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Industry Spent Fuel Storage Handbook

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EPRI Project Manager
J. Kessler

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ELECTRIC POWER RESEARCH INSTITUTE
3420 Hillview Avenue, Palo Alto, California 94304-1338 ▪ PO Box 10412, Palo Alto, California 94303-0813 ▪ USA
800.313.3774 ▪ 650.855.2121 ▪ askepri@epri.com ▪ www.epri.com

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Energy Resources International, Inc.
1015 18th St., NW
Suite 650
Washington, DC 20036

Principal Investigator
E. Supko

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ABSTRACT

The *Industry Spent Fuel Storage Handbook* ("the Handbook") addresses the relevant aspects of at-reactor spent (or used) nuclear fuel (SNF) storage in the United States. With the prospect of SNF being stored at reactor sites for the foreseeable future, it is expected that all U.S. nuclear power plants will have to implement at-reactor dry storage by 2025 or shortly thereafter. The Handbook provides a broad overview of recent developments for storing SNF at U.S. reactor sites, focusing primarily on at-reactor dry storage of SNF. The Handbook provides an overview of current regulations and regulatory guidance for dry storage of SNF; descriptions of dry storage technologies available in the United States, and an overview of the process for planning and implementing an at-reactor storage facility.

Keywords

Spent fuel storage and transportation
Spent nuclear fuel storage
Used nuclear fuel storage

LIST OF ACRONYMS

10CFR50	Title 10, U.S. Code of Federal Regulations, Part 50
10CFR71	Title 10, U.S. Code of Federal Regulations, Part 71
10CFR72	Title 10, U.S. Code of Federal Regulations, Part 72
AFR	Away-From-Reactor
ALARA	As Low As Reasonably Achievable
ALJ	Administrative Law Judge
ANO	Arkansas Nuclear One
ASME	American Society of Mechanical Engineers
BFS	BNFL Fuel Solutions
BWR	Boiling Water Reactor
CEA	Control Element Assemblies
CHM	Container Handling Machine
CMS	Computational Modeling Software
CoC	Certificate of Compliance
CP&L	Carolina Power and Light Company
DOE	Department of Energy
DPC	Dual-Purpose Cask, or Dual-Purpose Canister
DSC	Dry Shielded Canister
EA	Environmental Assessment

EIS	Environmental Impact Statement
EPRI	Electric Power Research Institute
ER	Environmental Report
EIS	Environmental Impact Statement
FCR	Full Core Reserve
FONSI	Finding of No Significant Impact
FSAR	Final Safety Analysis Report
GISF	Generic Interim Storage Facility
GTCC	Greater-than-Class C Low-Level Radioactive Waste
GWd/MTU	Giga-Watt-days/Metric Ton of Uranium
HLW	High-level Radioactive Waste
HRS	Hydraulic Ram System
HSM	Horizontal Storage Module
HTGR	High Temperature Gas-Cooled Reactor
IGSCC	Intergranular Stress-Corrosion Cracking
IIP	Integrated Inspection Plan
IMC	Inspection Manual Chapter
INL	Idaho National Laboratory
IP	Inspection Procedure
ISFSI	Independent Spent Fuel Storage Installation
ITTR	Instrument Tube Tie Rod
ISG	Interim Staff Guidance
LA	License Application

LLW	Low-Level Radioactive Waste
MAGNASTOR	Modular Advanced Generation, Nuclear All-Purpose Storage
MPC	Multi-Purpose Canister
MPUC	Minnesota Public Utility Commission
MRS	Monitored Retrievable Storage
MSB	Multi-assembly Sealed Basket
MTC	MSB Transfer Cask
MVDS	Modular Vault Dry Store
NAC	NAC International, Inc.
NEI	Nuclear Energy Institute
NFBC	Non-Fuel Bearing Components
NPV	Net Present Value
NRC	Nuclear Regulatory Commission
NUHOMS	Nutech Horizontal Modular Storage
NYSERDA	New York State Energy Research and Development Authority
OCA	Owner Controlled Area
OCRWM	Office of Civilian Radioactive Waste Management
PGE	Portland General Electric
PSB	Public Service Board
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAI	Request for Additional Information
RSI	Request for Supplemental Information

