Dresden 1, and Millstone 1). CSNF at the other 21 reactors (at 18 different sites) is "stranded" waste because the reactor is being, or has been, decommissioned and there is no longer a licensed operating reactor on-site.

Stranded CSNF at older "legacy" shutdown sites (all shut down by 1997 and have not had an operating reactor on-site for at least 20 years) includes 10 shutdown reactors at 9 sites: Humboldt Bay 3; LaCrosse; Rancho Seco; Yankee Rowe; Trojan; Haddam Neck; Maine Yankee; Zion 1 and 2; and Big Rock Point. Of the 9 sites, 6 have general licensed ISFSIs and 3 site-specific ISFSIs. At the end of 2019, these 9 sites had a projected total CSNF inventory of 2,815 MTHM, all in dry storage in a total of 248 casks (Peters et al. 2020, Table 2-5).

Stranded CSNF at the more recently (after 2010) shutdown sites includes 11 shutdown reactors at 9 sites: Crystal River; San Onofre 1, 2, and 3; Kewaunee, Vermont Yankee, Fort Calhoun, Oyster Creek, Pilgrim, Three Mile Island 1, and Duane Arnold. All 9 of these sites have general licensed ISFSIs. At the end of 2019, 8 of these sites (Duane Arnold is not included in these totals as it did not close until 2020) had a projected total inventory of 6,202 MTHM, in pool and dry storage (Peters et al. 2020, Table 2-5). This CSNF is in the process of being transferred to dry storage; 298 casks are currently loaded, and it is projected that 480 dry storage casks will eventually be required for all of the CSNF (Peters et al. 2020, Table 2-12). At the end of 2019, Duane Arnold had a projected inventory of 593 MTHM in pool storage (372 MT) and dry storage (221 MT in 20 casks) (Peters et al. 2020, Table B-1).

In addition to the 25 shutdown reactors with utility-owned CSNF on-site, 8 of the 94 operating reactors have also announced plans for shutdown: Indian Point 3 (in 2021), Byron 1 and 2 (in 2021), Dresden 2 and 3 (in 2021), Palisades (in 2022), and Diablo Canyon 1 and 2 (in 2024 and 2025) (StoreFUEL 2021, Table 10). These shutdowns would create stranded CSNF at 5 more sites.

Recently announced plans for shutdown of 8 other operating reactors have since been rescinded, often due to state-funded economic incentives. These include: James Fitzpatrick (2017); Clinton (2017); Quad Cities 1 and 2 (both 2018); Davis Besse (2020), Perry (2021), and Beaver Valley 1 and 2.

Table B-1. Operating Commercial Reactors in the US (End of 2020)

Reactor Name	State	Type	License Output (MWt)	Nameplate Capacity (MWe)	Date of Commercial Operations
Arkansas Nuclear One 1	AR	PWR	2568	903	12/19/1974
Arkansas Nuclear One 2	AR	PWR	3026	943	3/26/1980
Beaver Valley 1	PA	PWR	2900	923	10/1/1976
Beaver Valley 2	PA	PWR	2900	923	11/17/1987
Braidwood 1	IL	PWR	3645	1225	7/29/1988
Braidwood 2	IL	PWR	3645	1225	10/17/1988
Browns Ferry 1	AL	BWR	3952	1152	8/1/1974
Browns Ferry 2	AL	BWR	3952	1152	3/1/1975
Browns Ferry 3	AL	BWR	3952	1190	3/1/1977
Brunswick 1	NC	BWR	2923	1002	3/18/1977
Brunswick 2	NC	BWR	2923	1002	11/3/1975
Byron 1	IL	PWR	3645	1225	9/16/1985
Byron 2	IL	PWR	3645	1225	8/2/1987
Callaway	MO	PWR	3565	1236	12/19/1984
Calvert Cliffs 1	MD	PWR	2737	918	5/8/1975
Calvert Cliffs 2	MD	PWR	2737	932	4/1/1977
Catawba 1	SC	PWR	3469	1205	6/29/1985
Catawba 2	SC	PWR	3411	1205	8/19/1986
Clinton	IL	BWR	3473	1138	11/24/1987
Columbia Generating Station	WA	BWR	3544	1200	12/13/1984
Comanche Peak 1	TX	PWR	3612	1215	8/13/1990
Comanche Peak 2	TX	PWR	3612	1215	8/3/1993
Cooper	TX	BWR	2419	801	7/1/1974
Davis Besse	OH	PWR	2817	925	7/31/1978
Diablo Canyon 1	CA	PWR	3411	1159	5/7/1985
Diablo Canyon 2	CA	PWR	3411	1164	3/13/1986
Donald C. Cook 1	MI	PWR	3304	1152	8/28/1975
Donald C. Cook 2	MI	PWR	3468	1133	7/1/1978
Dresden 2	IL	BWR	2957	1009	6/9/1970
Dresden 3	IL	BWR	2957	1009	11/16/1971
Edwin I. Hatch 1	GA	BWR	2804	924	12/31/1975
Edwin I. Hatch 2	GA	BWR	2804	924	9/5/1979
Fermi 2	MI	BWR	3486	1217	1/23/1988
Grand Gulf 1	MS	BWR	4408	1440	7/1/1985
H.B. Robinson 2	SC	PWR	2339	769	3/7/1971
Hope Creek 1	NJ	BWR	3902	1291	12/20/1986
Indian Point 3	NY	PWR	3216	1012	8/30/1976
James A. Fitzpatrick	NY	BWR	2536	883	7/28/1975
Joseph M. Farley 1	AL	PWR	2775	888	12/1/1977
Joseph M. Farley 2	AL	PWR	2775	888	7/30/1981
La Salle 1	IL	BWR	3546	1170	1/1/1984
La Salle 2	IL	BWR	3546	1170	10/19/1984
Limerick 1	PA	BWR	3515	1139	2/1/1986
Limerick 2	PA	BWR	3515	1139	1/8/1990
McGuire 1	NC	PWR	3469	1220	12/1/1981
McGuire 2	NC	PWR	3469	1220	3/1/1984
Millstone 2	CT	PWR	2700	910	12/26/1975
Millstone 3	CT	PWR	3650	1253	4/23/1986
Monticello	MN	BWR	2004	685	6/30/1971

Reactor Name	State	Type	License Output (MWt)	Nameplate Capacity (MWe)	Date of Commercial Operations
Nine Mile Point 1	NY	BWR	1850	642	12/1/1969
Nine Mile Point 2	NY	BWR	3988	1259	3/11/1988
North Anna 1	VA	PWR	2940	980	6/6/1978
North Anna 2	VA	PWR	2940	980	12/14/1980
Oconee 1	SC	PWR	2568	887	7/15/1973
Oconee 2	SC	PWR	2568	887	9/9/1974
Oconee 3	SC	PWR	2568	893	12/16/1974
Palisades	MI	PWR	2565.4	812	12/31/1971
Palo Verde 1	AZ	PWR	3990	1403	1/28/1986
Palo Verde 2	AZ	PWR	3990	1403	9/19/1986
Palo Verde 3	AZ	PWR	3990	1403	1/8/1988
Peach Bottom 2	PA	BWR	4016	1499	7/5/1974
Peach Bottom 3	PA	BWR	4016	1377	12/23/1974
Perry 1	OH	BWR	3758	1312	11/18/1987
Point Beach 1	WI	PWR	1800	643	12/21/1970
Point Beach 2	WI	PWR	1800	643	10/1/1972
Prairie Island 1	MN	PWR	1677	593	12/16/1973
Prairie Island 2	MN	PWR	1677	593	12/21/1974
Quad Cities 1	IL	BWR	2957	1009	2/18/1973
Quad Cities 2	IL	BWR	2957	1009	3/10/1973
R.E. Ginna	NY	PWR	1775	614	7/1/1970
River Bend 1	LA	BWR	3091	1036	6/16/1986
Salem 1	NJ	PWR	3459	1170	6/30/1977
Salem 2	NJ	PWR	3459	1170	10/13/1981
Seabrook 1	NH	PWR	3648	1242	8/19/1990
Sequoyah 1	TN	PWR	3455	1221	7/1/1981
Sequoyah 2	TN	PWR	3455	1221	6/1/1982
Shearon Harris 1	NC	PWR	2948	951	5/2/1987
South Texas Project 1	TX	PWR	3853	1354	8/25/1988
South Texas Project 2	TX	PWR	3853	1354	6/19/1989
St. Lucie 1	FL	PWR	3020	1080	12/21/1976
St. Lucie 2	FL	PWR	3020	1080	8/8/1983
Surry 1	VA	PWR	2587	848	12/22/1972
Surry 2	VA	PWR	2587	848	5/1/1973
Susquehanna 1	PA	BWR	3952	1266	6/8/1983
Susquehanna 2	PA	BWR	3952	1266	2/12/1985
Turkey Point 3	FL	PWR	2644	877	12/14/1972
Turkey Point 4	FL	PWR	2644	760	9/7/1973
Virgil C. Summer	SC	PWR	2900	1030	1/1/1984
Vogtle 1	GA	PWR	3625.6	1215	6/1/1987
Vogtle 2	GA	PWR	3625.6	1215	5/20/1989
Waterford 3	LA	PWR	3716	1200	9/24/1985
Watts Bar 1	TN	PWR	3459	1270	5/27/1996
Watts Bar 2	TN	PWR	3411	1270	10/19/2016
Wolf Creek 1	KS	PWR	3565	1268	9/3/1985

(Source: NRC 2020a, Appendix A; NRC 2020l; EIA 2019)

Table B-2. Licensing and Storage Status of US Operating Commercial Reactors (End of 2020)

Reactor Name	State	Number of License Extensions	Current License Expiration	On-Site Storage License Status [a]
Arkansas Nuclear One 1	AR	1	5/20/2034	General
Arkansas Nuclear One 2	AR	1	7/17/2038	
Beaver Valley 1	PA	1	1/29/2036	General
Beaver Valley 2	PA	1	5/27/2047	
Braidwood 1	IL	1	10/17/2046	General
Braidwood 2	IL	1	12/18/2047	
Browns Ferry 1	AL	1	12/20/2033	General
Browns Ferry 2	AL	1	6/28/2034	
Browns Ferry 3	AL	1	7/2/2036	
Brunswick 1	NC	1	9/8/2036	General
Brunswick 2	NC	1	12/27/2034	
Byron 1	IL	1	10/31/2044	General
Byron 2	IL	1	11/6/2046	
Callaway	MO	1	10/18/2044	General
Calvert Cliffs 1	MD	1	7/31/2034	Specific
Calvert Cliffs 2	MD	1	8/13/2036	
Catawba 1	SC	1	12/5/2043	General
Catawba 2	SC	1	12/5/2043	
Clinton	IL	0	9/29/2026	General
Columbia Generating Station	WA	1	12/20/2043	General
Comanche Peak 1	TX	0	2/8/2030	General
Comanche Peak 2	TX	0	2/2/2033	
Cooper	TX	1	1/18/2034	General
Davis Besse	OH	1	4/22/2037	General
Diablo Canyon 1	CA	0	11/2/2024	Specific
Diablo Canyon 2	CA	0	8/26/2025	
Donald C. Cook 1	MI	1	10/25/2034	General
Donald C. Cook 2	MI	1	12/23/2037	
Dresden 2	IL	1	12/22/2029	General
Dresden 3	IL	1	1/12/2031	
Edwin I. Hatch 1	GA	1	8/6/2034	General
Edwin I. Hatch 2	GA	1	6/13/2038	
Fermi 2	MI	1	3/20/2045	General
Grand Gulf 1	MS	1	11/1/2044	General
H.B. Robinson 2	SC	1	7/31/2030	General and Specific
Hope Creek 1	NJ	1	4/11/2046	General
Indian Point 3	NY	1	4/30/2025	General
James A. Fitzpatrick	NY	1	10/17/2034	General
Joseph M. Farley 1	AL	1	6/25/2037	General
Joseph M. Farley 2	AL	1	3/31/2041	
La Salle 1	IL	1	4/17/2042	General
La Salle 2	IL	1	12/16/2043	
Limerick 1	PA	1	10/26/2044	General
Limerick 2	PA	1	6/22/2049	
McGuire 1	NC	1	6/12/2041	General
McGuire 2	NC	1	3/3/2043	
Millstone 2	CT	1	7/31/2035	General
Millstone 3	CT	1	11/25/2045	
Monticello	MN	1	9/8/2030	General

Reactor Name	State	Number of License Extensions	Current License Expiration	On-Site Storage License Status [a]
Nine Mile Point 1	NY	1	8/22/2029	General
Nine Mile Point 2	NY	1	10/31/2046	
North Anna 1	VA	1	4/1/2038	General and Specific
North Anna 2	VA	1	8/21/2040	
Oconee 1	SC	1	2/6/2033	General and Specific
Oconee 2	SC	1	10/6/2033	
Oconee 3	SC	1	7/19/2034	
Palisades	MI	1	3/24/2031	General
Palo Verde 1	AZ	1	6/1/2045	General
Palo Verde 2	AZ	1	4/24/2046	
Palo Verde 3	AZ	1	11/25/2047	
Peach Bottom 2	PA	2	8/8/2053	General
Peach Bottom 3	PA	2	7/2/2054	
Perry 1	OH	0	3/18/2026	General
Point Beach 1	WI	1	10/5/2030	General
Point Beach 2	WI	1	3/8/2033	
Prairie Island 1	MN	1	8/9/2033	Specific
Prairie Island 2	MN	1	10/29/2034	
Quad Cities 1	IL	1	12/14/2032	General
Quad Cities 2	IL	1	12/14/2032	
R.E. Ginna	NY	1	9/18/2029	General
River Bend 1	LA	1	8/29/2045	General
Salem 1	NJ	1	8/13/2036	General
Salem 2	NJ	1	4/18/2040	
Seabrook 1	NH	1	3/15/2050	General
Sequoyah 1	TN	1	9/17/2040	General
Sequoyah 2	TN	1	9/15/2041	
Shearon Harris 1	NC	1	10/24/2046	No Decision
South Texas Project 1	TX	1	8/20/2047	General
South Texas Project 2	TX	1	12/15/2048	
St. Lucie 1	FL	1	3/1/2036	General
St. Lucie 2	FL	1	4/6/2043	
Surry 1	VA	1	5/25/2032	General and Specific
Surry 2	VA	1	1/29/2033	
Susquehanna 1	PA	1	7/17/2042	General
Susquehanna 2	PA	1	3/23/2044	
Turkey Point 3	FL	2	7/19/2052	General
Turkey Point 4	FL	2	4/10/2053	
Virgil C. Summer	SC	1	8/6/2042	General
Vogtle 1	GA	1	1/16/2047	General
Vogtle 2	GA	1	2/9/2049	
Waterford 3	LA	1	12/18/2044	General
Watts Bar 1	TN	0	11/9/2035	
Watts Bar 2	TN	0	10/22/2055	General
Wolf Creek 1	KS	1	3/11/2045	No Decision

Notes for Table B-2
[a] General = at-reactor storage under a general license as per 10 CFR Part 72, Subpart K
Specific = at-reactor storage under a site-specific license as per 10 CFR Part 72.16
No Decision = no decision yet on at-reactor storage license

(Source: NRC 2020c, NRC 2020l)

Table B-3. Shutdown Commercial Reactors in the US (End of 2020)

Reactor Name	State	Type [a]	Output (MWt)	Operating License Issued	Shutdown Date	On-Site Storage License Status [b]
Power Generation (26)						
Indian Point 1	Buchanan, NY	PWR	615	3/26/1962	10/31/1974	X
Humboldt Bay 3	Eureka, CA	BWR	200	8/28/1962	7/2/1976	Specific
Dresden 1	Morris, IL	BWR	700	9/28/1959	10/31/1978	X
Three Mile Island 2	Middletown, PA	PWR	2,770	2/8/1978	3/28/1979	Specific [e]
Rancho Seco	Herald, CA	PWR	2,772	8/16/1974	6/7/1989	Specific
Shoreham	Wading River, NY	BWR	2,436	4/21/1989 [d]	6/28/1989	N/A
Yankee Rowe	Franklin Co., MA	PWR	600	12/24/1963	10/1/1991	General
Trojan	Ranier, OR	PWR	3,411	11/21/1975	11/9/1992	Specific
San Onofre 1	San Clemente, CA	PWR	1,347	3/27/1967	11/30/1992	General
Zion 2	Zion, IL	PWR	3,250	11/14/1973	9/19/1996	General
Zion 1	Zion, IL	PWR	3,250	10/19/1973	2/21/1997	X
Haddam Neck [c]	Meriden, CT	PWR	1,825	12/27/1974	12/5/1996	General
Maine Yankee	Wiscasset, ME	PWR	2,700	6/29/1973	12/6/1996	General
Big Rock Point	Charlevoix, MI	BWR	240	5/1/1964	8/29/1997	General
Millstone 1	Waterford, CT	BWR	2,011	10/31/1970	7/21/1998	X
Crystal River 3	Crystal River, FL	PWR	2,609	12/3/1976	2/20/2013	General
Kewaunee	Kewaunee, WI	PWR	1,772	12/21/1973	5/7/2013	General
San Onofre 2	San Clemente, CA	PWR	3,438	2/16/1982	6/7/2013	X
San Onofre 3	San Clemente, CA	PWR	3,438	11/15/1982	6/7/2013	X
Vermont Yankee	Vernon, VT	BWR	1,912	3/21/1972	12/29/2014	General
Fort Calhoun	Blair, NE	PWR	479	8/9/1973	10/24/2016	General
Oyster Creek	Forked River, NJ	BWR	1,930	4/9/1969	9/17/2018	General
Pilgrim	Plymouth, MA	BWR	2,028	6/8/1972	5/31/2019	General
Three Mile Island 1	Middletown, PA	PWR	2,079	4/19/1974	9/20/2019	General
Indian Point 2	Buchanan, NY	PWR	3,216	9/28/1973	4/30/2020	X
Duane Arnold	Palo, IA	BWR	1,912	6/22/1970	8/10/2020	General
Demonstration and Prototype (7)						
Carolinas-Virginia Tube Reactor (CVTR)	Parr, SC	PTHW	65	11/27/1962	1/1/1967	N/A
Pathfinder	Sioux Falls, SD	BWR	190	3/12/1964	9/16/1967	N/A
Fermi 1	Newport, MI	SCF	200	5/10/1963	9/22/1972	N/A
Peach Bottom 1	Peach Bottom, PA	HTGR	115	1/24/1966	10/31/1974	N/A
Shippingport	Shippingport, PA	PWR	236	NA	1/1/1982	N/A
LaCrosse	Genoa, WI	BWR	165	7/3/1967	4/30/1987	General
Fort St. Vrain	Platteville, CO	HTGR	842	12/21/1973	8/18/1989	Specific [f]
Demonstration and Prototype (AEC/DOE Owned, not NRC regulated) (4)						
Elk River	Elk River, MN	BWR	58	11/6/1962	2/1/1968	N/A
Hallam	Hallam, NE	SCGM	256	1/2/1962	9/1/1964	N/A
Piqua	Piqua, OH	OCM	46	8/23/1962	1/1/1966	N/A
GE Bonus	Punta Higuera, PR	BWR	50	4/2/1964	6/1/1968	N/A
Experimental Reactors (4)						
GE Vallecitos (VBWR)	Pleasanton, CA	BWR	50	8/31/1957	12/9/1963	N/A
Nuclear Ship (NS) Savannah	Baltimore, MD	PWR	74	8/1/1965	11/1/1970	N/A
GE ESADA Vallecitos (EVESR)	Pleasanton, CA	ESR	12.5	11/12/1963	2/1/1967	N/A
Saxton	Saxton, PA	PWR	24	11/15/1961	5/1/1972	N/A