SAR	Safety Analysis Report
SER	Safety Evaluation Report
SFST	Division of Spent Fuel Storage and Transportation
SRP	Standard Review Plan
SNF	Spent Nuclear Fuel, also referred to as used nuclear fuel or irradiated fuel
SSC	Systems, Structures and Components
TC	Transfer Cask
TLAA	Time-Limited Aging Analyses
TSAR	Topical Safety Analysis Report
TSC	Transportable Storage Canister
UMS	Universal MPC System
VCC	Ventilated Concrete Cask
VVM	Vertical Ventilated Module

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1
INTRODUCTION

This Industry Spent Fuel Storage Handbook (Handbook) addresses the relevant aspects of at-reactor spent (or used) nuclear fuel (SNF)[1] storage. The purpose of this document is to provide a resource to nuclear operating companies during the planning process for and operation of at-reactor SNF storage expansion projects, specifically dry storage of SNF in Independent Spent Fuel Storage Installations (ISFSI). This document may also be used as a resource for other interested members of the public or stakeholders who wish to better understand SNF storage issues. While this Handbook focuses on the experience and planning associated with the implementation of at-reactor dry storage facilities, information is also provided regarding SNF storage pool reracking projects and other in-pool storage expansion alternatives. The Handbook also addresses the planning of and regulatory requirements for the development of an away-from-reactor central interim storage facility.

Section 2 of this report provides a description of U.S. nuclear operating company experience with in-pool SNF storage expansion technologies, such as reracking and rod consolidation, and with dry storage of SNF at both operating and shutdown reactor sites.

Section 3 provides an overview of the regulations and regulatory guidance associated with SNF dry storage and transportation technologies. This includes an overview of the site-specific and general license processes; the Certificate of Compliance (CoC) process for storage and transportation; U.S Nuclear Regulatory Commission (NRC) guidance documents such as Standard Review Plans and Regulatory Guides; and implementation of NRC regulations for at-reactor storage.

Section 4 provides a brief overview of the alternative technologies for at-reactor storage of SNF. The focus of this section is on dry storage technologies that have been approved by the NRC under the site-specific or general license process, that are under NRC review, or that utilities have firm plans to use for dry storage.

Section 5 summarizes the staffing and scheduling requirements associated with the initial planning and implementation of a dry storage facility.

Section 6 examines the types of considerations that are generally addressed by nuclear operating companies when evaluating at-reactor SNF expansion alternatives. These considerations include economics of the alternatives considered; technical issues such as site-specific limitations,

[1] While the term "spent nuclear fuel (SNF)" is used throughout this document, the report applies as well to "used" fuel (i.e., with potential reuse either directly or via a recycling method).

technology licensing status, and applicability of alternatives to nuclear operating companies' storage needs; and institutional issues.

Section 7 provides an overview of technical issues that have arisen over the past decade that may need to be considered during the implementation and operation of at-reactor ISFSIs. These issues include licensing issues, cask loading and transfer issues, fuel-related issues such as storage of damaged fuel, long-term SNF management planning, and issues associated with long-term storage of SNF.

Section 8 provides a summary of the regulatory requirements, planning and scheduling issues, technical and institutional issues associated with the development of an away-from-reactor central interim storage facility.

Section 9 provides a summary of institutional issues that should be considered early in a company's evaluation and planning for a SNF expansion project.

2
SPENT FUEL STORAGE EXPANSION EXPERIENCE

This chapter describes U.S. nuclear operating company experience with in-pool SNF storage expansion technologies, such as reracking and rod consolidation, and with dry storage of SNF in ISFSIs. While most plants have reached the limit for expanding in-pool storage capacity through reracking, some plants may rerack SNF pools in the future to address SNF storage pool neutron absorber degradation or to add additional capacity to unracked areas of the SNF storage pool.

While reracking has been the most used method for expanding at-reactor SNF storage capacity over the past 40 years, utility experience with dry storage applications has grown exponentially since 1986. Dry storage at nuclear power plant sites is expected to be implemented at all nuclear operating companies by approximately 2025 as plants reach the limits for expanding in-pool capacity through reracking. In addition to the implementation and continued operation of dry storage at operating plant sites, numerous nuclear power plants that have permanently ceased operation have offloaded SNF from storage pools to at-reactor ISFSIs in order to facilitate decommissioning of the SNF storage pools.

Due to the uncertainty associated with the Federal program for long-term SNF and high-level radioactive waste (HLW) management in the U.S., EPRI expects that SNF will continue to be stored at reactor sites, in SNF storage pools and dry storage, for the foreseeable future.

2.1 Pool Capacity Expansion Experience

SNF storage capacity expansion through the use of high-density storage racks has been the technology most widely used by utilities to increase in-pool storage capacity over the past 40 years. The majority of nuclear operating companies have reracked their SNF storage pools at least once. Some companies have reracked SNF storage pools several times as storage rack technologies advanced. Improvements in storage rack designs have generally decreased the center-to-center spacing in the SNF storage racks to allow more SNF to be stored in the pool. Through the use of neutron absorbers, the center-to-center spacing has been reduced to dimensions similar to the spacing of fuel assemblies in the reactor core.

Since most nuclear operating companies have already reracked SNF storage pools at least once, over the past decade there have been are only limited opportunities for nuclear power plants to increase the capacity of SNF storage pools through conventional reracking. In addition to conventional SNF pool reracking which generally involves removal and replacement of old storage racks, recent SNF pool expansion experience has included adding additional racks to unracked areas of the SNF storage pool, licensing soluble boron credit in SNF storage pool, licensing temporary SNF storage racks for areas such as cask loading pits, and replacement of racks or installation of neutron poison inserts to address the degradation of neutron absorbers such as Boraflex™, Carborundum B_4C, and Boral™. The NRC addresses neutron absorber degradation in Generic Letter 96-04. [NRC, 1996a] In 2009, NRC issued an Information Notice regarding degradation of neutron absorbers in the SNF pool. [NRC 2009e]

Boraflex was used to provide reactivity control in SNF pool storage racks that were manufactured in the late 1970s and 1980s. The SNF pool environment caused the physical and chemical properties of Boraflex™ to change, resulting in continual deterioration of the material in SNF pools. EPRI has conducted a comprehensive research program to evaluate in-service performance of this material and to investigate mitigation measures as described in EPRI Report TR-108761. [EPRI 1997].

Blistering and bulging have been observed in Boral™ used in both BWR and PWR pools. According to an EPRI Handbook regarding neutron absorbing materials, these blisters have not been observed to alter the neutron absorption properties of Boral™, but may result in difficulties removing or inserting fuel assemblies from affected storage cells. In addition, blisters may displace waster in flux trap regions of SNF storage racks, with the possibility of an increase in reactivity in the SNF storage rack configuration. [EPRI 2005e, NRC2009e]

Similarly, licensees with SNF storage racks that use Carborundum B_4C plates have experienced degradation of the neutron absorber that has resulted in swelling of the storage racks. In addition to the reduction in neutron absorbing material in the racks, the swelling can cause difficulties in removing or inserting fuel assemblies into degraded storage locations. [NRC 2009e]

2.1.1 Full and Partial Reracking

Reracking of SNF storage pools generally involves replacing some or all existing storage rack modules with new higher density storage racks, or installation of new racks in unracked areas of the pool, thereby increasing pool storage capacity. As noted above, high density rack designs typically have smaller center-to-center spacing than lower density racks, and incorporate neutron absorber material in the rack matrix. The neutron absorbers used in storage rack designs include borated aluminum (such as Boral™) or aluminum/boron carbide metal matrix composite material (such as Metamic™). [EPRI 2009b]

Comprehensive safety analyses must be performed to support a full or partial rerack of the SNF storage pool. Such analyses will generally include SNF pool criticality analysis; analysis of the mechanical and structural design; seismic analysis; analysis of radiation protection requirements during rack removal and installation; radiological consequence analysis; evaluation of changes to plant technical specifications; analysis of heavy loads over the SNF storage pool during rack removal and installation; and SNF pool thermal-hydraulic and decay heat analyses.

2.1.1 Temporary Storage Racks

During the past decade, many nuclear operating companies have sought license amendments that would allow the installation and use of temporary, free-standing storage rack modules in the cask loading pit area of the SNF storage pool. These temporary cask pit racks provide additional storage capacity to allow offloads of the reactor core during refueling outages and during non-outage fuel shuffles (e.g., movement of SNF within the pool). Cask pit racks may also be used to temporarily store new fuel prior to loading into the reactor core during a refueling outage. These temporary racks are designed to be removed from the cask pit area, cleaned and stored elsewhere on site. Since the cask pit areas are needed to load SNF into dry storage canisters for at-reactor storage, the removal of these temporary racks would occur in advance of cask loading operations.