Notes for Table B-3

[a] PWR = pressurized water reactor
BWR = boiling water reactor
PTHW = Pressurized Tube Heavy Water Reactor
SCF = Sodium-Cooled Fast Breeder Reactor
HTGR = High-Temperature Gas-Cooled Reactor
ESR = Experimental Superheated Reactor
SCGM = Sodium-Cooled Graphite-Moderated Reactor
OCM = Organically Cooled and Moderated Reactor

[b] General = at-reactor storage under a general license as per 10 CFR Part 72, Subpart K
Specific = at-reactor storage under a site-specific license as per 10 CFR Part 72.16
X = site with multiple reactors
N/A = CSNF removed (some is now DOE-managed), no on-site storage needed

[c] Also known as Connecticut Yankee
[d] Reactor was never operated
[e] CSNF is DOE-managed; site-specific license is for post defueling monitored storage (PDMS) at DOE-TMI-2 ISFSI at INL
[f] CSNF is DOE-managed; specific license is for on-site DOE ISFSI, some CSNF has been shipped to INL

(Source: NRC 2020a, Appendix C; NRC 2020c)

APPENDIX C. ONGOING RESEARCH AND DEVELOPMENT

Ongoing DOE-NE SFWST research priorities are summarized in storage and transportation R&D gap reports (Hanson and Alsaed 2019; Teague et al. 2019) and a disposal R&D roadmap (Sevougian et al. 2019c). Five-year plans for storage and transportation R&D and for disposal R&D are outlined in Saltzstein et al. (2020) and Sassani et al. (2020), respectively. These R&D activities are focused on evaluating the integrity of cladding, including high-burnup fuel, canisters (i.e., DPCs), and other systems, structures, and components (SSCs) during extended storage (Section C.1), the transportability of CSNF, including high-burnup fuel, following extended storage (Section C.2), and the potential to dispose of CSNF in DPCs in repositories in a range of geologic media without repackaging (Section C.3).

Within Sections C.1, C.2, and C.3 the relevance and integration of these ongoing and planned activities, and the need for continued interaction with industry and the NRC, are identified to help ensure that CSNF can continue to be stored safely and is ready for transportation and disposal when a CISF and/or a repository become available.

C.1. Extended Storage at Surface Facilities

Since 2012, a series of gap analyses have been conducted to identify DOE-NE SFWST R&D priorities related to the extended storage and subsequent transportation of SNF (Hanson et al. 2012; Hanson and Alsaed 2019; Teague et al. 2019). Research performed between 2012 and 2019 addressed the higher priority gaps (Stockman et al. 2015); as a result, some gaps were either closed or downgraded to lower priority (Teague et al. 2019). These gaps included potential cladding degradation mechanisms – delayed hydride cracking (DHC), creep, annealing of radiation damage, and oxidation – which were downgraded in part due the NRC findings in the "Managing Aging Processes in Storage (MAPS) Report" (NRC 2019b) that they would not lead to significant cladding degradation during extended operation under anticipated temperatures and hoop stresses (NRC 2019b, Section 3.6.1).

A five-year plan for storage and transportation R&D is outlined in Saltzstein et al. (2020) to address the remaining highest priority gaps. The high-priority gaps, their current status, and planned R&D are summarized in the following subsections: cladding and canister thermal profiles (Section C.1.1), canister SCC (Section C.1.2), cladding and canister stress profiles (Section C.1.3), canister drying issues (Section C.1.4), and CSNF and cladding degradation mechanisms (Section C.1.5). There are also subsections addressing security (Section C.1.6) and the impacts from Fukushima (Section C.1.7).

C.1.1. Cladding and Canister Thermal Profiles

Nearly all degradation mechanisms for storage and transportation SSCs are dependent on temperature. Peak cladding temperatures (PCT)[20] are limited to 400°C during storage to minimize the potential for degradation of the cladding; high temperature can increase rod internal pressure which in turn can increase the pressure-induced hoop stress, which has a design limit of 90 MPa (NRC 2010, Sections 4.4.2 and 8.4.17). Thermal modeling and experiments conducted over the last several years have contributed to a better understanding of thermal profiles, including from

[20] The peak cladding temperature is only experienced by a small fraction of spent fuel in a canister (NRC 2010, Section 8.4.17).

high-burnup fuel, and their potential effects on cladding and canister integrity during extended storage. Specific R&D activities to address the thermal profiles gap include (Saltzstein et al. 2020):

- **EPRI/DOE High Burnup Dry Storage Research Project** – A bolted lid TN-32B cask (the Research Project Cask), instrumented with thermocouple lances to measure temperature distributions within the cask, was loaded with high-burnup PWR spent fuel (with different types of cladding: Zircaloy-4, ZIRLO™, M5™) at the Dominion North Anna Power Station in November 2017. Temperatures were recorded during drying, helium backfilling, and a two-week control period in the fuel building; continued monitoring and recording of temperatures inside the cask are ongoing and are planned to continue for about 10 years on the North Anna ISFSI pad (EPRI 2014; EPRI 2019). The data collected from the project can be used to support licensing for high-burnup fuel as well as for model validation to support future cask or canister designs. Cladding properties after 10 years of dry storage will be examined and compared to the pre-storage cladding properties from a set of 25 sibling pins (sister rods) extracted during loading of the Research Project Cask (see Section C.1.5). Key lessons learned are:
    - Industry typically employs conservative or bounding assumptions that result in significantly higher calculated cladding temperatures than are really experienced. In the Research Project Cask, the PCT (in a neighboring guide tube) was measured to be 237°C, well below the PCT of 348°C originally estimated by industry in the licensing process (EPRI 2019) and significantly below the NRC guidance of 400°C (NRC 2010, Sections 4.4.2 and 8.4.17).

- **Thermal Modeling of the High Burnup Research Project Cask** – Detailed predictive thermal models were developed to compare to actual collected data in the Research Project Cask, with a focus on PCT and distributions/profiles as well as on the cask surface temperatures both when the cask was indoors (Fort et al. 2019a) and on the ISFSI (Fort et al. 2019b). A round robin modeling exercise was performed where participants used the same inputs applied to different models to predict cask temperatures (EPRI 2020a). An international round robin exercise is currently underway to again use the same inputs applied to different models. The key lessons learned are:
    - Using the best available information, the thermal models were able to make blind predictions of most cladding and other SSC temperatures to within 20-95°C, but typically in the 20-50°C range, although biased high (EPRI 2020a)
    - An adjusted best estimate model (Fort et al. 2019a) was able to predict most measured temperatures to within 15°C, again biased high.
    - Because of the large axial and radial thermal gradients within a cask or canister, only a very small (for the case of the Research Project Cask <3%) amount of the cladding is at the PCT with the overwhelming majority significantly cooler (Fort et al. 2019a)

- **BWR Dry Cask Simulator (DCS)** – There is ongoing DOE-NE SFWST-sponsored thermal-hydraulic testing at Sandia National Laboratories (SNL) using a full-scale BWR heater assembly in a confined vessel that is a scaled representation of a canister and storage cask (overpack). The test apparatus provides thermocouples to collect temperature and air mass flow data associated with convective cooling of simulated spent fuel in a helium atmosphere under representative thermal power (assembly heat load) and gas (helium and air) backfill pressure environments. A suite of tests at different representative combinations of thermal power and gas pressure has been completed in a vertically-oriented DCS, with configurations representative of above ground and below ground vertical DCSSs (Durbin and Lindgren 2017; Durbin and Lindgren 2018). Testing is ongoing in a horizontal configuration (Lindgren et al. 2019).

- **Thermal-Hydraulic Model Validation with DCS Data** – Round robin model validation exercises were carried out using the data obtained from the vertical DCS testing (Pulido et al. 2020a) and are ongoing with the horizontal DCS testing (Pulido et al. 2020b). Collaborators from the US (SNL, NRC, Pacific Northwest National Laboratory (PNNL)) and Spain (Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (CIEMAT), and Empresa Nacional del Uranio, SAS (ENUSA)) performed blind model predictions of temperature profiles under various thermal power (heat load) and gas backfill (pressure and composition) conditions. The key lessons learned are:
  - The vertical DCS modelers were given design and boundary conditions and were able to calculate temperatures within ~1-20 °C of the measured temperatures, though all were biased high (Pulido et al. 2020a). All models were able to predict the PCT within a 5% root mean square error.
  - The horizontal DCS modelers were given design and boundary conditions and the results for two of the tests. In an initial round robin exercise, the models were able to predict a combined metric of PCT, temperature profiles, and air mass flow rate within a 5% root mean square error.
  - This modeling validation effort dovetails with the thermal modeling of the High Burnup Research Project Cask by providing another reference by which to judge the adequacy of modern thermal-hydraulic codes and techniques.

- **EPRI Thermal Performance Phenomena Identification and Ranking Tables (PIRTs)** – EPRI sponsored two separate PIRT studies on thermal performance of DCSSs, which confirmed that additional work is recommended to be able to accurately predict realistic component temperatures.
  - The thermal modeling PIRT (EPRI 2020b) identified five key phenomena that could provide a significant reduction in bias or uncertainty. Many of these opportunities can be realized by the nuclear industry.
  - The decay heat PIRT (EPRI 2020c) found that most methodologies significantly overestimate decay heat, very often because they utilize simplified inputs. One area needing additional data to provide validation is measurement of decay heat for short cooled (<2 years) and higher burnup (>55 GWd/MTU) fuels.

Research results to date suggest that DCSSs have significantly lower heat loads than estimated in licensing calculations. This could have an impact on cladding behavior, the ability to de-inventory pools and to transport CSNF sooner, and to reduce the footprint of the repository through reduced drift spacing. Continuation of ongoing testing, the EPRI/DOE High Burnup Dry Storage Research Project and the horizontal DCS, will provide additional data on temperature profiles within canisters in dry storage that includes high-burnup fuel that will support further characterization of cladding and other SSC degradation during the decades of extended storage now expected. Scaling up the DCS testing to larger systems (i.e., more than one assembly) could provide additional data and insights. Continuation of the thermal-hydraulic modeling efforts will improve the ability to simulate decay heat transfer within canisters and casks to more accurately predict PCTs and temperature profiles without excess conservatism.

Thermal management of CSNF across the operational boundaries (i.e., storage, transportation, and disposal) must also be considered. For example, thermal aging requirements for transportation and/or disposal could extend storage times. Evaluation of thermal profiles in CSNF canisters, and in storage and transportation casks, can have significant positive benefits to downstream operations. These issues are discussed further in Section C.2.1.

The cross-cutting nature of thermal issues into other technical gaps is also important. For example, accurate estimation of canister thermal environments is critical to the understanding of canister SCC (Section C.1.2), canister drying issues (Section C.1.4), and cladding degradation mechanisms (Section C.1.5).

C.1.2. Canister Stress Corrosion Cracking

In dry storage, CSNF is commonly stored in welded stainless-steel canisters enclosed in passively-ventilated overpacks. Over time, dust accumulates on the canister surfaces, and as the CSNF cools, salts within that dust will deliquesce to form concentrated brines. If the salts contain aggressive species such as chloride, then the resulting brine can cause localized corrosion, and if sufficient tensile stresses are present in the metal, SCC can occur (Schaller et al. 2020). Over time, SCC cracks could penetrate the canister wall. The risk of corrosion and SCC is greatest in near-marine settings, where chloride-rich sea-salt aerosols are deposited on the canister surface (Schaller et al. 2020); however, chlorides may also be present in the atmosphere near cooling towers, salted roads, or other locations (NRC 2019b, Section 3.2.2.5). The NRC (2019b, Section 3.2.2.5) has identified the potential for SCC of austenitic stainless-steel (e.g., Types 304 and 316), in particular around the canister welds, as credible during extended storage. The susceptibility of austenitic stainless steels to SCC tends to increase as the chloride concentration in the solution increases, but the level of chlorides required to produce SCC is very low and is dependent on the type of chloride salts present. The material is more resistant to SCC in NaCl solutions but crack readily in $MgCl_2$ solutions (NRC 2019b, Section 3.2.2.5).

High-priority research gaps related to SCC include welded canister atmospheric corrosion, consequence assessment of canister failure, and canister external monitoring (Saltzstein et al. 2020). Specific R&D activities to address the SCC-related gaps, which are being performed at SNL and PNNL and also at universities funded as part of DOE's Nuclear Energy University Program (NEUP), include (Saltzstein et al. 2020; Schaller et al. 2020):

- **Analysis of Canister Surface Environments** – Work to define dry storage canister surface environments to evaluate the potential for SCC includes several tasks (Schaller et al. 2020):
    - Collection of dust deposition specimens from ISFSI site locations helped to establish a more complete understanding of the potential chemical environment formed on the canister. Under the sponsorship of EPRI, there have been seven site visits in different geographic areas of the US. to assess the amount of general corrosion (visual inspection) and salt loadings deposited on the canisters (Saltzstein et al. 2020, Section

2.1.3). In general, there was little indication of any noteworthy corrosion on any of the inspected canisters. After analyzing samples, soluble salt deposition was confirmed at all sites, but the surface concentration of salts varied widely over the canister surface at each site and the amount of corrosion-promoting chloride varied widely between sites. Low chloride salt loads found at some sites are a positive factor indicating potentially lower risk than previously assumed. Collection of additional in-service samples will provide a better understanding of the diversity of dust depositions in different geographic areas of the country.

  - Continued evaluation of the stability of magnesium chloride brines. Brine stability experiments have shown that some important salt phases, including ammonium minerals and magnesium chloride, the most deliquescent component in sea-salts, are not stable at elevated temperatures, potentially limiting the conditions at which a deliquescent brine can form, and allow corrosion to occur (Saltzstein et al. 2020, Section 2.1.3). Moreover, once corrosion initiates, magnesium chloride brines can precipitate as hydroxide phases, eventually drying out, and potentially limiting the extent of corrosion. This work illustrates the importance of thermal profiles (Section C.1.1) on SCC.

- **Corrosion Testing and Modeling** – Testing and modeling in canister relevant environments includes (Schaller et al. 2020):
  - Continued evaluation of pitting observed in large-scale exposure tests and continued pit size modeling to better understand pit growth, both size and morphology, and SCC initiation. Two major observations produced from these testing and modeling activities are: (1) $MgCl_2$ brines influence the pit morphology towards the formation of more irregular shapes with near surface microcracks, and (2) $MgCl_2$ brines influence the controlling cathodic kinetics through dominance of the hydrogen evolution reaction in the cathode and formation of precipitates that lower the predicted maximum pit size. Understanding the potentially opposing roles of these two phenomena on SCC initiation will better enhance understanding of the susceptibility of CSNF canisters to SCC.
  - Initiation of load frame testing of the crack growth rate of stainless-steel canister materials in canister-relevant environments. This new crack growth rate data will improve the understanding how cracks, once initiated, will behave under canister-relevant conditions.
  - Development of the SNL probabilistic SCC model, which includes mechanistically-based sub-models to describe (1) brine evolution, both before and after initiation of corrosion; (2) corrosion processes, and (3) SCC crack initiation and growth. The current model framework contains numerous simplifying assumptions, many with insufficient scientific basis. Continued model validation against newly acquired data for salt compositions and deposition rates and statistical pitting and SCC initiation and growth data for canister-relevant conditions will reduce uncertainty and improve the capability to estimate the timing and occurrence of SCC.

- **Consequence Assessment of a Canister Through-Wall Crack** – Experimental testing, coupled with modeling and analysis, will be used to estimate the radiological consequences (from gaseous and particulate release) of a potential breach of confinement (i.e., a through-wall crack) due to SCC during extended storage (Saltzstein et al. 2020, Section 2.2.2).

- A system for testing flow rates and aerosol transmission through engineered microchannels that have dimensions similar to stress corrosion cracks, and preliminary test results, are described in Durbin et al. (2020). Parallel modeling efforts are underway (e.g., Chatzidakis 2018; Lanza et al. 2019) to understand the results of the tests and to provide iterative feedback for future testing. Assessment of particle/aerosol (e.g., from fuel degradation) release and depletion as a function of time is needed for modern dry storage systems, which have relatively strong internal natural convection that would prevent gravitational settling of particulates. Model validation to real-world data will enable the assessment dose consequences from potential through-wall cracks.
- These results will also inform the monitoring tasks by identifying the rates of release and thus the sensitivity of sensors necessary to detect any potential leak.

- **Canister Monitoring** – Activities to evaluate the effectiveness of external and internal monitoring of canisters include (Saltzstein et al, 2020, Section 2.3.1):
  - EPRI has sponsored projects for the development of robotics and sensors capable of accessing the tight space between the concrete overpacks and welded canisters. The primary focus of these technologies is the detection of SCC of the canister welds. The latest results of this ongoing effort are summarized in EPRI (2016).
  - There is a desire to be able to monitor the internals of the cask without penetrations through the confinement. Specifically, it would be beneficial to detect helium leakage either directly or by interrogation of the canister internals.

- **SCC Repair and/or Mitigation** – Ongoing work, largely through EPRI and NEUP, is evaluating options for repair and mitigation of SCCs under dry storage conditions (Saltzstein et al. 2020, Section 2.1.3). An initial screening of potential coatings that could be used for repair and mitigation of canisters has been completed; possible candidates have been identified. The use of cold spray techniques is being investigated by Ross and Alabi (2019) and will be evaluated by multiple university programs (e.g., Knight et al. 2020).

Continuation of these testing and analysis activities will improve the capability and accuracy in evaluating SCC initiation and growth rates as a function of environmental parameters (salt load, temperature, humidity, and salt/brine composition, overpack design and airflow), material properties (e.g., degree of sensitization, surface roughness, degree of cold work), and stress state (Saltzstein et al. 2020, Section 2.1.3). A full-scale canister deposition demonstration at various heat loads would provide useful data on deposition and brine stability and serve as a platform for inspection and eventual repair and mitigation techniques

C.1.3. Cladding and Canister Stress Profiles

Testing and modeling conducted over the last several years have contributed to a better understanding of the types of stresses (magnitude, frequency, duration, etc.) and their potential effects on cladding and canister integrity during extended storage and transportation. These activities have primarily focused on stresses encountered under transportation conditions and are described in Section C.2.2. Cladding and canister stress profile considerations and needs during extended storage include (Saltzstein et al. 2020, Section 2.1.2):

- Development of a cumulative effects model that considers all external loads during loading/handling, extended dry storage (including design basis seismic events), transportation, and any other additional storage or handling (see Section C.2.2 for additional discussion).

- Quantifying seismic loads (e.g., assembly hardware loads, rod-to-rod impacts) that may be experienced during a design basis seismic event throughout the period of extended storage.

C.1.4. Canister Drying Issues

Many degradation mechanisms are dependent on or accelerated by the presence of water (e.g., fuel oxidation, hydrogen buildup). The drying process after transfer of spent fuel to a canister is designed to remove excess water from the assemblies, if standardized drying procedures are followed it is estimated that only about 0.43 mole of water will remain (NRC 2010). In addition to free water, other potential sources of water in a canister include bulk trapped water such as in dashpots or in failed fuel rods, precipitated boric acid, structural and adsorbed water in crud or hydrated corrosion products, water trapped in porous materials such as Boral™, and chemisorbed and physisorbed water (Bryan et al. 2019). Jung et al. (2013) performed an analysis of possible degradation resulting from water remaining after drying and concluded that below 1 L (55.5 moles) of water, degradation of the fuel, cladding, or other internal components would not be significant over 300 years. While there is no direct evidence that the amount of water that remains in a cask/canister after a normal drying process is of concern, water in excess of 1 L could result in a number of degradation mechanisms that are not currently considered (e.g., galvanic corrosion) (Saltzstein et al. 2020, Section 2.2.1).

Specific R&D activities to address the drying issues gap, which are being performed at SNL, PNNL, and through NEUP, include (Saltzstein et al. 2020, Section 2.2.1):

- The University of South Carolina led a NEUP project to experimentally evaluate residual water in a test mock-up of a dry canister after both cold vacuum and forced helium drying procedures. General results showed evidence of freezing on the spacer discs and siphon tube, as well as small amounts of bulk water in the simulated failed fuel rod, and the spacer discs siphon tube (Knight 2018).
- Gas samples were pulled from the Research Project Cask after vacuum drying, helium backfilling, and sealing in the operational storage condition. These samples suggested that about 100 ml of water remained in the cask atmosphere after drying (Bryan et al. 2019; EPRI 2019).
- Salazar et al. (2020) have performed initial experiments to examine spent fuel drying operations conducted by industry. They have shown that water can be removed using sequential drying hold points in a small-scale pressure vessel with a partially submerged heater rod. These initial efforts are expected to be leveraged into an intermediate-scale drying test, capable of fully exploring water retention via failed fuel, fuel assembly voids (e.g. PWR dashpots), canister hardware, and residual borated water.

The amount of water observed in the NEUP work and the Research Project Cask gas sampling is small, below the 1 L volume that could cause concern. However, collection of additional gas samples (to get residual water data) from in-service storage systems and experiments using more advanced and larger-scale heater and assembly designs will help to confirm estimates of residual water in canisters.

C.1.5. CSNF and Cladding Degradation Mechanisms

As noted in Section C.1, several potential cladding degradation mechanisms – DHC, creep, annealing, and oxidation – were determined not to lead to significant cladding degradation during extended operation under anticipated temperatures and hoop stresses (NRC 2019b, Section 3.6.1), which includes spent fuel of all burnups the NRC currently licenses. The remaining cladding degradation gap - hydride reorientation and embrittlement for high-burnup fuel – is addressed by NRC (2020e) and supported by NRC (2020f, Section 8.5.15). Specifically, while hydride reorientation can occur below the 400°C PCT limit, it would not have a significant impact on the fatigue strength and bending stiffness of high-burnup spent fuel under bending moments that produce longitudinal tensile stresses in the rod and it would not be sufficient to decrease the ductility of the cladding for standard high-burnup PWR fuel (NRC 2020e, Section 1.5; Saltzstein et al. 2020, Section 2.3.3). This position is supported by embrittlement data obtained from Ring Compression Tests (RCT) conducted at Argonne National Laboratory (ANL) (Billone and Burtseva 2018) and data obtained from thermal measurements taken after loading and drying of the Research Project Cask described in Section C.1.1 (Fort 2019a).

Although the cladding degradation gaps are essentially closed, the following activities are ongoing to further understand cladding degradation of high-burnup spent fuel subjected to extended storage (Saltzstein et al. 2020, Section 2.3.3):

- **Sibling Pin Testing** - There is an ongoing sibling pin (sister rod) experiment to establish material properties of cladding from high-burnup fuel rods with similar high-burnup irradiation histories as those in the EPRI/DOE High Burnup Dry Storage Research Project (Section C.1.1). Prior to the Research Project Cask being loaded, 25 fuel rods were removed from seven different fuel assemblies. These so-called sibling pins were shipped to Oak Ridge National Laboratory (ORNL) and PNNL for detailed examination and testing (Saltzstein et al. 2018). The sibling pin testing will provide (1) baseline characterization and mechanical property data that can be used as a comparison to the fuel loaded into the Research Project Cask, and (2) mechanical property data at temperatures greater than those achieved in the Research Project Cask to provide data representative of DCSSs that had PCTs closer to the 400°C limit (Saltzstein et al. 2020, Section 2.1.4). Lessons learned to date include (Saltzstein et al. 2020, Section 2.1.4):
    - End-of-life rod internal pressures have been measured on 18 of the 25 sibling pins. In all but three instances, the pressure was <4 MPa (Montgomery et al. 2019) at ambient temperature. Two of the rods had pressure of ~5 MPa, but both were from a test program over 30 years ago and the initial helium fill pressures were significantly higher than modern high-burnup fuel designs. Assuming as-fabricated cladding dimensions and a uniform 400°C along the length of a rod, even with an ambient temperature pressure of 5 MPa, the hoop stress will still be <82 MPa. This provides confidence that RCTs conducted with a hoop stress of 90 MPa represents a reasonable upper bound for standard US high-burnup fuel from PWRs and is below the hoop stress needed to form enough radial hydrides to significantly affect cladding ductility (see e.g., Billone and Burtseva 2020).
    - In addition to the effects of hydride orientation on bending strength and ductility, hydride orientation has also been postulated to affect the failure stresses and strains under pinch-type loads. Pinch-type loads could potentially occur during postulated drop

accidents in storage, NCT, or HAC during transportation (NRC 2020e, Section 1.5). These issues are addressed in Section C.2.3.
  - It is assumed that PWR cladding mechanical properties bound those of BWR cladding since the PWR cladding is at much higher internal pressure. However, corrosion of cladding is typically much higher in BWRs and thus some confirmatory data on BWR cladding is desired.
  - Fuel rods with higher internal pressure, such as Integral Fuel Burnable Absorber (IFBA) rods, may have hoop stresses above the threshold for radial hydrides to decrease ductility and data needs to be obtained.
  - Newer cladding alloys and designs, such as are planned for ATF, and especially designs meant to go to significantly higher burnups may behave differently during storage and transportation, so some confirmatory testing is necessary.

C.1.6. Security

ISFSI licensees must implement a layered defensive security that includes on-site protective forces with appropriate skills, weaponry, and other response equipment, and security systems. The strategy must include procedures to defend against physical attacks, insider threats, and cyber attacks (BRC 2012, Section 5.5.1).

Following the terrorist attacks of September 11, 2001, the NRC has engaged in the process of proposed a rulemaking that would revise the existing security requirements for the storage of SNF at an ISFSI or an MRS (i.e., a CISF) (NRC 2020g). This proposed rulemaking would also make conforming changes to the licensing requirements for security plans and programs at ISFSIs and MRSs. The NRC issued a draft "regulatory basis" document in December 2009 and has received numerous comments on proposed technical approaches. Among other issues, the NRC is considering whether to require comprehensive "denial" capability on site—that is, sufficient security forces and weaponry for facility personnel to repel an attack on their own—or instead to require a detect/assess/communicate strategy that would rely on assistance from local, state and federal authorities (BRC 2012, Section 5.5.1).

There are ongoing DOE-NE SFWST R&D activities to identify the needs for site-specific physical security and monitoring during extended storage and during transportation (e.g., Durbin and Luna 2013).

C.1.7. Fukushima

In response to the earthquake, tsunami, and resulting reactor accidents in 2011 at the Fukushima Dai-ichi facility in Japan, the NRC conducted a review of the relevant regulatory requirements, programs, and processes, including their implementation and recommended enhancements to ensure the continued safety of US commercial nuclear power plants (NRC 2020m; NRC 2014a, Section 2.1.2). The enhancements included adding capabilities to maintain key plant safety functions following a large-scale natural disaster; updating evaluations on the potential impact from seismic and flooding events; new equipment to better handle potential reactor core damage events; and strengthening emergency preparedness capabilities (NRC 2020m).

Specific to storage operations, the NRC concluded that some additional improvements to spent fuel pool storage and other structures, systems, and components would be beneficial, including requiring operating reactor licensees to (1) implement mitigating strategies to ensure that spent fuel pool cooling can be accomplished through alternative means to prevent fuel damage, and (2) have a reliable means to remotely monitor a wide range of spent fuel pool levels to support effective prioritization of event mitigation and recovery actions in the event of a beyond-design-basis external event (NRC 2014a, Appendix F).

While the events at Fukushima resulted in the aforementioned enhancements to reactor and spent fuel pool operational safety, an additional impact of Fukushima, like previous events at Three Mile Island and Chernobyl, was to reduce public confidence in the safety of nuclear operations in general (Section 4.4.3).

C.2. Transportation

As noted in Section 3.2, the standards, regulations, and safety record for past shipments of SNF, HLW, and other nuclear materials are proven, but the transportation of the projected ~136,000 MTHM of CSNF, including high-burnup fuel and after extended storage, will require a large-scale shipping campaign and infrastructure spread over several decades. For example, the mid-century repository scenarios outlined in Section 5.1 all have an assumed emplacement period of 57 years.

Since 2012, a series of gap analyses have been conducted to identify DOE-NE SFWST R&D priorities related to the extended storage and subsequent transportation of SNF (Hanson et al. 2012; Hanson and Alsaed 2019; Teague et al. 2019). A five-year plan for storage and transportation R&D is outlined in Saltzstein et al. (2020) to address the remaining highest priority gaps. The high-priority gaps for transportation, their current status, and planned R&D are summarized in the following subsections: cladding and canister thermal profiles (Section C.2.1), cladding and canister stress profiles (Section C.2.2), and CSNF and cladding integrity (Section C.2.3).

In addition to the gap analysis priorities, a strategy for a large-scale transportation campaign following extended storage must also consider transportation planning and system logistics (Section C.2.4), transportation fleet design and certification (Section C.2.5), and security (C.2.6). The transportation strategy must be integrated with extended storage (Section C.1) and ultimate disposal in DPCs and/or purpose-designed disposal containers (Section C.3).

C.2.1. Cladding and Canister Thermal Profiles

The thermal experiments and modeling outlined in Section C.1.1 contribute to the understanding the temperature profiles for cladding, canisters and other SSCs during dry storage. These analyses provide the basis for the integration of thermal management of CSNF across the operational boundaries, from storage to transportation to disposal. Specific R&D activities and considerations for transportation include:

- The surface temperature of the transportation cask must not exceed 50°C (10 CFR Part 71.43). As a result, thermal aging may be required before transportation, which could extend storage times. Transportation of large DPCs, especially with high-burnup fuel, may not be feasible for decades to come due to the high heat loads generated and/or external dose limits (see Figure 4-1). Similarly, canister thermal profiles impact disposability following transportation (e.g., geologic-media and backfill-specific thermal constraints, waste package and drift spacing, aging, etc.).

- The development of a thermal model that estimates PCT and temperature profiles for a canister from the beginning of storage, through transportation, to disposal would significantly enhance integration across the back end of the fuel cycle. This "canister thermal lifetime" model could evaluate temperature histories for a range of DPCs types, loadings, and burnups to inform canister and cask designs, transportation timing, and disposal aging/cooling and emplacement spacing. An example of a thermal model platform, currently just applied to storage lifetime, is documented in Bhatt et al. (2019). A canister thermal lifetime model could also inform the thermal environment to evaluate injectable fillers for the direct disposal of CSNF in DPCs (Section 5.1.2.4.2).

C.2.2. Cladding and Canister Stress Profiles

Degradation can be exacerbated by external loads (forces, strains, accelerations, etc.) that SSCs might be subjected to during extended storage and subsequent transportation. A degradation mechanism does not have to proceed all the way to failure on its own; sufficient external loads on a weakened material may also result in failure. Testing and modeling conducted over the last several years have contributed to a better understanding of the types of stresses (magnitude, frequency, duration, etc.) and their potential effects on cladding and canister integrity during extended storage and transportation. Specific R&D activities to address the stress profiles gap include (Saltzstein et al. 2020, Section 2.1.2):

- **Multi-Modal Transportation Test** – The MMTT was a large collaborative project between ENSA in Spain, DOE-NE SFWST (principally SNL and PNNL), and partners from the Republic of Korea that was performed in 2017 (McConnell et al. 2018). An ENUN 32P cask was loaded with three surrogate assemblies each instrumented with strain gauges and accelerometers. The other 29 spaces were loaded with dummy assemblies to provide the proper mass. The basket, cask, cradle, and conveyance systems were also instrumented. Data were collected during the following NCT segments: handling, heavy-haul truck, coastal vessel, ocean-going ship, open rail, and captive-rail (Section 3.2.1).
- **MMTT Data Analysis** – Kalinina et al. (2018) analyzed the data from 8,539 miles of transportation data from the MMTT and reported the peak strains and accelerations for each type of transportation. When combined with previous transportation test results (e.g., shaker table, over road truck, ENSA cask), the maximum strains considering both shock and vibration in SNF cladding were determined to be approximately 300 $\mu\epsilon$ (μm/m). Yield values, in strain, for irradiated SNF cladding are on the order of 7000 - 9000 $\mu\epsilon$. Therefore, NCT are not expected to produce any additional cladding failures.
- **MMTT Modeling** – Klymyshyn et al. (2018) performed a fatigue analysis using MMTT data and determined that the cumulative fatigue damage from one approximately 2000-mile rail trip was about 10 orders of magnitude below the cladding fatigue failure limit. Therefore, fatigue of cladding during NCT is considered a non-issue. Klymyshyn et al. (2019) expanded the fatigue analysis to examine the various transportation and cradle designs in the US as well as varying fuel rod stiffness to determine how a "real" system would respond. Fatigue analysis results using the Atlas railcar model (Section C.2.5) showed even lower fatigue damage than predicted in the MMTT. The conclusion is that shock and vibration to SNF cladding during NCT is negligible.
- **Drop Tests** – A 30-cm drop test and analysis are considered by NRC to be part of NCT. Kalinina et al. (2019) performed 30-cm drop tests on an instrumented 1/3-scale ENUN 32P

cask belonging to ENSA with dummy fuel assemblies and surrogate impact limiters. The tests were performed in December 2018 at the Bundesanstalt für Materialforschung und -prüfung (BAM). Accelerations of the cask and fuel assemblies were measured. This data was used to design drop tests at SNL, first using first a dummy assembly and then a full-scale surrogate instrumented assembly such that the accelerations would be equivalent to those measured in the 1/3-scale test. Kalinina et al. (2019) gives the results of the 1/3-scale drop tests and the 30-cm drop of the full-scale dummy assembly. Kalinina et al. (2020) describes the results of 30-cm drop of the full-scale surrogate assembly.

- **Cumulative Effects of Structural Loads** – An analytic framework has been developed for a cumulative effects model that considers all external loads during loading/handling, extended dry storage (including design basis seismic events), and NCT (Klymyshyn et al. 2020). The phenomena being considered are structural failure, fatigue failure, and cracking failure (design basis and from SCC) over the full life cycle of an CSNF system. Considered together, there is the possibility that multiple individual low-magnitude events could cause enough incremental damage to SSCs of the CSNF system to challenge its structural integrity if the time span is long enough or if the life cycle includes a sufficient number of loading events. The cumulative effects model will be used to perform a systems analysis to evaluate the integrity of the cladding, canister, and other SSCs in preparation for transportation following extended storage.

The work quantifying operational loads on SNF cladding under NCT has shown that induced stresses are well below yield levels. Similar knowledge of external loads that other SSCs will experience during their lifetime up to final disposal is equally necessary. This includes quantifying seismic loads throughout the period of extended storage and collecting additional data on assembly hardware loads and rod-to-rod impacts to provide model validation data on SNF cladding potential failure (Saltzstein et al. 2020, Section 2.1.2). Together, the data and analyses will inform transportation system decisions and designs.