In order to ensure that nuclear operating companies will have the capability to unload SNF to a SNF storage cask, NRC has imposed license conditions that require a licensee to restrict the combined number of fuel assemblies loaded in the existing SNF pool storage racks and cask pit rack to no more than the capacity of the SNF pool storage racks. This condition has been imposed to ensure that licensees will have the capability to unload and remove the temporary cask pit rack when SNF dry storage loading operations are ready to commence. The condition applies at all times, except during activities associated with a reactor core offload and refueling.

2.1.2 Soluble Boron Credit

During the period 1999 through the present, some pressurized water reactor (PWR) licensees have received license amendments that revise the SNF pool criticality analyses to allow credit for soluble boron. In some cases, the reliance on soluble boron credit for reactivity control was needed due to neutron absorber degradation in SNF storage racks. In other cases, the reliance on soluble boron credit was utilized in order to increase SNF pool capacity by allowing more flexibility in positioning SNF within existing storage racks.

Comprehensive safety analyses, with a focus on the SNF pool criticality analysis are generally performed to support the license amendment. Criticality analyses would consider the effects of post-irradiation cooling time, fuel depletion due to burnup, the presence of fuel rod axial blankets or control element assemblies (CEAs) in some fuel assemblies, integral burnable absorbers in fuel assemblies, and soluble boron concentrations. The minimum concentration of soluble boron must also be determined. For license amendments needed due to SNF pool storage rack neutron absorber degradation, the criticality analysis must also include the examination of the reactivity characteristics for a range of fuel storage configurations in the defined SNF pool storage regions without crediting a neutron absorber in the analyses. In addition to the revised criticality analysis, license amendments must address any expected changes to the structural or seismic analysis of SNF storage pool; radiological impacts such as increase in occupational exposure, fuel handling accident consequences; and impacts on the SNF storage pool thermal hydraulic analysis.

2.1.3 Storage Rack Inserts

In addition to employing soluble boron credit as part of the SNF pool criticality analysis, several nuclear operating companies have had their licenses amended or are in the process of amending licenses to allow the installation of SNF storage rack inserts, such as Metamic™ inserts or NETCO-Snap-In® inserts, to provide additional reactivity control due, in part, to continued degradation of SNF storage rack neutron absorbers. [EPRI, 2009b, Exelon 2009] NRC has approved license amendments for Turkey Point Units 3 and 4, and Arkansas Nuclear One (ANO) Unit 1 that would allow the installation of Metamic™ storage rack inserts to the current storage rack configurations. The license amendment for both Turkey Point and ANO require the licensees to perform a Metamic™ surveillance program to monitor how the absorber material properties change over time under the radiation, chemical, and thermal environment found in the SNF storage pools. [NRC 2007a, NRC 2007b]

The safety analyses performed to support storage rack inserts generally include an updated criticality analysis that analyzes the impact of the rack inserts, taking into account the SNF storage configurations in the defined SNF pool storage regions; evaluation of the light load handling operations associated with installation of rack inserts and subsequent SNF movements in the pool associated with the rack insert installation; evaluation of structural and seismic impacts associated with the use of rack inserts including the impact of the additional weight of the inserts on the SNF storage racks and the SNF pool structure; and evaluation of impacts on SNF pool thermal hydraulic analysis.

2.2 Spent Fuel Consolidation Experience

SNF rod consolidation was another option for increasing SNF pool capacity that was explored by the U.S. nuclear industry during the 1980s. An intact fuel assembly has fuel rods arranged in an open array with spacing between the fuel rods to permit cooling and neutron moderation. It was anticipated that rod consolidation would reduce the fuel volume by a factor of two compared to the volume of an intact fuel assembly by removing the fuel rods from the fuel assembly hardware and reconfiguring them into a metal storage canister is a closely packed array. After removing the fuel rods from the fuel assembly structure, the remaining non-fuel bearing components (NFBC), which include grid spacers, end fittings, guide tubes, etc., would be compacted and placed in a metal storage canister. The goal of the rod consolidation was to reduce the required number of storage locations by reducing the space required to store each fuel assembly. The term "consolidation ratio" refers to several physical quantities: the rod consolidation ratio or the ratio of assemblies consolidated to canisters of fuel rods; the NFBC hardware compaction ratio or volume-reduction ratio; and the total consolidation ratio, a combination of the rod consolidation ratio and the hardware-compaction ratio.