C.2.3. CSNF and Cladding Integrity

Several cladding degradation mechanisms – hydride reorientation, DHC, creep, annealing, and oxidation – were discussed in Section C.1.5 and were determined not to lead to significant cladding degradation under temperatures and hoop stresses anticipated during extended storage. These findings provide the basis for evaluating the integrity of the spent fuel (including thermally hotter and more radioactive high-burnup fuel), cladding, and canisters during transportation operations following extended storage. Additional R&D activities and considerations for transportation include:

- Even with the expectation that hydride orientation would not have a significant impact on the fatigue strength and bending stiffness of high-burnup SNF under bending moments that produce longitudinal tensile stresses in the rod (Section C.1.5), the NRC staff expressed concern that hydride orientation could affect the failure stresses and strains under pinch-type loads (e.g., from rod-to-rod or grid-to-rod contact) (NRC 2020e, Section 1.5). Pinch-type loads could potentially occur during postulated drop accidents in storage, or in NCT or HAC during transportation. Recent data from Cyclic Integrated Reversible Fatigue Tester (CIRFT) bending tests, documented in NRC (2017), and stress data obtained from the MMTT load quantification tests (Section C.2.2) addressed these concerns. Results of these tests, integrated with mechanical data from the ANL RCTs and thermal measurements from the Research Project Cask (Section C.1.5), indicate that risks associated with hydride reorientation and embrittlement to SNF cladding integrity are low for current US fuel designs, burnups, and reactor operational limits (Saltzstein et al. 2020, Section 2.3.3).

- NRC (2020e, Section 5) confirms this conclusion; "Further, the NRC staff concludes that the orientation of the hydrides is not a critical consideration when evaluating the adequacy of cladding-only mechanical properties. Therefore, the use of mechanical properties for cladding in either the as-irradiated or hydride-reoriented condition is considered acceptable for the evaluation of drop and cask tipover accident scenarios. If an applicant is unable to demonstrate satisfactory performance of the high-burnup SNF rod by assuming cladding-only mechanical properties, the staff has proposed an alternative approach for using the results from static bend testing to account for the increased flexural rigidity imparted by the fuel pellet."
- The ongoing Sibling Pin Testing (Section C.1.5) will provide further data on cladding integrity to inform models of high-burnup fuel behavior under normal, off-normal, and accident conditions, during both extended storage and transportation.
- CSNF (e.g., cladding) that has been stored for extended periods (e.g., greater than 100 years (NRC 2014a)) may require additional handling and preparation, or even repackaging (Section 5.1.3) before it can be transported.
- Subcriticality, which can be maintained during dry storage by a combination of fuel geometry, neutron poisons, moderator exclusion, and burnup credit, was initially a high-priority gap under transportation conditions (NCT and HAC) (Hanson et al. 2012). However, the gap was downgraded based on NRC (2012) and supported by NRC (2020h, Section 6), which allows for reliance on PWR SNF burnup credit during transportation. BWR SNF typically does not need to rely on burnup credit during transportation. The potential for criticality of both PWR and BWR SNF during direct disposal of DPCs is discussed in Section 5.1.2.4.

C.2.4. Transportation Planning and System Logistics

Transportation of the projected ~136,000 MTHM of CSNF, including high-burnup fuel and after extended storage, will require a large-scale shipping campaign and infrastructure spread over several decades. At a nominal annual waste acceptance rate at the repository of 3,000 MTHM (Section 3.3.2), it would take ~45 years to transport the entire projected CSNF inventory. The mid-century repository scenarios outlined in Section 5.1 all have an emplacement period of 57 years. The presence of a CISF would potentially double the number of shipments. Planning for an efficient and resilient transportation system should balance competing priorities to evaluate the impact of operational decisions on system-level metrics such as resource utilization, safety, security, and cost. Transportation planning and systems logistics considerations are summarized in Section 3.2. Specific activities that would support transportation include:

- Advance planning timeframes on the order of a decade could be required to plan and coordinate a transportation strategy and to establish the institutional and physical infrastructure to conduct a large-scale shipping operation.

- Budget availability is needed for acquisition of the transportation fleet and for the associated capital investments, such as maintenance facilities and/or refurbishments to the infrastructure (see Appendix E).

- Advance planning for a transportation system should also address:
  - Early implementation and testing of institutional arrangements involving state, tribal and local officials, as suggested by BRC (2012, Chapter 9). The NWPAA (1987, Section 180(c)) mandates technical assistance and funding from the Nuclear Waste Fund for training local government and Indian tribes to cover procedures required for safe routine transportation of SNF and HLW, as well as procedures for dealing with emergency response situations. The TSLCC assumes that these planning and training grants will be awarded three to five years prior to the start of operations and will continue during the transportation operations period (DOE 2008b, Section 3.2).
  - Design, fabrication, testing, and licensing of the necessary transportation fleet (see Section C.2.5).
- Planning can be supported by transportation system logistics modeling (Section 3.2.3 and Appendix E). Important logistics considerations are (1) identification of transportation routes (e.g., from reactor sites and shutdown sites to a repository, and possibly also to a CISF), (2) development of a transportation schedule, and (3) availability of the equipment (transportation fleet) required to move the fuel in accordance with the selected schedule.
- Planning and logistics modeling should also consider the following:
  - The possibility of prioritizing shipments of stranded fuel from shutdown reactor sites, which may be complicated by lack of rail access (Sections 3.1.3 and 3.2.2).
  - The regulatory and technical challenges associated with "72-71-72" (storage to transportation to storage) or "72-71-63" (storage to transportation to disposal) sequencing for system component inspections (e.g., canister, fuel, cladding integrity) (Sections 3.2 and 4.4.1)
  - Integration across operational boundaries (i.e., the upstream effects from extended storage and the downstream effects on disposal). This includes integration of thermal management (such as considerations from the thermal lifetime model described in Section C.2.1), criticality, inspections (see previous bullet), and other factors.
  - Socio-economic, socio-political, and safety factors.

DOE-NE IWM has ongoing logistics modeling activities.

C.2.5. Transportation Fleet Design and Certification

The transportation strategy will be enabled by rail line construction, system support, maintenance, and operations, and the design, fabrication, testing, licensing, and acquisition of the needed vehicles and equipment. While a transportation campaign could begin using trucks and the small current fleet of licensed LWT casks, a sizeable fleet of rail rolling stock and transportation casks will be needed to move larger quantities of CSNF (such as those currently loaded in DPCs in dry storage at reactor sites) (BRC 2012, Section 9.3). Specific considerations, supported by ongoing DOE-NE IWM activities, include:

- The necessary transportation fleet includes transportation casks, rail rolling stock (cask cars, buffer cars and escort cars), and heavy-haul trucks and barges at the sites without direct rail access. Each rail shipment includes a "consist" of one escort car, two buffer cars, and one or more cask cars. Estimates of rolling stock needed for different DPC transportation scenarios, summarized in Table E-1, range as high as 10 escort cars, 20 buffer cars, and ~50 cask cars. A larger fleet will be required if smaller standardized canisters are implemented sometime in the future.

- Certified transportation casks exist for about 85% of the current CSNF inventory in dry storage (Section A.2.4); the publication of NRC (2020e) and NRC (2020f) facilitates licensing of transportation casks for high-burnup fuel. However, the existence of storage-only canisters (typically older) complicates the task of eventually transporting the spent fuel in them to a CISF site or to a disposal facility. Also, some transportation equipment still remains to be fabricated, such as spacers and impact limiters for most transportation casks (NWTRB 2019, Section A2.2).
- DOE is developing a 12-axle cask car, the Atlas railcar, for the transportation of CSNF with all NRC-approved transportation cask designs. The Atlas railcar is designed to meet all federal requirements and rail industry guidance for transporting CSNF, including Association of American Railroads (AAR) Standard S-2043 (Areva 2016; Schwab 2019; NWTRB 2019, Sections 2.1 and A2.3). A prototype Atlas railcar has been fabricated (Orano 2019).
- DOE is also planning for the design, fabrication, and testing of an 8-axle railcar, with similar design functionality as the Atlas railcar (Areva 2016) including compliance with AAR Standard S-2043 (Saltzstein et al. 2020, Section 2.1.2).
- There have been suggestions for conducting a full-scale rail cask test, similar to the Package Performance Study (Section 3.2.1), that would further examine extreme transportation accidents (e.g., NAS 2006; BRC 2012).

C.2.6. Security

As with security during extended storage (Section C.1.6), transportation security concerns have become more pronounced since the terrorist attacks of September 11, 2001. The National Academy of Sciences (NAS 2006) noted that "Malevolent acts against spent fuel and HLW shipments are a major technical and societal concern" and that "an integrated evaluation of the threat environment, the response of packages to credible malevolent acts, and operational security requirements for protecting spent fuel and HLW while in transport" should be carried out prior to the commencement of large-quantity shipments to a repository or CISF.

Certain transportation security concerns, especially with respect to layovers in rail yards and routing through large population centers, have not been fully addressed by previous studies (NAS 2006). The use of dedicated trains would help reduce transit times and therefore reduce opportunities for malevolent acts, especially in rail yards. Additional security escorts also could be added more easily to dedicated trains when needed.

NRC has sponsored studies to assess the vulnerability of transportation packages to certain types of terrorist attacks (NAS 2006) and there are ongoing DOE-NE SFWST R&D activities to identify the needs for site-specific physical security and monitoring during extended storage and during transportation.

C.3. Disposal

With the suspension of the Yucca Mountain licensing process, DOE-NE SFWST disposal research has focused on the evaluation of the viability of mined repositories in three generic geologic media, bedded salt, argillite (including clays and shales), and crystalline rock (e.g., Freeze et al. 2013; Mariner et al. 2019). The host lithologies were selected because they have been considered and analyzed as potential repository host rocks both in the US and internationally for several decades. To evaluate criticality consequence approaches for direct disposal of DPCs, generic mined repository concepts in unsaturated alluvium and saturated shale are also being analyzed.

In addition to these mined repository disposal concepts, the viability of deep borehole disposal in generic crystalline basement rock was evaluated for certain waste types (NWTRB 2016; Freeze et al. 2016; Freeze et al. 2019b; Hardin et al. 2019; Kuhlman et al. 2019). DOE-NE SFWST research on deep borehole disposal was discontinued in 2017.

Ongoing DOE-NE SFWST disposal research priorities are summarized in a disposal R&D roadmap (Sevougian et al. 2019c) and a five-year plan for disposal R&D (Sassani et al. 2020). Specific R&D technical areas include (Sassani et al. 2020, Section 2):

- **Argillite Disposal R&D** – Elucidation of the coupled thermal-hydrologic-chemical-mechanical (THCM) processes affecting the performance of a repository in a broad range of fine-grained sedimentary rock types including shales, argillites, and claystones, as well as soft clays (e.g., Jove-Colon et al. 2020). Activities are related to Engineered Barrier System (EBS) R&D and most depend on data from International Collaborations at underground research laboratory (URL) facilities.
- **Crystalline Disposal R&D** – Improve understanding and model representations of processes affecting the performance of crystalline rock repository, which includes a range of lithologies such as metamorphic gneisses, granite, and other igneous rock types. Specific R&D is focused on (1) coupled thermal-hydrologic-mechanical processes affecting flow and transport in fractures, including matrix diffusion, (2) buffer performance, and (3) hydrologic properties of the disturbed rock zone (DRZ) (e.g., Wang et al. 2018).
- **Salt Disposal R&D** – Develop conceptual and numerical models describing the response of natural and engineered features in a salt repository to excavation of the repository and emplacement of heat-generating waste. The primary R&D focus in on (1) measuring the effect of heat-generating waste on salt deformation and brine availability and migration, supported by the large-scale Brine Availability Test in Salt (BATS) (Kuhlman et al. 2020a), and (2) evolution of engineered barriers (e.g., granular salt backfill and cement seals) and DRZ (e.g., Kuhlman et al. 2020b).
- **Geologic Disposal Safety Assessment** – Develop an open source state-of-the-art software framework for probabilistic postclosure performance assessment (PA) analyses of facilities for deep geologic disposal of nuclear waste that can support site selection, site characterization, repository design, and licensing. The GDSA Framework (https://pa.sandia.gov) leverages HPC to integrate concepts, models, and understanding developed in other DOE-NE SFWST disposal R&D technical areas into PFLOTRAN (Lichtner et al. 2019), an open-source multiphase flow and reactive transport simulator, and into Dakota (Adams et al. 2020) for uncertainty quantification and sensitivity analysis

(UQ/SA), optimization, and parameter estimation. Reference cases describing generic repositories in argillite (Mariner et al. 2017; Sevougian et al. 2019a; Sevougian et al. 2019b), salt (Sevougian et al. 2019a), crystalline (Mariner et al. 2016), and unsaturated host rocks (Mariner et al. 2018; Sevougian et al. 2019a; Sevougian et al. 2019b) have been developed provide a platform for integrating concepts, demonstrating capability, and driving development of simulation software and analysis methods, including conceptual models and simulations that account for impacts associated with direct disposal of DPCs.

Longer-term R&D plans are focused on the continued seamless integration of complex process-level models for the generic geologic disposal systems, including expansion of high-fidelity modeling capability in both the near-field (i.e., source-term and EBS transport) and far-field (i.e., the natural barriers) portions of the system model, a biosphere model, and integration of multi-fidelity model forms that adequately represent performance-affecting processes while maximizing computational efficiency. For example, for complex coupled chemistry, development of a drift-based model that includes major chemical components reacting with materials over appropriate temperature ranges is also being pursued.

- **Direct Disposal of Dual-Purpose Canisters** – Investigate the technical feasibility of direct disposal of CSNF in DPCs, as an alternative to repackaging the CSNF into purpose-designed disposal containers. Direct disposal of CSNF in DPCs has the potential to simplify disposal operations, minimize the number of CSNF shipments, reduce occupational worker doses associated with repackaging, and decrease the costs associated with geologic disposal. DPCs tend to be large, heavy, and have a high thermal output; they present technical challenges for disposal operations (e.g., handling and emplacement), thermal management, and postclosure criticality control (Hardin et al. 2015). The potential for higher peak temperatures in generic repository systems is being addressed through analysis of coupled THCM processes in argillite, crystalline, salt, EBS, and GDSA technical areas. Current research is focused on postclosure criticality control, because modern DPCs depend on aluminum-based materials for neutron absorption during storage and transportation, and those materials will degrade in tens to hundreds of years when exposed to groundwater in a repository (SNL 2020; SNL 2021a). Specific research focus areas include: (1) direct disposal without modification (postclosure criticality consequence analysis, as-loaded reactivity margin analysis); (2) modification of already-loaded DPCs (with injectable filler materials for criticality control); and (3) modification of DPCs to be loaded in the future, or the fuel they contain (changing loading maps, adding disposal criticality control features, or basket redesign). These are described in more detail in Section 5.1.2.4.

  Longer-term plans are focused on continued R&D on direct disposal of DPCs with and/or without modification, with consideration of postclosure criticality control, effects of potential modifications on the thermal-chemical evolution of the EBS, and design of DPC disposal overpacks that could enhance postclosure performance.

- **International Collaborations in Disposal Research** – International collaboration enables (1) leveraging a deep knowledge base in regards to alternative repository environments developed across the world, (2) utilizing international research facilities (such as underground research laboratory testing) not available in the US, and (3) sharing the cost of major research efforts such as full-scale in situ experiments or complex modeling efforts. DOE-NE SFWST currently has in place formal collaboration agreements with several international initiatives (DECOVALEX Project, Mont Terri Project, SKB Task Forces, HotBENT, FEBEX-DP) and various international partners (Birkholzer and Faybishenko 2019). National laboratory scientists associated with the campaign are conducting a number of collaborative R&D activities that align with R&D priorities across most of the technical areas discussed in this section.

Longer-term R&D plans are focused on continued participation at a number of international URLs in a range of geologic systems, which provides collaborative access to world-leading URL investigations (i.e., analyses, models, and data for both process-level science and system risk and PA modeling) and peer review by the international scientific community.

- **Engineered Barrier System R&D** – Activities support all three of the generic host-rock types (argillite, crystalline, salt); the specific EBS characteristics differ depending on the host-rock system. Because of this, EBS research activities address a wide range of topics on design materials and barrier performance (e.g., Matteo et al. 2020). In addition, consideration of direct disposal of DPCs expands the consideration of higher temperature conditions and chemical effects from potential additional DPC filler materials within the EBS environment.

  Longer-term R&D plans are focused on engineered barriers (e.g., the waste form including cladding, the waste package, canisters, and backfill) that are common to a number of disposal concepts, with a goal of providing integrated capabilities for analyzing generic repository systems in argillite, crystalline, and salt host rocks, with integrated EBS treatments and GDSA demonstration capabilities. A specific focus is the development and implementation of thermal-chemical dependent models for both waste package degradation and cladding evolution. Processes in glass degradation, as well as potential new waste forms from either ATF or advanced reactor fuel cycle development will also be evaluated.
- **Inventory and Waste Characterization** – Provide the inventory data on DSNF and DHLW for safety assessment analyses of generic mined repositories for these wastes and their waste forms. The inventory data is stored in the Online Waste Library (OWL) (Sassani et al. 2019). R&D on specific waste form characteristics includes assessing the inventory data and ensuring information exists for disposal relevant radionuclides, as well as evaluating the waste form degradation behavior of waste forms for implementation of waste form degradation models (e.g., HLW glass degradation) in the GDSA Framework.
- **Technical Support for Underground Research Laboratory Activities** – The DOE has a shutdown URL that has not been actively operated for approximately 10 years; all offsite electrical utility power has been disconnected remote from the location. Short-term operational objectives are to re-establish lighting and communications within the URL, and to develop planning and facility access protocols that could support R&D pertaining to generic disposal concepts, especially disposal concepts for unsaturated zone repositories.
- **Knowledge Management** – Establish an efficient and rigorous approach to capturing, cataloging, archiving, and transferring institutional nuclear waste management knowledge, with a focus on relevance/benefit to the DOE-NE SFWST R&D program. The loss of senior staff members, who are not only leaders and mentors but also seasoned subject matter experts in nuclear waste management, necessitates the development of tools to archive, transfer, and use institutional tacit knowledge from senior staff members to newer/junior staff.