SNF consolidation demonstrations were conducted as part of six different programs in the U.S. The programs involved the U.S. Department of Energy, EPRI, New York State Energy Research and Development Authority (NYSERDA), equipment vendors, and nuclear operating companies. A summary of in-pool rod consolidation projects is presented in Table 2-1. While the demonstrations successfully achieved rod consolidation ratios of 2:1, the compaction of NFBC proved to be complicated and required significant cutting in the SNF pool. Ultimately, none of the companies that were involved in the demonstration projects pursued rod consolidation on a full scale basis.

Table 2-1
Summary of SNF Rod Consolidation Experience

Reactor/Location	Companies Involved in Demonstration	Date	Consolidation Ratio
Oconee	Westinghouse Electric, Duke Energy	1982	Rod: 2:1 NFBC: 6:1
Maine Yankee	Proto-Power Corporation, Maine Yankee	1981-1984	Rod: 1.6:1
West Valley	NAC International, Rochester Gas & Electric, EPRI, NYSERDA	1985-1986	Rod: 1.8: 1
Battelle Columbus	U.S. Tool & Die, Rochester Gas & Electric, EPRI, NYSERDA	1986	Rod: 1.85:1 to 2:1
Millstone 2	Combustion Engineering, Northeast Utilities, Baltimore Gas & Electric, EPRI	1983-1988 1991	Rod: 2:1
Prairie Island	Westinghouse, Northern States Power	1987	Rod: 2:1 NFBC: 6:1

2.3 Dry Storage Experience

The NRC first developed a separate regulatory framework for storage of SNF outside of the reactor SNF storage pools in November 1980 with the issuance of U.S. Code of Federal Regulations, Title 10, Part 72, "*Licensing Requirements for the Independent Storage of Spent Fuel and High-Level Radioactive Waste*" (10CFR72). This new regulation was supported by NRC's "*Final Generic Environmental Impact Statement on Handling and Storage of Spent Light Water Power Reactor Fuel,*" in which NRC determined that additional SNF storage capacity would be needed outside of reactor SNF storage pools. [NRC 1979] The regulation is applicable to both wet and dry storage systems at reactor sites or away from reactors.

In 1990, the NRC revised 10CFR72 to include new regulations that govern the general license procedures for dry storage. Rather than seeking a site-specific license under 10CFR72, any nuclear operating company with a 10CFR Part 50 (10CFR50) operating license was granted a general license that allowed them to store SNF at their nuclear power plant site in an NRC-certified dry storage cask. The regulations for dry storage of SNF under site-specific and general licenses are discussed in more detail in Section 3.

2.3.1 Dry Storage Demonstration Projects

Section 218 of the Nuclear Waste Policy Act required the Secretary of Energy to establish a demonstration program, in cooperation with the private sector, for dry storage of SNF at civilian nuclear power reactor sites. The objective of this demonstration program was to establish storage technologies that the NRC could approve by rulemaking, for use at reactor sites without the need for additional site-specific NRC approvals.

In 1986, Virginia Power (now, Dominion Generation) initiated a research program to provide additional SNF storage at its Surry station using dry storage technologies. DOE and EPRI signed cooperative agreements with Virginia Power to demonstrate dry storage technologies at a federal site; selection and shipping of SNF to the federal site for storage; and the design, licensing, and initial operation of an ISFSI at the Surry station. The cooperating agreement program called for more than one cask design to be demonstrated at the Surry ISFSI and at the Idaho National Laboratory (INL, at that time, the Idaho National Engineering Laboratory). The cask designs tested at INL included the CASTOR V/21, Westinghouse MC-10, and the Transnuclear TN-24P metal storage casks. EPRI published the results of the testing of these cask designs at INL in several EPRI reports.[EPRI 1986, EPRI 1987a, EPRI 1987b] Following receipt of a site-specific license from the NRC for an ISFSI at the Surry station, Virginia Power loaded three separate metal cask designs with SNF and placed these casks in storage at the ISFSI as part of the cooperative agreement. The cask designs loaded under the cooperative agreement included the CASTOR V/21, Westinghouse MC-10, and NAC I28. Virginia Power and EPRI documented this full-scale application of metal cask fuel storage technologies at the Surry station in an EPRI Report. [EPRI 1989]

DOE and EPRI also signed a cooperative agreement with Carolina Power and Light Company (CP&L, now Progress Energy) to demonstrate the NUHOMS technology at the H.B. Robinson plant for at-reactor dry storage. The project started in 1983 and culminated in July 1989 with the loading of SNF into the eight NUHOMS-07P storage modules, following receipt of a site-specific ISFSI license. CP&L and EPRI documented the results of this demonstration program in two EPRI Reports. [EPRI 1990a, EPRI 1990b]

DOE and EPRI participated in a third cooperative agreement with Wisconsin Electric Power Company (Wisconsin Electric) and Sierra Nuclear Corporation to demonstrate a concrete ventilated storage cask (VSC-17) at the INL. The project started in 1988 and its objectives were to demonstrate the thermal, shielding, and operational performance of the VSC-17 concrete cask when loaded with consolidated SNF assemblies and accurately predict thermal performance. In addition to the concrete cask testing program on the VSC-17 system, EPRI published a report regarding the comparative system economics associated with concrete SNF storage casks. [EPRI 1993] This demonstration program was part of the technical basis for development of the Sierra Nuclear's VSC-24 technology that was later used at several ISFSIs.

2.3.2 Early Dry Storage Experience

The first dry storage systems that were licensed for use in at-reactor ISFSIs were storage-only systems[2] that were licensed in accordance with NRC's site-specific licensing regulations, described in more detail in Section 3.

In July 1986, Virginia Power's Surry station received the first site-specific ISFSI license in accordance with 10CFR72. The site-specific license for the Surry ISFSI has been amended several times to include new metal cask designs and to modify the fuel characteristics allowed to

[2] While the first dry storage cask designs deployed at reactor sites in the U.S. were only licensed for storage, there was consideration by the industry in the 1980s regarding the benefits of using dual purpose dry storage systems (that is, systems licensed for both storage and transport).

be stored in the various cask designs. Under Surry's site-specific license, SNF has been loaded into a wide-range of metal casks: the CASTOR V/21, MC-10, NAC-I28/ST, which were part of the demonstration program discussed in Section 2.3.1; the CASTOR X, and TN-32. These cask designs are described in Section 4.

The dry storage demonstration projects yielded several early dry storage technologies. In addition to the metal storage casks used at the Surry ISFSI, the demonstration program at CP&L's H.B. Robinson site lead to the licensing of modular concrete storage system – NUHOMS (**Nu**tech **Ho**rizontal **M**odular **S**torage). The NRC granted a site-specific ISFSI license for the H.B. Robinson site in August 1986 for the NUHOMS-07P horizontal modular concrete storage system. The site-specific license allowed eight NUHOMS-07P modules to be loaded. This NUHOMS demonstration project led to the development of a higher capacity NUHOMS storage system that was first used at Duke Energy's Oconee site. The Oconee ISFSI received a site-specific license for a NUHOMS-24P storage system in January 1990. Information on the NUHOMS design is provided in Section 4.

In November 1991, Public Service Company of Colorado was granted a site-specific license by the NRC for the shutdown Fort St. Vrain high-temperature gas-cooled reactor (HTGR). The ISFSI was licensed to use the Modular Vault Dry Store (MVDS) system designed by FW Energy. The MVDS system is a vault-type structure that houses individual fuel assemblies in separate storage locations within the vault structure. This design has not been used at any other commercial nuclear power plant sites, but the technology has been used in Europe and by DOE at the INL for storage of DOE-owned SNF. The MVDS system is described in Section 4.

In April 1993, Consumers Energy Company's ISFSI at the Palisades site was the first to store SNF in a NRC-certified storage cask under the general license rules outlined by NRC in 10CFR72, Subpart K, "*General License for Storage of Spent Fuel at Power Reactor Sites*." The Palisades ISFSI utilized the Sierra Nuclear VSC-24 system, a vertical concrete cask system based on the VSC-17 system that was demonstrated through a cooperative agreement between Wisconsin Electric, Sierra Nuclear, and EPRI at INL. Since that time, the majority of dry storage facilities that have been commissioned in the U.S. have loaded SNF into NRC-certified casks under the general license regulations in 10CFR72. These regulations are described in more detail in Section 3.