These core efforts need to have continual integration among the process-level generic geologic systems studies and the PA modeling in the GDSA Framework. Integration across the relevant storage (Section C.1), transportation (Section C.2), and disposal R&D technical areas should lead to consistency in process interfaces and data sets on aspects such as the expected physical condition and characteristics of CSNF, cladding, and storage canisters after they are transported to a repository for disposal (Sassani et al. 2020, Section 4). For example, studies of direct disposal of DPCs integrate the existing stored spent fuel as packaged with disposal studies where models of the engineered barriers and natural system are actively being modified to address conditions based on the DPC characteristics.

Beyond this internal DOE-NE SFWST integration, potential new technology (e.g., ATF and/or advanced fuel cycles) may generate additional/new waste forms and R&D that could be facilitated by coordination with relevant efforts in other DOE-NE Offices.

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APPENDIX D. REGULATIONS

Excerpts from Regulations. CFR Title 10 downloaded on 07/28/2021.

D.1. Reactors

10 CFR Part 50 – Domestic Licensing of Production and Utilization Facilities
(NRC) per Atomic Energy Act of 1954, as amended and Energy Reorganization Act of 1974

[50.2] *Production facility* means:
(1) Any nuclear reactor designed or used primarily for the formation of plutonium or uranium-233; or
(2) Any facility designed or used for the separation of the isotopes of plutonium, except laboratory scale facilities designed or used for experimental or analytical purposes only; or
(3) Any facility designed or used for the processing of irradiated materials containing special nuclear material, except …

[50.2] *Utilization facility* means:
(1) Any nuclear reactor other than one designed or used primarily for the formation of plutonium or U–233; or
(2) An accelerator-driven subcritical operating assembly used for the irradiation of materials containing special nuclear material and described in the application assigned docket number 50–608.

[50.21] Class 104 licenses; for medical therapy and research and development facilities.
[50.22] Class 103 licenses; for commercial and industrial facilities.

[50.34] Contents of applications; technical information.
(a) *Preliminary safety analysis report*. Each application for a construction permit shall include a preliminary safety analysis report.
(b) *Final safety analysis report*. Each application for an operating license shall include a final safety analysis report.

[50.51] Continuation of license.
(a) Each license will be issued for a fixed period of time to be specified in the license but in no case to exceed 40 years from date of issuance. …Licenses may be renewed by the Commission upon the expiration of the period. Renewal of operating licenses for nuclear power plants is governed by 10 CFR part 54.
[50.57] Issuance of operating license.
(a) Pursuant to §50.56, an operating license may be issued by the Commission, up to the full term authorized by §50.51 …

[50.82] Termination of license.
(a) For power reactor licensees –
(3) Decommissioning will be completed within 60 years of permanent cessation of operations. Completion of decommissioning beyond 60 years will be approved by the Commission only when necessary to protect public health and safety. Factors that will be considered by the Commission

in evaluating an alternative that provides for completion of decommissioning beyond 60 years of permanent cessation of operations include unavailability of waste disposal capacity and other site-specific factors affecting the licensee's capability to carry out decommissioning, including presence of other nuclear facilities at the site.

10 CFR Part 54 – Requirements for Renewal of Operating Licenses for Nuclear Power Plants
(NRC) per Atomic Energy Act of 1954, as amended and Energy Reorganization Act of 1974

[54.1] Purpose.
This part governs the issuance of renewed operating licenses and renewed combined licenses for nuclear power plants licensed pursuant to Sections 103 or 104b of the Atomic Energy Act of 1954, as amended, and Title II of the Energy Reorganization Act of 1974.
[54.31] Issuance of a renewed license.
(b) A renewed license will be issued for a fixed period of time, which is the sum of the additional amount of time beyond the expiration of the operating license or combined license (not to exceed 20 years) that is requested in a renewal application plus the remaining number of years on the operating license or combined license currently in effect. The term of any renewed license may not exceed 40 years.
(d) A renewed license may be subsequently renewed in accordance with all applicable requirements.

D.2. Disposal

10 CFR Part 60 – Disposal of High-Level Radioactive Wastes in Geologic Repositories
(NRC) per NWPA 1982, as amended (does not apply to Yucca Mountain)

40 CFR Part 191 – Environmental Radiation Protection Standards for Management and Disposal of Spent Nuclear Fuel, High-Level and Transuranic Radioactive Wastes
(EPA) per NWPA 1982, as amended

10 CFR Part 63 – Disposal of High-Level Radioactive Wastes in a Geologic Repository at Yucca Mountain, Nevada
(NRC) per NWPA 1982, as amended and Energy Policy Act of 1992

40 CFR Part 197 – Public Health and Environmental Radiation Protection Standards for Yucca Mountain, Nevada
(EPA)

D.3. Transportation

10 CFR Part 71 – Packaging and Transportation of Radioactive Material
(NRC) also subject to regulations of US Department of Transportation (DOT)

Subpart A – General Provisions
[71.0] (c) The regulations in this part apply to any licensee authorized by specific or general license issued by the Commission to receive, possess, use, or transfer licensed material, if the licensee delivers that material to a carrier for transport, transports the material outside the site of usage as specified in the NRC license, or transports that material on public highways.
[71.3] Except as authorized in a general license or a specific license issued by the Commission …
[71.4] *Certificate of Compliance* (CoC) means the certificate issued by the Commission under subpart D of this part which approves the design of a package for the transportation of radioactive material.
[71.4] *Packaging* means the assembly of components necessary to ensure compliance with the packaging requirements of this part. It may consist of one or more receptacles, absorbent materials, spacing structures, thermal insulation, radiation shielding, and devices for cooling or absorbing mechanical shocks. The vehicle, tie-down system, and auxiliary equipment may be designated as part of the packaging.
[71.4] *Package* means the packaging together with its radioactive contents as presented for transport.
(1) Fissile material package or Type AF package, Type BF package, Type B(U)F package, or Type B(M)F package means a fissile material packaging together with its fissile material contents.
(2) Type A package means a Type A packaging together with its radioactive contents.
(3) Type B package means a Type B packaging together with its radioactive contents.

Subpart B – Exemptions

Subpart C – General Licenses
[71.17] General license: NRC-approved package.
(a) A general license is issued to any licensee of the Commission to transport, or to deliver to a carrier for transport, licensed material in a package for which a license, certificate of compliance (CoC), or other approval has been issued by the NRC.

Subpart D – Application for Package Approval
[71.31] Contents of application.

Subpart E – Package Approval Standards
[71.41] Demonstration of compliance.
(a) The effects on a package of the tests specified in §71.71 ("Normal conditions of transport"), and the tests specified in §71.73 ("Hypothetical accident conditions"), and § 71.61 ("Special requirements for Type B packages containing more than 10^5 A_2"), must be evaluated by subjecting a specimen or scale model to a specific test, or by another method of demonstration acceptable to the Commission, as appropriate for the particular feature being considered.
[71.43] General standards for all packages.
(f) A package must be designed, constructed, and prepared for shipment so that under the tests specified in §71.71 ("Normal conditions of transport") there would be no loss or dispersal of

radioactive contents, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging.
(g) A package must be designed, constructed, and prepared for transport so that in still air at 38°C (100°F) and in the shade, no accessible surface of a package would have a temperature exceeding 50°C (122°F) in a nonexclusive use shipment, or 85°C (185°F) in an exclusive use shipment.
[71.47] External radiation standards for all packages.
(a) Except as provided in paragraph (b) [exclusive use shipment] of this section, each package of radioactive materials offered for transportation must be designed and prepared for shipment so that under conditions normally incident to transportation the radiation level does not exceed 2 mSv/h (200 mrem/h) at any point on the external surface of the package, and the transport index does not exceed 10.
[71.51] Additional requirements for Type B packages.
(a) A Type B package, in addition to satisfying the requirements of §§ 71.41 through 71.47, must be designed, constructed, and prepared for shipment so that under the tests specified in:
(1) Section 71.71 ("Normal conditions of transport"), there would be no loss or dispersal of radioactive contents – as demonstrated to a sensitivity of 10^{-6} A_2 per hour, no significant increase in external surface radiation levels, and no substantial reduction in the effectiveness of the packaging; and
(2) Section 71.73 ("Hypothetical accident conditions"), there would be no escape of krypton-85 exceeding 10 A_2 in 1 week, no escape of other radioactive material exceeding a total amount A_2 in 1 week, and no external radiation dose rate exceeding 10 mSv/h (1 rem/h) at 1 m (40 in) from the external surface of the package.

Subpart F – Package, Special Form, and LSA-III Tests
[71.71] Normal conditions of transport.
(a) *Evaluation*. Evaluation of each package design under normal conditions of transport must include a determination of the effect on that design of the conditions and tests specified in this section. Separate specimens may be used for the free drop test, the compression test, and the penetration test, if each specimen is subjected to the water spray test before being subjected to any of the other tests.
(c) *Conditions and Tests*. [(1) Heat, (2) Cold, (3) Reduced external pressure, (4) Increased external pressure, (5) Vibration, (6) Water spray, (7) Free drop (from 0.3 - 1.2 m (1 - 4 ft)), (8) Corner drop, (9) Compression, (10) Penetration]
[71.73] Hypothetical accident conditions.
(a) *Test procedures*. Evaluation for hypothetical accident conditions is to be based on sequential application of the tests specified in this section, in the order indicated, to determine their cumulative effect on a package or array of packages. An undamaged specimen may be used for the water immersion tests specified in paragraph (c)(6) of this section.
(c) *Tests*. [(1) Free drop (from 9 m (30 ft)), (2) Crush, (3) Puncture, (4) Thermal, (5) Immersion – fissile material, (6) Immersion – all packages]

D.4. Storage

10 CFR Part 72 – Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste (NRC)

Subpart A – General Provisions
[72.2] (c) The requirements of this regulation are applicable, as appropriate, to both wet and dry modes of storage of –
(1) spent fuel in an independent spent fuel storage installation (ISFSI) and
(2) spent fuel and solid high-level radioactive waste in a monitored retrievable storage installation (MRS) [derived from the NWPA].
[72.3] Certificate of Compliance or CoC means the certificate issued by the Commission that approves the design of a spent fuel storage cask in accordance with the provisions of subpart L of this part.
[72.3] *Independent spent fuel storage installation or ISFSI* means a complex designed and constructed for the interim storage of spent nuclear fuel, solid reactor-related GTCC waste, and other radioactive materials associated with spent fuel and reactor-related GTCC waste storage. An ISFSI which is located on the site of another facility licensed under this part or a facility licensed under part 50 of this chapter and which shares common utilities and services with that facility or is physically connected with that other facility may still be considered independent.
[72.3] *Monitored Retrievable Storage Installation or MRS* means a complex designed, constructed, and operated by DOE for the receipt, transfer, handling, packaging, possession, safeguarding, and storage of spent nuclear fuel aged for at least one year, solidified high-level radioactive waste resulting from civilian nuclear activities, and solid reactor-related GTCC waste, pending shipment to a HLW repository or other disposal.
[72.6] License required; types of licenses.
(a) Licenses for the receipt, handling, storage, and transfer of spent fuel or high-level radioactive waste are of two types: general and specific. Licenses for the receipt, handling, storage, and transfer of reactor-related GTCC are specific licenses. Any general license provided in this part is effective without the filing of an application with the Commission or the issuance of a licensing document to a particular person. A specific license is issued to a named person upon application filed pursuant to regulations in this part.
(b) A general license is hereby issued to receive title to and own spent fuel, high-level radioactive waste, or reactor-related GTCC waste without regard to quantity. Notwithstanding any other provision of this chapter, a general licensee under this paragraph is not authorized to acquire, deliver, receive, possess, use, or transfer spent fuel, high-level radioactive waste, or reactor-related GTCC waste except as authorized in a specific license.
(c) Except as authorized in a specific license and in a general license under subpart K of this part issued by the Commission in accordance with the regulations in this part …
[72.13] Applicability.
(b) The following sections apply to activities associated with a specific license: Secs. 72.1; 72.2(a) through (e); 72.3 through 72.13(b); 72.16 through 72.34; 72.40 through 72.62; 72.70 through 72.86; 72.90 through 72.108; 72.120 through 72.130; 72.140 through 72.176; 72.180 through 72.186; 72.190 through 72.194; and 72.200 through 72.206.
(c) The following sections apply to activities associated with a general license: 72.1; 72.2(a)(1), (b), (c), and (e); 72.3 through 72.6(c)(1); 72.7 through 72.13(a) and (c); 72.30(b), (c), (d), (e) and

(f); 72.32(c) and (d); 72.44(b) and (f); 72.48; 72.50(a); 72.52(a), (b), (d), and (e); 72.60; 72.62; 72.72 through 72.80(f); 72.82 through 72.86; 72.104; 72.106; 72.122; 72.124; 72.126; 72.140 through 72.176; 72.190; 72.194; 72.210 through 72.220, and 72.240(a).
(d) The following sections apply to activities associated with a certificate of compliance: Secs. 72.1; 72.2(e) and (f); 72.3; 72.4; 72.5; 72.7; 72.9 through 72.13(a) and (d); 72.48; 72.84(a); 72.86; 72.124; 72.140 through 72.176; 72.214; and 72.230 through 72.248.

Subpart B – License Application, Form, and Contents
[72.16] Filing of application for specific license.
[72.24] Contents of application: Technical information.
Each application for a license under this part must include a Safety Analysis Report describing the proposed ISFSI or MRS.

Subpart C – Issuance and Conditions of License
[72.40] Issuance of license.
[72.42] Duration of license; renewal
(a) Each license issued under this part must be for a fixed period of time to be specified in the license. The license term for an ISFSI must not exceed 40 years from the date of issuance. The license term for an MRS must not exceed 40 years from the date of issuance. Licenses for either type of installation may be renewed by the Commission at the expiration of the license term upon application by the licensee for a period not to exceed 40 years and under the requirements of this rule.

Subpart D--Records, Reports, Inspections, and Enforcement
[72.70] Safety analysis report updating.
(a) Each specific licensee for an ISFSI or MRS shall update periodically, as provided in paragraphs (b) and (c) of this section, the final safety analysis report (FSAR) to assure that the information included in the report contains the latest information developed.

Subpart E – Siting Evaluation Factors
[72.96] Siting limitations.
(a) An ISFSI which is owned and operated by DOE must not be located at any site within which there is a candidate site for a HLW repository.
(b) An MRS must not be sited in any State in which there is located any site approved for site characterization for a HLW repository. This limitation shall continue to apply to any site selected for construction as a repository.
(c) If an MRS is located, or is planned to be located, within 50 miles of the first HLW repository, any Commission decision approving the first HLW repository application … shall prohibit the storage of a quantity of spent fuel containing in excess of 70,000 metric tons of heavy metal, or a quantity of solidified high-level radioactive waste resulting from the reprocessing of such a quantity of spent fuel, in both the repository and the MRS until such time as a second repository is in operation.
(d) An MRS … may not be constructed in the State of Nevada.

Subpart F – General Design Criteria

Subpart G – Quality Assurance

Subpart H – Physical Protection

Subpart I – Training and Certification of Personnel

Subpart J – Provision of MRS Information to State Governments and Indian Tribes

Subpart K – General License for Storage of Spent Fuel at Power Reactor Sites
[72.210] General license issued.
A general license is hereby issued for the storage of spent fuel in an independent spent fuel storage installation at power reactor sites to persons authorized to possess or operate nuclear power reactors under 10 CFR part 50 or 10 CFR part 52.
[72.212] Conditions of general license issued under §72.210.
(a)(1) The general license is limited to that spent fuel which the general licensee is authorized to possess at the site under the specific license for the site.
(2) This general license is limited to storage of spent fuel in casks approved under the provisions of this part.
(3) The general license for the storage of spent fuel in each cask fabricated under a Certificate of Compliance shall commence upon the date that the particular cask is first used by the general licensee to store spent fuel, shall continue through any renewals of the Certificate of Compliance, unless otherwise specified in the Certificate of Compliance, and shall terminate when the cask's Certificate of Compliance expires.
[72.214] List of approved spent fuel storage casks.

Subpart L – Approval of Spent Fuel Storage Casks
[72.230] Procedures for spent fuel storage cask submittals.
(a) An application for approval of a spent fuel storage cask design must be submitted in accordance with the instructions contained in §72.4. A safety analysis report describing the proposed cask design and how the cask should be used to store spent fuel safely must be included with the application.
(b) Casks that have been certified for transportation of spent fuel under part 71 of this chapter may be approved for storage of spent fuel under this subpart. An application must be submitted in accordance with the instructions contained in §72.4, for a proposed term not to exceed 40 years. A copy of the CoC issued for the cask under part 71 of this chapter, and drawings and other documents referenced in the certificate, must be included with the application. A safety analysis report showing that the cask is suitable for storage of spent fuel, for the term proposed in the application, must also be included.
[72.238] Issuance of an NRC Certificate of Compliance.
A Certificate of Compliance for a cask model will be issued by NRC for a term not to exceed 40 years on a finding that the requirements in §72.236(a) through (i) are met.
[72.240] Conditions for spent fuel storage cask renewal.
(a) The certificate holder may apply for renewal of the design of a spent fuel storage cask for a term not to exceed 40 years.

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APPENDIX E. TRANSPORTATION SYSTEM LOGISTICS

A crucial component of planning for an efficient and resilient transportation system is characterizing the major risks, costs, and other considerations, including both qualitative and quantitative factors, against the associated benefits to determine the appropriate routes and schedules. It is essential that this decision-making occur at a system-wide level to most effectively use resources. One way to address this is with a transportation system logistics model capable of balancing competing priorities to evaluate the impact of operational decisions on metrics such as resource utilization, safety, security, and cost.

Results and observations from existing models are described in Section E.1 (routes), Section E.2 (schedule), Section E.3 (transportation fleet requirements), and Section E.4 (costs). Section E.5 discusses potential improvements to existing modeling approaches to better integrate the broader constraints and objectives of a transportation system.

E.1. Transportation Routes

Transportation routes (e.g., from reactor sites and shutdown sites to a repository, and possibly also to a CISF) cannot be fully identified prior to the identification of a CISF or repository location. While reactor locations are known, distances and infrastructure associated with the transportation of CSNF to a repository and/or a CISF can only be estimated.

As noted in Section 3.2, transportation is most likely to be by rail, especially for the large DPCs.