2.3.3 Evolution to Dual-Purpose Storage and Transport Systems

In the late 1980s, the utility industry began to explore the concept of "dual-purpose" dry storage systems – that is, dry storage systems that are licensed by the NRC for both storage and transport without the need to re-handle individual SNF assemblies prior to shipment off-site. The interest in dual-purpose technologies was due, in part, to the industry's concern that DOE would not be able to begin SNF acceptance at a repository by 1998 and that DOE's plan to develop monitored retrievable storage was also faltering. [DOE 1989][3] In December 1992, Secretary of Energy James Watkins unveiled a "new strategy" for managing commercial SNF. [Watkins 1992] This new strategy included the development of canister based systems for at-reactor storage, eventual

[3] In 1989, DOE announced that the start of repository operations would be delayed until 2010 [DOE 1989]

transportation, and possible disposal – DOE referred to the concept as a "multi-purpose" canister (MPC) system (the term "multi-purpose" referring to storage, transport and disposal). DOE conducted feasibility studies on the concept of MPCs and issued a conceptual design report on MPCs in 1993. [DOE 1994] In 1994, DOE released a request for proposals for the design and possible certification of an MPC system. [DOE 1995] In 1995, DOE's management and operating contractor awarded a contract to Westinghouse Electric Corporation to design an MPC system. Activities associated with MPC system development by the DOE were later terminated due to funding constraints in Fiscal Year 1996 appropriations. [DOE 1996]

While DOE's MPC program was terminated, it spurred the development of dual-purpose canister (DPC) technologies by private industry for at-reactor dry storage. Facing an uncertain schedule for removal of SNF from plant sites, one of the benefits of dry storage using dual-purpose technologies for at-reactor dry storage is that, once SNF has been loaded into the sealed dual-purpose casks or canisters, the individual SNF assemblies would not have to be handled two or more times for the eventual transfer to a Federal waste management system. With a storage-only system, SNF is transferred from the SNF storage pool to a dry storage system; the SNF is stored in an ISFSI for an indefinite period of time; the storage system may need to be transferred back to the pool to be unloaded[4]; and SNF is then reloaded into a transportation cask for transport off-site. If storage-only systems are relied on for at-reactor dry storage, the SNF storage pool may need to be maintained in operating condition in order to transfer fuel from storage-only systems to transportation casks for transport off-site at some point in the future. The development of dual-purpose dry storage technologies has been particularly important for shutdown nuclear power plants that have off-loaded SNF to dry storage. Because the SNF at shutdown plants is stored in dual-purpose dry storage technologies, licensees have been able to dismantle and decommission their nuclear power plants, including the SNF storage pools, at these shutdown sites, allowing decreased long-term operation and maintenance costs associated with storing the SNF until it is ultimately removed from the sites.

With the prospect of very long-term dry storage at nuclear power plant sites, the majority of ISFSIs that have been commissioned since 2000 have loaded SNF into dual-purpose dry storage technologies. Even those companies that began ISFSI operation in the 1980s and 1990s have transitioned from storage-only technologies to dual-purpose technologies. In fact, only dual-purpose dry storage technologies are currently being marketed in the U.S. Table 2-2 provides a chronological summary of dry storage facility development at commercial nuclear power plants in the U.S. from 1986 to the present. As of April 2010, there were 47 operational ISFSIs storing SNF from approximately 77 power plants. At year-end 2009, approximately 63,000 MTU of permanently discharged SNF was in storage (both wet and dry) with more than 13,500 MTU of SNF loaded into more than 1,200 dry storage cask systems.

Eight (8) of the operational ISFSIs store SNF solely from nuclear power plants that have permanently ceased operation and have been decommissioned or are in the process of decommissioning (Fort St. Vrain, Rancho Seco, Trojan, Yankee Rowe, Big Rock Point, Maine Yankee, Haddam Neck, and Humboldt Bay). SNF from shutdown nuclear power plants is also stored in three additional ISFSIs along with SNF from the operating nuclear power plants at

[4] Some utilities with storage-only systems are discussing with NRC the possibility of authorization for a one-time shipment using the original storage-only canister.

those sites (Dresden, San Onofre, and Indian Point.) All of the shutdown plants are storing SNF in DPC-based technologies, except for Fort St. Vrain which uses a unique SNF storage vault system from which the fuel can be retrieved for eventual transport.

As shown in Table 2-3, it is expected that by 2020 almost every commercial nuclear power plant in the U.S. will have operational ISFSIs at their sites. In the period 2010 to 2020, an additional 22 ISFSIs are expected to implement dry storage with many of the facilities beginning operation between 2010 and 2015, as indicated by the approximate year in which SNF is expected to be loaded into dry storage packages. All of these sites are expected to utilize dual-purpose dry storage technologies. The remaining nuclear power plant sites are expected to implement dry storage by approximately 2025.