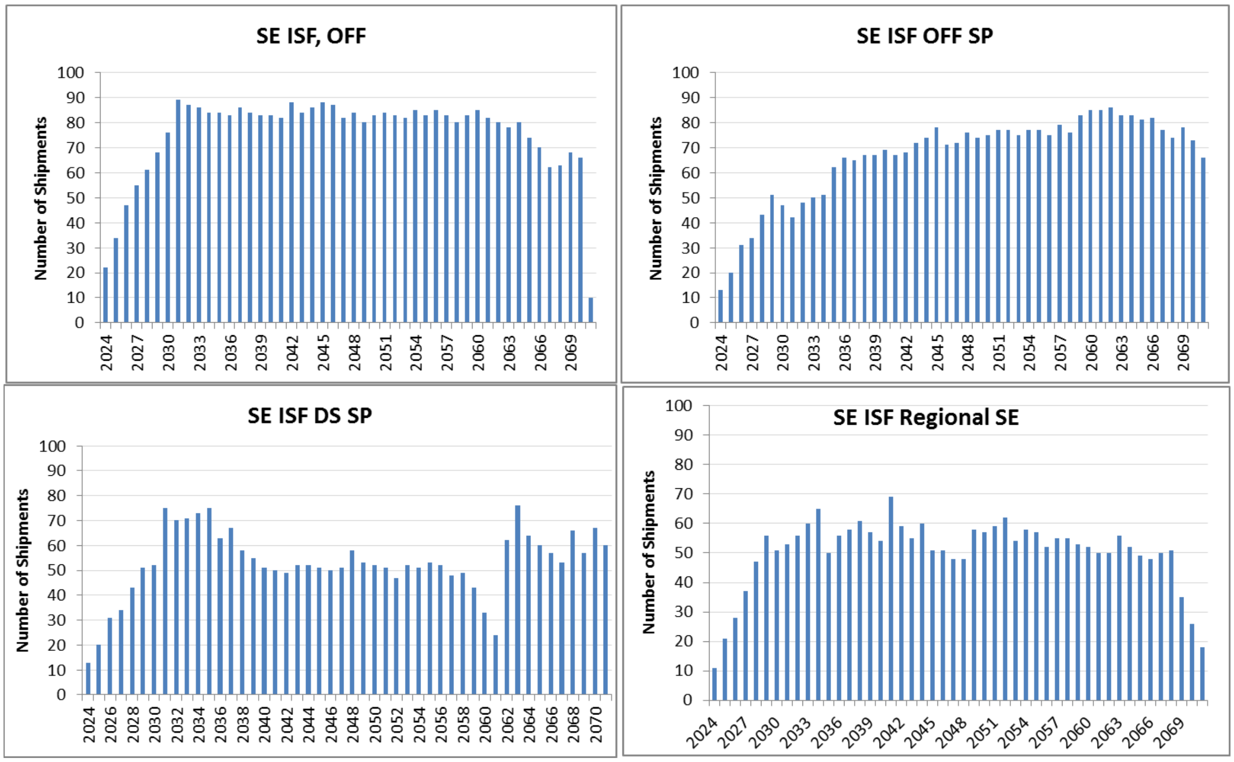
E.2. Transportation Schedule

The annual number of shipments from the reactor sites to a CISF and/or a repository will be significantly affected by the allocation strategy. The allocation strategy in the Standards Contract is the Oldest Fuel First (OFF), whereby sites with older fuel are unloaded first. Each site receives annual acceptance based on this allocation. However, each can decide which fuel to ship within the provided allocation and it is anticipated that the site's selection would be to ship younger fuel first.

It is possible that the Standard Contract could be amended to reflect different allocation strategies. Figure E-1 shows the projected annual number of shipments from 2024 to 2070 for the following four allocation strategies:

- OFF Priority – sites with older fuel are unloaded first (SE ISF OFF) (Figure E-1a)
- Shutdown Reactor Site Priority – fuel at shutdown reactors would have a priority over the OFF sites (SE ISF OFF SP) (Figure E-1b)
- Dry Storage Minimization and Shutdown Reactor Site Priority – give priority to (a) unloading fuel from pools close to their maximum capacities (to avoid loading more fuel in dry storage), and (b) fuel at shutdown reactors (SE ISF DS SP) (Figure E-1c)
- Regional Priority – unload fuel based on the regional considerations (by NRC region) (SE ISF Regional SE) (Figure E-1d)

In all cases it was assumed that SNF was transported by dedicated train to one location (e.g., from reactor sites directly to a repository). Each train includes a “consist” of one or more cask railcars, one escort railcar and two buffer railcars.

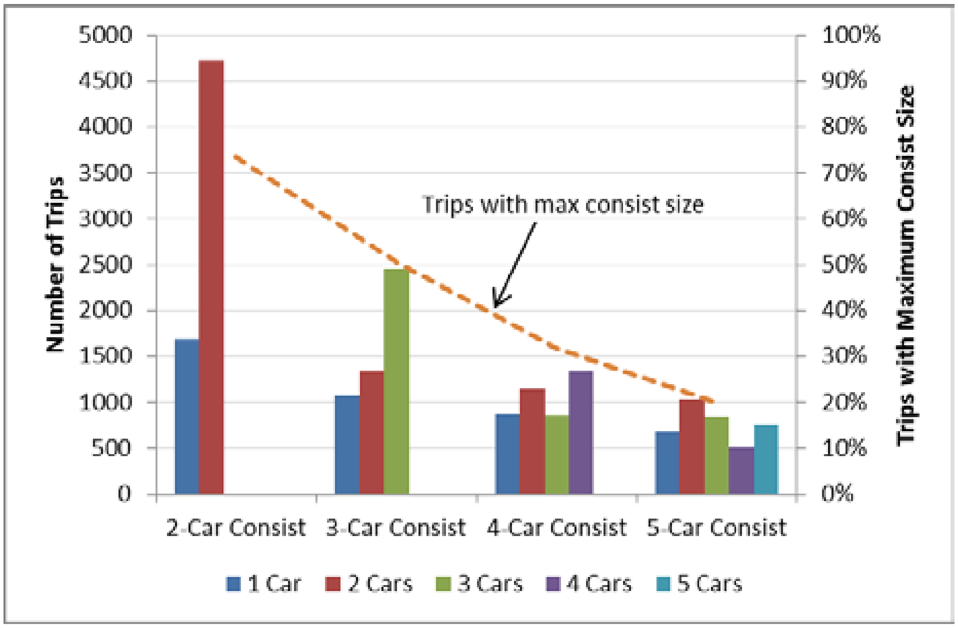


(Source: Connolly et al. 2017)

Figure E-1. Annual Number of Shipments from Reactor Sites to a Repository Under Different Allocation Strategies

The OFF allocation strategy results in a similar number of shipments each year. The Dry Storage Minimization and Shutdown Reactor Site allocation strategies result in greater variability of the annual shipments. While maintaining a steady number of annual shipments seems to be an attractive option, it does not necessarily facilitate the most efficient transportation operations and may not be the most cost effective, as illustrated by the following example.

Figure E-2 shows the number of trips in an OFF transportation campaign for scenarios with four different maximum consist sizes: 2 cask car, 3 cask car, 4 cask car, and 5 cask car. Because of the complexity of the pickup schedules in the OFF allocation strategy, the maximum consist size is achievable in only 74%-77% of the trips for the 2-car maximum consist size scenario, 50% to 59% of trips for the 3-car maximum consist size scenario, 31% to 42% of the trips for the 4-car maximum consist size scenario, and 20% to 31% of the trips for the 5-car maximum consist size scenario.



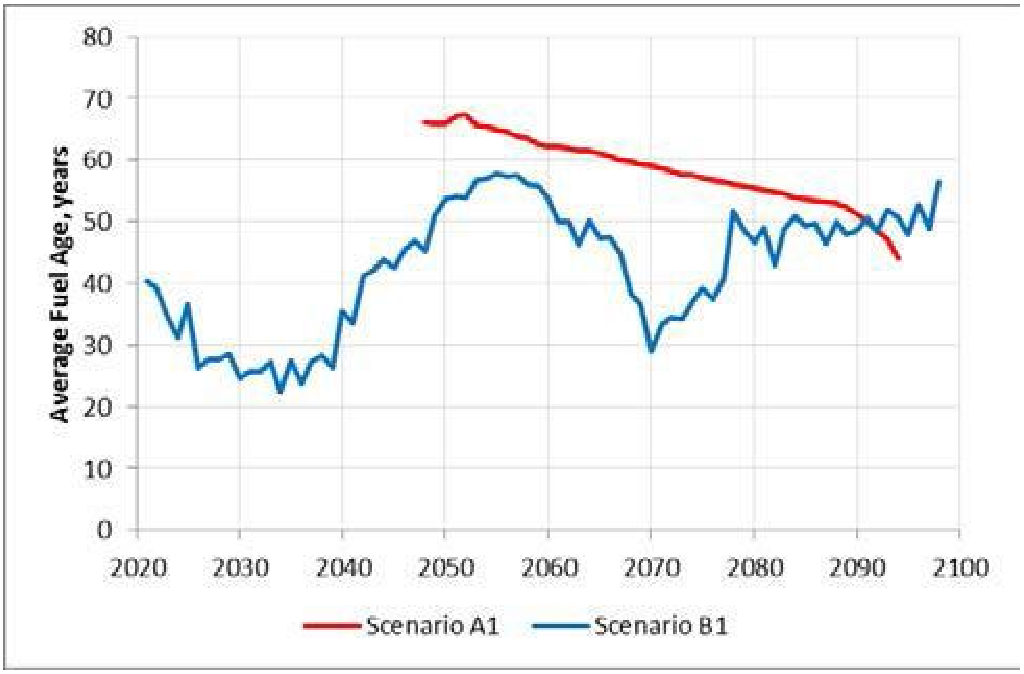
(Source: Kalinina and Busch 2016)

Figure E-2. Number of Trips in in an OFF Allocation Strategy for Scenarios with Different Maximum Consist Sizes

An alternative allocation strategy that would target unloading fewer sites per year while maintaining the same number of annual shipments as OFF would allow for using larger consists (5 cask cars or more) most of the time. This would lead to fewer trips, a smaller fleet, and lower transportation costs.

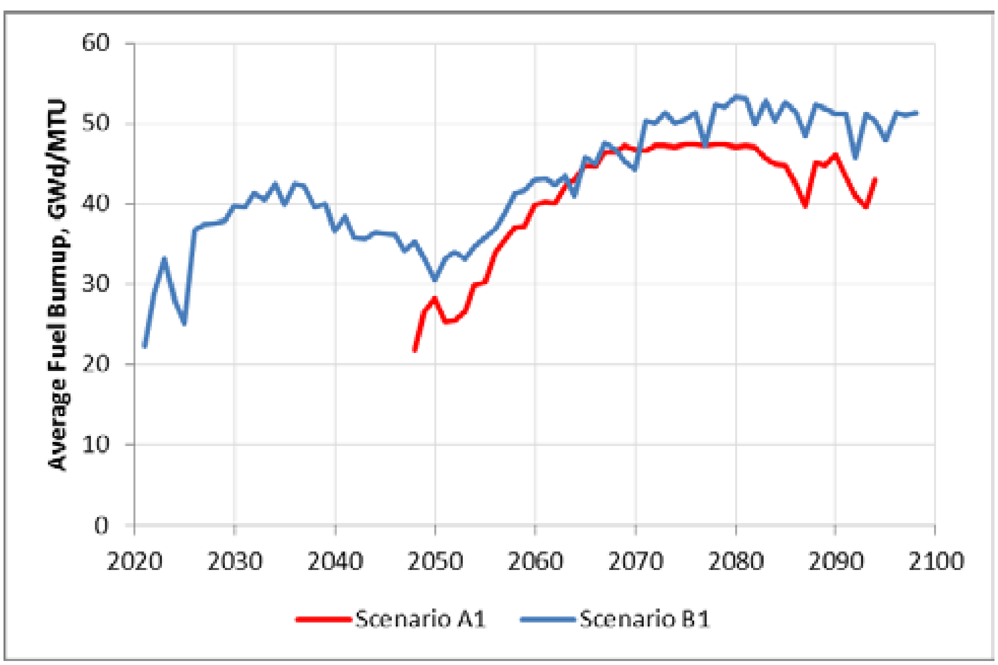
The examples in Figures E-1 and E-2 assume that the reactor sites continue to load DPCs as needed to maintain the pool capacity below maximum. If smaller standardized multi-purpose canisters (e.g., TADs or STADs) are implemented sometime in the future, then the number of trips needed to unload the reactor sites may nearly double. This will increase the transportation cost, but will decrease the disposal costs (e.g., no repackaging costs). Significantly more trips will be also required if the fuel is first transported to a CISF and then to a repository.

Figures E-3 and E-4 compare the fuel age and burnup during the transportation of DPCs under two scenarios: (A1) transport from reactors sites to a repository, when it becomes available (assumed in 2048), and (B1) transport from reactor sites to a CISF (starting in 2020); transport from the CISF to the repository and from reactors sites to the repository, once the repository becomes available (in 2048).



(Source: Kalinina and Busch 2015)

Figure E-3. Average SNF Age During Transportation



(Source: Kalinina and Busch 2015)

Figure E-4. Average SNF Burnup During Transportation

The average fuel age, burnup, and canister heat output during the transportation campaign will change over time. The age and burnup profiles will depend on the amounts of fuel loaded for transportation from pools and dry storage, on the time when a CISF and/or repository become available, and on whether the fuel is directly transported to a repository.

Other factors affecting transportation schedules are related to site specific conditions. There are at least 25 reactor sites that will require heavy-haul truck transport to a nearby rail node. A few reactors sites will require barge transport to the nearest rail node. This will increase the overall travel time required to unload these sites. At the operational sites, the loading of transportation casks may only take place during refueling. This poses the limitations on when during the year loading of the transportation casks may take place and for how long. The weather conditions, such as winter storm, can affect the transportation. The transport at some sites can be limited during the tourist season.

E.3. Transportation Fleet

An important part of the transportation campaign is the acquisition of the vehicles and equipment required to move the fuel in accordance with the selected schedule. The transportation of SNF will be done by dedicated train. Each train includes a "consist" of one or more cask railcars, one escort railcar and two buffer railcars. Table E-1 shows the fleet acquisition needs for the 4 allocation strategies in Figure E-1.

Table E-1. Fleet Acquisition for Allocation Strategies Shown in Figure E-1

Acquisition	Scenario			
	OFF SE	OFF SP SE	DS SP SE	Regional SE
Casks	116	152	192	196
Max casks	11	20	20	20
Buffer railcars	16	14	18	20
Cask railcars	32	38	43	48
Escort railcars	8	7	9	10

(Source: Connolly et al. 2017)

The fleet acquisition is greatly affected by the allocation strategy. The OFF strategy requires 8 consists and 116 transportation casks, whereas the Regional Priority strategy requires 10 consists and 196 transportation casks. Also, a significantly larger fleet will be required if smaller standardized canisters are implemented sometime in the future.

The fleet acquisition is also affected by the number of different types of the transportation casks or overpacks that are required. At present, 25 different transportation casks are needed (9 different BWR and 16 PWR casks), because different utilities use different DPC designs, which typically require different transportation casks. A more uniform cask/canister design would result in a smaller transportation fleet.

The acquisition of the transportation fleet occurs on an annual basis, as needed to move the fuel. However, the transportation casks must be available (i.e., licensed and manufactured) at the time of acquisition. The same is true for the rail rolling stock (cask cars, buffer cars and escort cars), and for the heavy-haul trucks and barges at the sites without direct rail access. Finally, the budget needs to be in place to make these purchases and for the associated capital investments, such as maintenance facilities and/or refurbishments to the infrastructure.

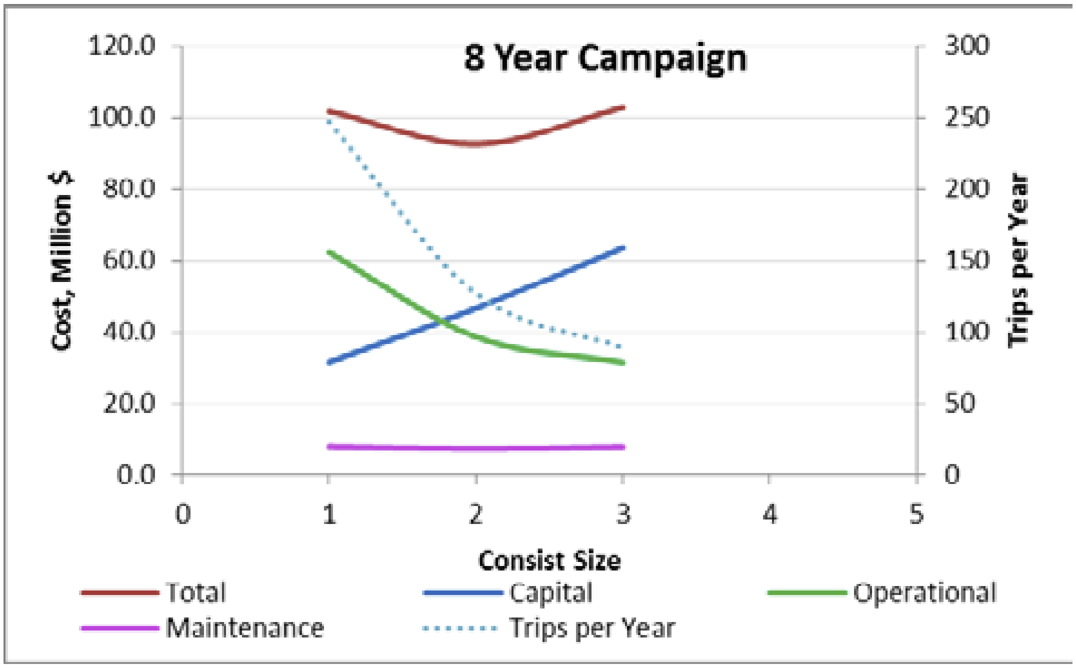
There are different alternatives on how the transportation fleet can be acquired and operated. It can be done by the DOE or a responsible federal organization in charge of transporting SNF, it can be done via contractors, or by some combination of the two. For example, the fleet can be acquired by DOE or leased from a contractor. The fleet can be operated by the DOE or the contractor. The maintenance facilities can be constructed and operated by DOE or by a contractor.

E.4. Cost Considerations

Estimating rail costs is complex. The existing cost estimates are based on an old formula; the parameters in the formula are taken from the Uniform Rail Costing System, which is updated every year. A significant part of the cost is due to switching from one rail company to another along the route. On some routes, there may be 8 or more switches (one way). Also, the cost of shortline rail is high. Shortline rail will be required at many routes to get the fuel from the reactor sites to the nearest rail node. The old formula needs to be revised and the agreement with the major rail companies will have to be established to provide a more realistic cost estimate for rail.

Virtually every contiguous state, except Alaska, will be affected by transportation. The NWPA requires that the states be trained; funding to the states and the training (also known as Section 180(c) costs) should be done prior to the beginning of transportation. The routes need to be selected for this process to begin.

Total costs of a transportation campaign involve tradeoffs between operational, maintenance, capital costs. These tradeoffs are demonstrated for a small 8-year transportation campaign that considers unloading the 9 oldest shutdown reactor sites. Figure E-5 shows the various costs as a function of consist size. Small consist sizes result in lower capital costs (less vehicles are required) and higher operational costs (more trips). Larger consist sizes result in higher capital costs and lower operational costs. The lowest total cost is for an intermediate consist size.



(Source: Kalinina et al. 2013)

Figure E-5. Transportation Costs as a Function of Consist Size

Another example compares the cost of the different unloading schedules for shutdown sites. In the first scenario the shutdown sites are unloaded first and in the second scenario they are unloaded last. While the annual transportation costs for the two scenarios are similar, the at-reactor costs are significantly lower in the first scenario because they go to zero once the site is completely unloaded.

E.5. Improved Transportation System Logistics Model Approaches

Existing transportation system logistics modeling approaches focus on route selection, transportation risk, or operations analysis in isolation. The models are not interrelated, do not allow for optimization (let alone across multiple objectives), and do not incorporate soft constraints.

This section discusses the potential for developing a mathematical transportation system logistics model capable of balancing competing priorities to evaluate the impact of operational decisions on metrics such as resource utilization, safety, security, and cost. The purpose of the mathematical model would be to optimize transportation routes and schedules while considering the broader (e.g. social, economic, and safety) constraints and objectives of the system. This type of interrelated approach has not been accomplished successfully to date, in large part because of the hurdles in adapting the state-of-the-art methodologies in each relevant discipline into a single mathematical framework. A key challenge will be the integration of a network model of the physical components (such as the rail network and sites) with objectives and constraints that factor in social issues such as (1) avoiding disproportionate routing impact on depressed socio-economic areas, (2) political considerations such as the need to prioritize waste removal from certain states,

(3) security and safety issues associated with certain routes or schedules, (4) cost, and (5) the impact of disruptions and uncertainty on those routes or schedules to the other metrics.

Network flow modeling experience at SNL could be helpful in evaluating the impacts of a limited set of uncertainties related to nuclear waste and transportation system logistics decisions by attempting to connect it with flexible multi-objective optimization. This approach can support the balancing of traditional logistics elements and soft constraints, as well as incorporating output from recent safety and security models and socio-economic data to evaluate feasibility and desirability of given routes.

APPENDIX F. SYSTEM LIFE CYCLE COST ESTIMATES FOR SPENT NUCLEAR FUEL MANAGEMENT ALTERNATIVES

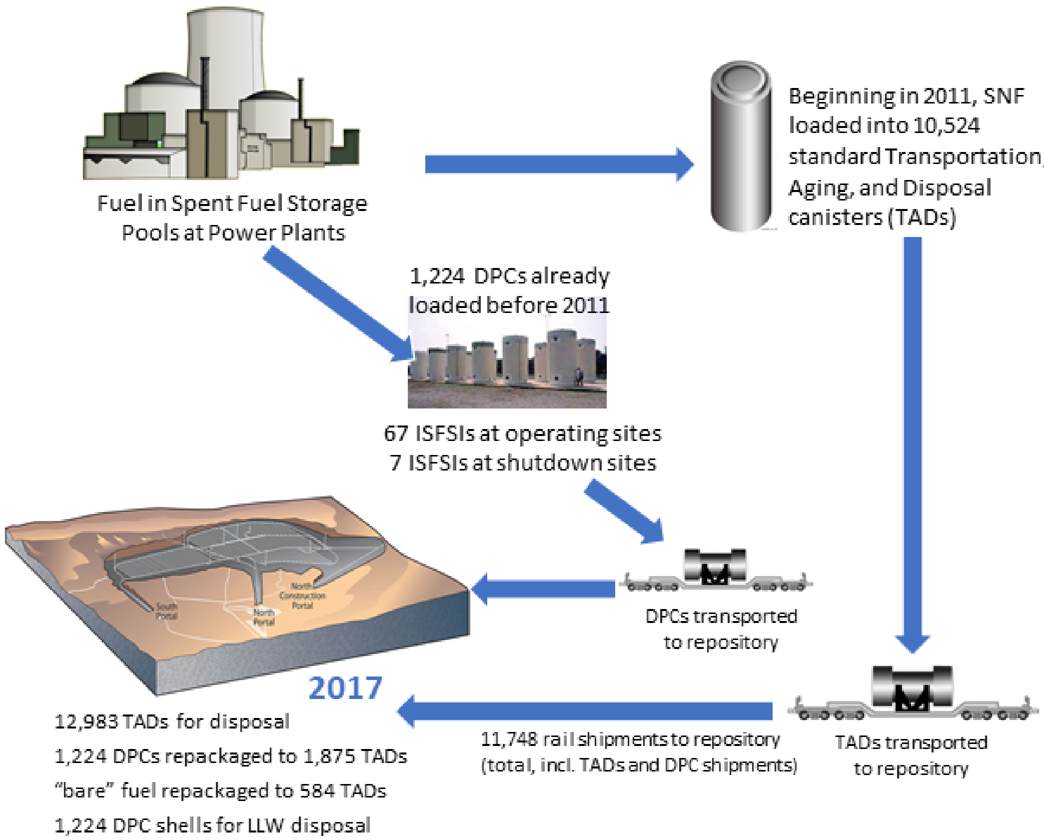
Freeze et al. (2019a) presents simple cost estimates for a set of future spent nuclear fuel management scenarios to provide a basis for examining spent fuel management options during the next century. The cost estimates derive from a total system life cycle cost (TSLCC) analysis for disposal at Yucca Mountain, published in 2008 (DOE 2008b).

The TSLCC estimate, summarized in Section 4.5, includes transportation and disposal of the projected (in 2008) US inventory of commercial (civilian) and defense wastes at the Yucca Mountain repository based on the project-specific TAD-canister-based system design (see Section 3.3.2 and A.2.5.2). For waste receipt and repository operations starting in 2017 and repository closure in 2133, the TSLCC estimate was $96.18 billion (in 2007$) with a commercial (CSNF) cost share allocation of $77.38 billion (in 2007$).

An adjusted baseline Yucca Mountain repository cost estimate was developed by Freeze et al. (2019a, Section 2 and Appendix A) to consider the commercial cost share only, to show costs in constant 2018$[21], and to include costs associated with new information and activities identified since 2008. Consideration of only the commercial cost share allocation, which represents costs for the disposal of CSNF (and the small amount of CHLW), allows the cost comparisons between future scenarios to focus on the impacts of decisions related to the management of CSNF. The resulting Reference Case (also referred to as Scenario 1), shown graphically in Figure F-1, follows the TSLCC baseline schedule (DOE 2008b, Sections 1.3 and 2.2):

- Repository Construction Authorization by the NRC in 2011
- Utilities stop loading CSNF into DPCs and start loading it into TAD canisters in 2011
- Initial waste receipt and start of repository surface and subsurface operations in 2017
- End of 57-year period of waste emplacement in 2073
- End of 50-year period of monitoring with drift ventilation in 2123
- End of 10-year period of closure operations in 2133.

[21] All costs are reported in constant 2018 dollars. Measuring change in the same year constant dollars is a commonly accepted practice to measure real program cost growth because it removes the effects of inflation, which are beyond the control of individual programs. As a result, these cost estimates are considered representative for the purposes of comparative analysis between scenarios, but they should not be taken as formal projections of the life cycle cost for any specific future scenario.



(Source: Freeze et al. 2019a, Figure 2-1)

Figure F-1. Reference Case (Scenario 1) for a Repository Opening in 2017

The Reference Case includes (Freeze et al. 2019a, Section 2):

- Repository operations for a projected inventory of 109,300 MTHM of CSNF. Although the NWPA, as amended, limits the amount of SNF and HLW in Yucca Mountain to 70,000 MTHM (~63,000 MTHM CSNF) prior to the start of operations at a second repository, the Yucca Mountain TSLCC estimate included the entire future US inventory for disposal, projected at the time (in 2008) to be 109,300 MTHM of CSNF. The TSLCC assumption was appropriate because (1) no basis for cost information for a second repository existed, and (2) then-proposed legislation was being considered to remove the 70,000 MTHM limit. As shown in Figure 2-3, current projections estimate a future total of ~136,000 MTHM of CSNF, with 109,300 MTHM CSNF in total inventory by about 2032. Due to the uncertainties in future inventory projections, and to be consistent with the Yucca Mountain TSLCC estimate and allow comparison with it as a baseline, all scenarios assume an inventory of 109,300 MTHM of CSNF is disposed of in a repository similar to Yucca Mountain.
- Disposal of 12,983 TAD-based waste packages (7,978 with PWR assemblies and 5,005 with BWR assemblies). This includes 10,524 TAD canisters loaded and sealed at the utility sites starting in 2011 and shipped to the repository and 2,459 TAD canisters loaded at the repository. For the TAD canisters loaded at the repository, 1,875 come from 1,224 DPCs loaded through May 2011 (StoreFUEL 2011) and shipped from the utility sites (~1.53

TADs/DPC) and 584 come from spent fuel in other types of containers (e.g., casks of uncanistered “bare” fuel) shipped from the utility sites.

- The TAD canisters, with capacities of 21 PWR or 44 BWR assemblies, would be provided to the utilities by the DOE. Upon receipt at the repository, DOE would remove the TAD canisters from transportation casks/overpacks and place them in waste package overpacks suitable for disposal underground.
- Transportation includes a total of 11,748 rail shipments (10,524 TADs and 1,224 DPCs) using dedicated rail rolling stock. A shipment consists of one loaded canister (TAD or DPC) in a cask/overpack transported on a single rail car. Waste pickup uses the “Oldest Fuel First” acceptance priority in the Standard Contract for CSNF acceptance. Nominal acceptance rates for CSNF are 3,000 MTHM/yr.
- The transportation campaign assumes dedicated rolling stock (100 cask cars, 37 buffer cars, and 18 escort cars)[22] and transportation casks and overpacks (~40 CSNF overpacks, ~30 CSNF medium/small casks, ~30 CSNF truck casks).

The Reference Case cost estimate of $112.084 billion (2018$) is summarized in Table F-1. This Reference Case represents the current but suspended policy as a baseline against which spent fuel management alternatives and their costs can be compared. Table F-1 provides an indication of cost category for each cost element, which includes the likely funding source (Nuclear Waste Fund, Judgement Fund, or Other) and the cost type (Common or Discriminating). The likely source of funding depends on a number of variables, most importantly the location (e.g., at the utility sites or at the repository site) and timing (before or after DOE begins to take receipt of the CSNF) of the activities. The funding sources include (Freeze et al. 2019a, Section 2):

- **Nuclear Waste Fund** – This fund, established by the NWPA (Section 1.1), collects or accrues payments from the utilities based on nuclear electricity generation (i.e., at a rate of $0.001/kWh). The utilities in turn collect these funds from their ratepayers. In 2014, in response to a finding by the U.S. Court of Appeals, the rate (ongoing fee) was reduced to zero (OIG 2018, p. 6). Prospective Nuclear Waste Fund expenditures are primarily related to transportation and repository activities. In FY2017, the Nuclear Waste Fund had a balance of $37.7 billion (DOE 2017, pp. 31 and 81), which included interest from investments in U.S. Treasury securities of approximately $1.3 billion for the year (OIG 2018, p. 20), without any additional fees being collected.
- **Taxpayer Liability (Judgment Fund)** – As described in Section 1.1, these are funds paid to the utilities as a result of litigation or settlement agreements due to breach of the Standard Contract. These funds, colloquially known as “Judgment Funds” (31 CFR Part 256; 31 CFR Section 1304) are paid from the US Treasury general fund (i.e., a taxpayer liability) as opposed to appropriated DOE funds or the Nuclear Waste Fund. These costs are primarily related to utility site/ISFSI storage activities; the duration is dependent on the time at which DOE takes receipt of the CSNF. As of 2017, $6.9 billion had been paid from the Judgment Fund, with a remaining liability estimated to be $27.2 billion (DOE 2017, p. 78).

[22] These rolling stock estimates are for transportation of the TSLCC-based inventory of 109,300 MTHM and are correspondingly higher than the rolling stock estimates for 70,000 MTHM provided in Section 5.1.2.5 and Appendix E.

- **Other** – These are costs that are not currently identified as either being paid by the Nuclear Waste Fund or the Judgment Fund. Examples include the loading of TAD canisters at utility sites and LLW disposal of DPCs shells following repackaging. The loading of TAD canisters at utility sites is a cost that was not previously included in the TSLCC, but which will contribute to the total cost of spent fuel management. Utilities presently get compensated from the Judgment Fund for loading DPCs. Loading TAD canisters will have similar operational impacts on the utilities, but it is not clear at this point if such costs will be covered by the Nuclear Waste Fund (as opposed to the Judgment Fund).

Table F-1. Reference Case Scenario Cost Estimates (Billions of 2018$)

COST ELEMENT		COST	COST CATEGORY
REPOSITORY COSTS	**Development and Evaluation (1983-2002)**	**8.099**	NWF Common
	Engineering, Procurement, and Construction (2003-2053)	**17.628**	
	Licensing and Procurements	2.508	NWF Common
	Surface and Subsurface Construction	15.120	NWF Common
	Repository Emplacement Operations (2017-2073)	**26.550**	
	Waste Package Fabrication (12,983 disposal overpacks)	12.232	NWF Discriminating
	DPC Repackaging Operations (1,224 DPCs)	1.829	NWF Discriminating
	LLW Disposal of DPC Shells	0.214	Other
	Repository Operations and Infrastructure	12.276	NWF Common
	Monitoring (2074-2123)	**9.869**	
	Drip Shield Fabrication and Emplacement	7.419	NWF Common
	Monitoring Activities	2.450	NWF Common
	Closure (2124-2133)	**1.352**	NWF Common
	REPOSITORY COSTS TOTAL	**63.498**	
TRANSPORTATION COSTS	**Transportation Operations and Infrastructure (1983-2073)** (11,748 shipments)	**10.022**	NWF Discriminating
	TAD Canister Fabrication (12,983 TADs)	**11.922**	NWF Discriminating
	TRANSPORTATION COSTS TOTAL	**21.944**	
BALANCE OF PROGRAM COSTS	**Development and Evaluation (1983-2002)**	**2.299**	NWF Common
	Program Management (2003-2133)	**8.897**	NWF Common
	BALANCE OF PROGRAM COSTS TOTAL	**11.196**	
OTHER COSTS	**Utility Site/ISFSI Operations (TAD Loading/DPC Transport Preparation)**	**7.226**	Other
	Taxpayer Liability (Judgment Fund)	**8.219**	Judgment Fund
	OTHER COSTS TOTAL	**15.445**	
TOTAL LIFE CYCLE COSTS		**112.084**	

NWF = Nuclear Waste Fund
Column totals may not add due to rounding

(Source: Freeze et al. 2019a, Tables 2-1, 2-2, and A-1)

In addition to the likely funding source, Table F-1 also provides an indication of the cost type (Common or Discriminating) as defined in Freeze et al. (2019a, Section 2):

- **Common Costs** – Costs that are expected to be similar for all scenarios. For the purposes of the simple cost comparisons, these common costs are assumed to be fixed across all scenarios. Common costs are predominantly related to program management and repository development, construction, operation, and closure.
- **Discriminating Costs** – Costs that may vary across scenarios. These costs, and the uncertainty in the input parameters that control them, are responsible for the differences in costs that discriminate between scenarios.

Table F-2 shows the common and discriminating costs for the Reference Case. Nuclear Waste Fund costs may be common or discriminating; Judgment Fund and Other costs are discriminating.

Table F-2. Reference Case Cost Categorization (Billions of 2018$)

COST TYPE	COST ELEMENT	NUCLEAR WASTE FUND	JUDGMENT FUND	OTHER	TOTAL
Common	Repository (Development and Evaluation)	8.099			8.099
	Repository (Engineering, Procurement, Construction)	17.628			17.628
	Repository (Operations and Infrastructure)	12.276			12.276
	Repository (Monitoring)	9.869			9.869
	Repository (Closure)	1.352			1.352
	Balance of Program	11.196			11.196
	Total Common Costs	**60.420**	**0.000**	**0.000**	**60.420**
Discriminating	Repository (Disposal Overpack Fabrication)	12.232			12.232
	TAD Canister Fabrication	11.922			11.922
	Utility/ISFSI Taxpayer Liability (Judgment Fund)		8.219		8.219
	Utility/ISFSI Packaging (TAD Loading/DPC Preparation)			7.226	7.226
	Transportation (Operations and Infrastructure)	10.022			10.022
	Repository Packaging (DPC Repackaging)	1.829			1.829
	Repository Packaging (LLW Disposal)			0.214	0.214
	Total Discriminating Costs	**36.005**	**8.219**	**7.440**	**51.664**
	TOTAL COSTS	**96.425**	**8.219**	**7.440**	**112.084**

Column totals may not add due to rounding

(Source: Freeze et al. 2019a, Table 2-2)

The Reference Case (Scenario 1) serves as a useful baseline for comparison purposes by providing detailed cost estimates for what might have been had the Yucca Mountain project proceeded as planned, with initial waste receipt and start of emplacement operations in 2017. The future spent fuel management alternative scenarios (Freeze et al. 2019a, Section 3), which derive from the Reference Case, are summarized in Table F-3.

Table F-3. Summary of Scenarios and Variants

NO.	SCENARIO	UTILITY SITE ACTIVITIES	REPOSITORY ACTIVITIES
Disposal of 109,300 MTHM of CSNF at Yucca Mountain (YM) in 2017 (Reference Case)			
1	**YM Baseline (Adjusted TSLCC)** Start loading TADs in 2011 Repackage DPCs at repository Taxpayer liability (DPCs) to 2011 Taxpayer liability (ISFSIs) to 2027	1,224 DPCs loaded ≤ 2011 10,524 TADs loaded ≥ 2011 11,748 rail shipments to repository 67 ISFSIs at operating sites 7 ISFSIs at shutdown sites	12,983 TADs for disposal 0 DPCs for disposal “bare” fuel repackaged to 584 TADs 1,224 DPCs repackaged to 1,875 TADs 1,224 DPC shells for LLW disposal
Disposal at Yucca Mountain (YM) in 2031			
2	**YM Delayed to 2031** Start loading TADs in 2025 Repackage DPCs at repository Taxpayer liability (DPCs) to 2025 Taxpayer liability (ISFSIs) to 2041	3,900 DPCs loaded ≤ 2025 6,423 TADs loaded ≥ 2025 10,323 rail shipments to repository 46 ISFSIs at operating sites 28 ISFSIs at shutdown sites	12,983 TADs for disposal 0 DPCs for disposal “bare” fuel repackaged to 584 TADs 3,900 DPCs repackaged to 5,976 TADs 3,900 DPC shells for LLW disposal
2B	**Variant of Scenario 2** Directly dispose DPCs loaded ≤ 2025	[same as Scenario 2]	7,007 TADs for disposal 3,900 DPCs for disposal 3,900 DPCs modified for disposal 0 DPC shells for LLW disposal
2C	**Variant of Scenario 2B** Start loading larger DPCs ≥ 2025 Directly dispose DPCs loaded ≤ 2043 Taxpayer liability (DPCs) to 2043	7,305 DPCs loaded 0 TADs loaded 7,305 rail shipments to repository	7,649 DPCs for disposal “bare” fuel repackaged to 344 DPCs 7,305 DPCs modified for disposal 0 DPC shells for LLW disposal
2D	**Variant of Scenario 2** CISF available in 2025 Store DPCs and TADs at CIS Taxpayer liability (ISFSIs) to 2035	10,323 rail shipments to CISF 10,323 rail shipments to repository 55 ISFSIs at operating sites 19 ISFSIs at shutdown sites	[same as Scenario 2]
Disposal at Yucca Mountain (YM) in 2041			
3	**YM Delayed to 2041** Start loading larger DPCs ≥ 2025 Start loading TADs in 2035 Repackage DPCs at repository Taxpayer liability (DPCs) to 2035 Taxpayer liability (ISFSIs) to 2051	5,812 DPCs loaded ≤ 2035 2,534 TADs loaded ≥ 2035 8,346 rail shipments to repository 31 ISFSIs at operating sites 43 ISFSIs at shutdown sites	12,983 TADs for disposal 0 DPCs for disposal “bare” fuel repackaged to 584 TADs 5,812 DPCs repackaged to 9,865 TADs 5,812 DPC shells for LLW disposal
3A	**Variant of Scenario 3** CISF available in 2025 Store DPCs and TADs at CIS Taxpayer liability (ISFSIs) to 2035	8,346 rail shipments to CISF 8,346 rail shipments to repository 55 ISFSIs at operating sites 19 ISFSIs at shutdown sites	[same as Scenario 3]
Disposal at Yucca Mountain (YM) in 2117 (Extended Storage)			
4	**YM-Like Repository in 2117** Start loading larger DPCs ≥ 2025 Repackage DPCs at repository Taxpayer liability (DPCs) to 2043 Taxpayer liability (ISFSIs) to 2127	7,305 DPCs loaded 0 TADs loaded 7,305 rail shipments to repository 0 ISFSIs at operating sites 74 ISFSIs at shutdown sites	12,983 TADs for disposal 0 DPCs for disposal “bare” fuel repackaged to 584 TADs 7,305 DPCs repackaged to 12,399 TADs 7,305 DPC shells for LLW disposal
4A	**Variant of Scenario 4** Directly dispose DPCs loaded ≤ 2043	[same as Scenario 4]	7,649 DPCs for disposal “bare” fuel repackaged to 344 DPCs 7,305 DPCs modified for disposal 0 DPC shells for LLW disposal
4B	**Variant of Scenario 4** CISF available in 2025 Store DPCs at CIS Taxpayer liability (ISFSIs) to 2035	7,305 rail shipments to CISF 7,305 rail shipments to repository 55 ISFSIs at operating sites 19 ISFSIs at shutdown sites	[same as Scenario 4]

(Source: Freeze et al. 2019a, Table 3-1)

The selected set of scenarios represent a range of possible combinations of alternative spent fuel management approaches over the next century. The future alternative scenarios each make assumptions about the timing of repository availability and provide choices about integration between storage, transportation, and disposal that combine one or more of: direct disposal of CSNF in DPCs without repackaging, repackaging of existing CSNF and/or loading future CSNF into disposal-ready canisters, and extended dry storage of CSNF at utility sites and/or a CISF.

The future alternative scenarios are constructed around three representative dates for the first receipt of spent fuel at the repository: 2031, which corresponds to an early date for the opening of Yucca Mountain should licensing activities resume immediately (Scenario 2); 2041, which represents an additional ten-year delay in restarting Yucca Mountain (Scenario 3); and 2117, which represents a 100-year delay in the repository program from the original 2017 date assumed in the Yucca Mountain TSLCC (Scenario 4). These dates are chosen simply for the purpose of investigating relative cost impacts associated with delay and should not be interpreted as more or less likely than other dates.

Variants (i.e., "one-off" sub-scenarios) within these scenarios examine the relative cost impacts of various decisions regarding repackaging of spent fuel from DPCs into TAD canisters and/or modifying repository operations (and licensing requirements) to allow for direct disposal of DPCs without repackaging. Cost impacts of having a federal CISF available in 2025, thereby reducing taxpayer liabilities paid through the Judgment Fund, are also considered.

Assumptions that apply to the cost estimates for future alternative scenarios are consistent with the Reference Case assumptions. Key assumptions include (Freeze et al. 2019a, Section 3.4):

- Disposal of the TSLCC-based inventory (109,300 MTHM of CSNF) at Yucca Mountain in TAD canisters and/or DPCs
- None of the alternative scenarios explore repository geologies or designs other than the Yucca Mountain design outlined in the TSLCC (DOE 2008b). Scenarios 4, 4A, and 4B include some extra costs for "redevelopment" of the Yucca Mountain design and licensing basis that would likely be necessary following a 100-year delay.
- The repository footprint is the same for all alternatives. The number of drifts, drift lengths, and associated drip shields are the same for each alternative because they are a function of thermal load, not number of waste packages.
- The cost for a TAD disposal overpack is the same as that for a DPC disposal overpack.
- Additional costs that may be associated with extended storage up to 100 years (e.g., remediation or replacement of storage system components), or to transport DPCs after extended storage, are not included.
- The cost for a rail shipment for a TAD canister is the same as that for a DPC. A shipment consists of one loaded canister (TAD or DPC) in a cask/overpack transported on a single rail car.
- DPCs existing as of 2025 have capacities of 32 PWR or 68 BWR assemblies. DPCs loaded from 2025 onward have capacities of 37 PWR or 89 BWR assemblies.
- TAD canisters have capacities of 21 PWR or 44 BWR assemblies.

- Costs associated with modifications that may be required to existing or planned wet handling facilities for spent fuel at utility sites or the repository are not included.
- Costs for a federal CIS facility include design, construction, staging pads, overpacks, storage modules, maintenance, operations, and security. Transportation of TAD canisters and/or DPCs to the CIS facility and subsequent transportation to the repository are captured as part of the transportation cost element. The CIS costs are based on an operational lifetime of 40 years, which is consistent with a 40-year initial license for an ISFSI. For scenarios where the CIS facility lifetime exceeds 40 years, additional annual operating costs are applied, and license renewals in 40-year increments are assumed.

While not comprehensive, the alternative scenarios span a representative range of spent fuel management options that provide useful information to examine (1) the technical integration needs between direct disposal of DPCs, repackaging/loading CSNF into disposal-ready canisters, and extended dry storage including the possibility of a CISF, and (2) the comparative costs associated with each of the options.

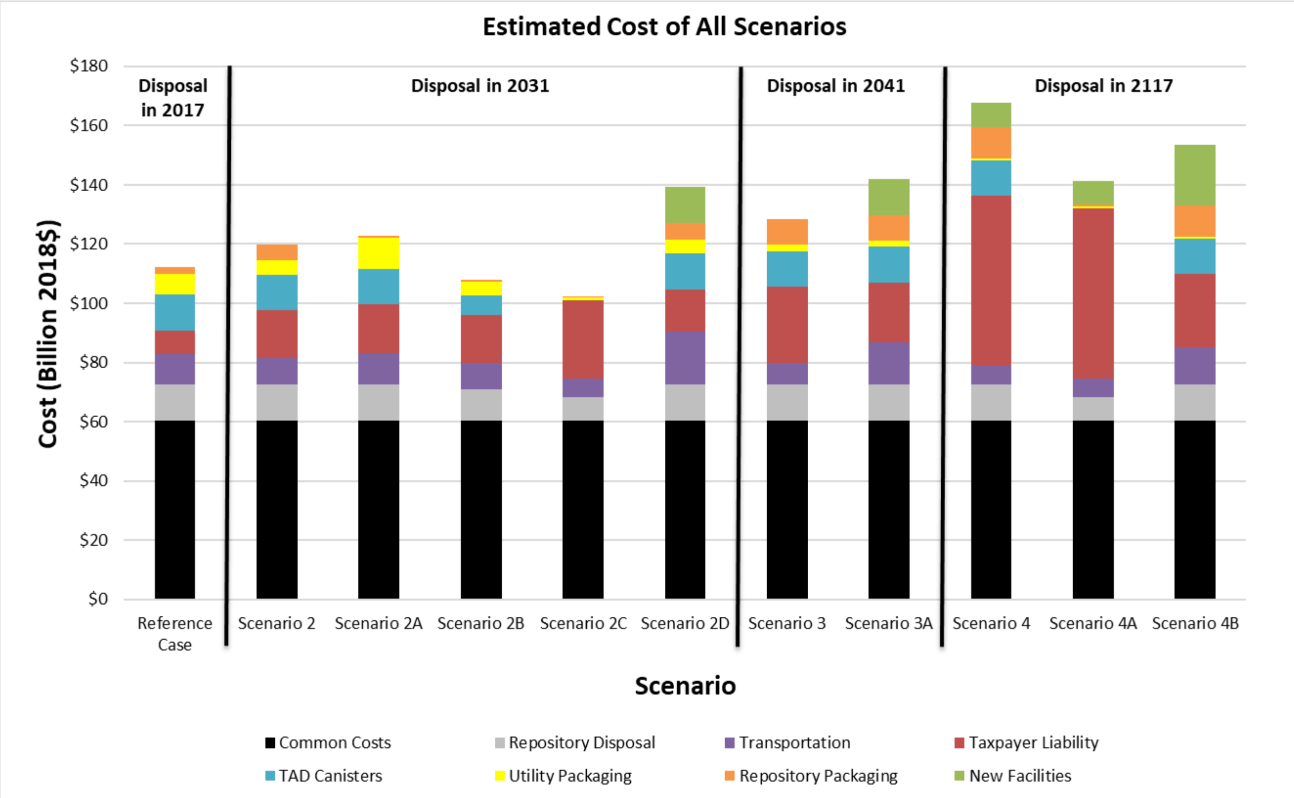
It should be noted that, for the purpose of the cost comparisons, Yucca Mountain is assumed to be the repository for all of the alternative scenarios. Although there are likely to be cost differences between repository systems in different geologic media (e.g., salt, argillite, crystalline) (e.g., Hardin 2016; Hardin and Kalinina 2016), they were not considered in the cost comparisons. While certain geologies favor certain repository characteristics (e.g., hard rock unbackfilled or salt for thermal constraints (see Figure 4-2)), similar observations about comparative costs would likely arise from a repository in other geologic media.

The cost estimates for each of the future spent nuclear fuel management scenarios all have the same common costs but have varying discriminating costs. The differences in the discriminating costs between scenarios are driven by decisions about repository timing, canister selection (e.g., direct disposal of DPCs and/or repackaging), and a CISF. To facilitate the cost comparisons between scenarios, the discriminating costs, identified in Table F-2 for the Reference Case, are grouped as follows (Freeze et al. 2019a, Section 4):

- **Repository Disposal** – These are in addition to the common costs for repository development, operation, and closure:
  - Cost of modifying DPCs to facilitate direct disposal (i.e., added criticality controls)
  - Cost of disposal overpacks (waste packages)
- **TAD Canisters** – TAD canisters may be loaded at utility sites and/or the repository:
  - Cost of TAD canister fabrication
- **Utility/ISFSI Taxpayer Liability (Judgment Fund)** – These costs, primarily associated with dry storage at utility sites/ISFSIs, are dependent on the time at which DOE takes receipt of the CSNF and on the time at which reactors at shutdown sites are decommissioned (at which point annual ISFSI maintenance costs increase):
  - Cost of loading CSNF from a pool into a DPC
  - Annual operations cost per ISFSI-only (shutdown) site
  - Annual operations cost per ISFSI at operating reactor site

- **Utility/ISFSI Packaging** – These costs are not covered by payments from the Judgment Fund:
    - Cost of loading CSNF from a pool into a TAD
    - Cost of unloading CSNF already stored in a DPC and loading it into a TAD
    - Cost of storage overpack for TAD
    - Cost of transferring a TAD or a DPC from a storage cask to a transportation cask
- **Transportation** – Operations and infrastructure:
    - Cost of rail shipments
- **Repository Packaging** – These costs are for repackaging CSNF from DPCs to TAD canisters; costs of disposal overpacks are captured under Repository Disposal
    - Cost of unloading CSNF from a DPC and loading it into a TAD
    - LLW disposal cost of DPC shells
- **New Facilities** – These costs apply to some, but not all, scenarios:
    - Cost for construction and operation of a federal CISF (Scenarios 2D, 3A, and 4B)
    - Re-incurred cost for repository development and evaluation 100 years in the future (Scenarios 4, 4A, and 4B)

The estimated total life cycle costs for each of the scenarios and variants is shown graphically in Figure F-2 and detailed in Table F-4.



(Source: Freeze et al. 2019a, Figure 4-1)

Figure F-2. Estimated Costs for All Scenarios

Table F-4. Summary of Estimated Costs for All Scenarios (Billions of 2018$)

Cost Element	1	2	2B	2C	2D	3	3A	4	4A	4B
Transportation	10.022	8.807	8.807	6.232	17.615	7.121	14.241	6.232	6.232	12.465
TAD Canisters	11.922	11.922	6.435	0.000	11.922	11.922	11.922	11.922	0.000	11.922
Repository Packaging	2.043	5.562	0.434	0.434	5.562	8.547	8.547	10.621	0.434	10.621
Utility Packaging	7.226	4.726	4.726	0.731	4.726	2.292	2.292	0.731	0.731	0.731
Taxpayer Liability (Judgment Fund)	8.219	16.307	16.307	26.521	14.515	25.869	20.190	57.480	57.480	24.729
Repository Disposal	12.232	12.232	10.666	7.937	12.232	12.232	12.232	12.232	7.937	12.232
New Facilities	0.000	0.000	0.000	0.000	12.290	0.000	12.142	8.099	8.099	20.518
Common Costs	60.420	60.420	60.420	60.420	60.420	60.420	60.420	60.420	60.420	60.420
Total	**112.084**	**119.975**	**107.795**	**102.275**	**139.281**	**128.403**	**141.986**	**167.737**	**141.333**	**153.637**

Based on comparative cost analyses between the scenarios in Table F-4, the following conclusions are made (Freeze et al. 2019a, Section 5):

- Scenarios that delay the beginning of disposal operations, with all other elements of the system unchanged, increase estimates of total life cycle cost due to ongoing taxpayer liability associated with continued storage at utility sites, paid from the Judgment Fund.
  - Delaying disposal for 14 years to 2031 (Scenario 2) adds about $8 billion (2018$) to the total cost as compared to the Reference Case (Scenario 1), delaying another 10 years to 2041 (Scenario 3) adds another $8 billion (2018$).
  - Doing nothing and delaying disposal for 100 years (Scenario 4) is the most expensive option, costing the taxpayers nearly $50 billion (2018$) in additional payments from the Judgment Fund as compared to the Reference Case. This increase includes about $15 billion (2018$) for loading more DPCs and about $35 billion (2018$) for continued operation of ISFSIs at shutdown sites.
- Scenarios that allow direct disposal of DPCs without repackaging to TAD canisters reduce estimated life cycle costs.
  - For a repository that opens in 2031, directly disposing of all CSNF in DPCs (Scenario 2C) has the potential to reduce costs by approximately $18 billion (2018$) as compared to repackaging all CSNF into TAD canisters (Scenario 2).
- The relative cost impact of implementing a federal CIS facility depends on the date at which the repository begins disposal operations. If a repository is available relatively soon after the CIS facility begins operations, costs for construction and operation of the CIS facility and for transportation are greater than the savings associated with the earlier termination of Judgment Fund liabilities (Scenarios 2D and 3A). If disposal operations are delayed for a longer period, the earlier termination of Judgment Fund liabilities from a CIS facility can lead to overall cost savings (Scenario 4B).
- Decisions about where spent nuclear fuel is repackaged for disposal (i.e., at the commercial nuclear power plants or at the repository) result in significant changes in where in the system costs are incurred, but the impact on overall total life cycle costs is less important than other factors considered in the analysis.
- Cost estimates are relatively insensitive to uncertainty in component costs. Uncertainty in costs to the taxpayer from Judgment Fund liabilities cause the costs associated with lengthy delays before disposal and prolonged ISFSI operations to increase more than those in other scenarios. Otherwise, uncertainty in costs is not a discriminator among the scenarios.
- The primary funding source for all scenarios is the Nuclear Waste Fund. Taxpayer liability, in the form of payments from the Judgment Fund, increases in scenarios where there is a delay opening a repository.

Analysis of these costs are provided in Freeze et al. (2019a) and summarized in Section 5.3.

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