Table 2-2
Summary of Dry Spent Fuel Storage Facility Development at U.S. Commercial Nuclear Power Plants: 1986 to Present

Plant Name	Company Name	Fuel Type	License Type	Storage Technology	Year SNF Loaded
Surry 1 & 2	Dominion Generation	PWR	Site-specific	CASTOR V/21 MC-10, NAC I-28 CASTOR X/, TN-32	1986
			General	NUHOMS-32PTH	2007
H.B. Robinson	Progress Energy	PWR	Site-specific	NUHOMS-07P	1989
			General	NUHOMS-24PTH	2004
Oconee 1, 2, 3	Duke Energy	PWR	Site-specific	NUHOMS-24P	1990
			General	NUHOMS-24P NUHOMS-24PHB	2000
Fort St. Vrain (shutdown)	U.S. DOE (Previously owned by Public Service Colorado)	HTGR	Site-specific	Foster Wheeler MVDS	1991
Calvert Cliffs 1 & 2	Constellation Energy	PWR	Site-specific	NUHOMS-24P NUHOMS-32P	1992
Palisades	Entergy Nuclear Operations	PWR	General	VSC-24 NUHOMS-32PT, NUHOMS-24PTH	1993
Prairie Island 1 & 2	Xcel Energy	PWR	Site-specific	TN-40	1993
Point Beach 1 & 2	FPL Energy Point Beach	PWR	General	VSC-24 NUHOMS-32PT	1995
Davis Besse	FirstEnergy Nuclear Operating Co.	PWR	General	NUHOMS-24P	1995
Arkansas Nuclear One 1 & 2	Entergy Nuclear Operations	PWR	General	VSC-24 HI-STORM 24P HI-STORM 32P	1996
North Anna 1 & 2	Dominion Generation	PWR	Site-specific	TN-32	1998
			General	NUHOMS-32PTH	2008

Table 2-2 (continued)
Summary of Dry Spent Fuel Storage Facility Development at U.S. Commercial Nuclear Power Plants: 1986 to Present

Plant Name	Company Name	Fuel Type	License Type	Storage Technology	Year SNF Loaded
Susquehanna 1 & 2	PPL Susquehanna LLC	BWR	General	NUHOMS-52B NUHOMS-61BT	1999
Peach Bottom 2 & 3	Exelon Generation	BWR	General	TN-68	2000
Dresden 1, 2, 3 (Unit 1 – shutdown)	Exelon Generation	BWR	General	HI-STAR 68B HI-STORM 68B	2000
Hatch 1 & 2	Southern Nuclear Operating Co.	BWR	General	HI-STAR 68B HI-STORM 68B	2000
Rancho Seco (shutdown)	Sacramento Municipal Utility District	PWR	Site-specific	NUHOMS-24P	2001
McGuire 1 & 2	Duke Energy	PWR	General	TN-32 NAC UMS	2001
Trojan (shutdown)	Portland General Electric	PWR	Site-specific	TranStor Overpack HI-STORM 24P MPC	2002
Oyster Creek	Exelon Generation	BWR	General	NUHOMS-61BT	2002
Yankee Rowe (shutdown)	Yankee Atomic Electric Co.	PWR	General	NAC MPC	2002
Columbia	Energy Northwest	BWR	General	HI-STORM 68B	2002
Big Rock Point (shutdown)	Entergy Nuclear Operations	BWR	General	FuelSolutions W150	2002
FitzPatrick	Entergy Nuclear Operations	BWR	General	HI-STORM 68B	2002
Maine Yankee (shutdown)	Maine Yankee Atomic Power	PWR	General	NAC UMS	2002
Palo Verde 1, 2, 3	Arizona Public Service	PWR	General	NAC UMS	2003
San Onofre 1, 2, 3 (Unit 1 – shutdown)	Southern California Edison	PWR	General	NUHOMS-24PT	2003
Duane Arnold	FPL Energy.	BWR	General	NUHOMS 61BT	2003
Haddam Neck (shutdown)	Connecticut Light & Power	PWR	General	NAC MPC	2004

Table 2-2 (continued)
Summary of Dry Spent Fuel Storage Facility Development at U.S. Commercial Nuclear Power Plants: 1986 to Present

Plant Name	Company Name	Fuel Type	License Type	Storage Technology	Year SNF Loaded
Sequoyah 1 & 2	Tennessee Valley Authority	PWR	General	HI-STORM 32P	2004
Millstone 1, 2, 3 (Unit 1 – shutdown)	Dominion Generation	Unit 1 – BWR Unit 2, 3 - PWR	General	NUHOMS-32PT	2005
Farley 1 & 2	Southern Nuclear Operating Co.	PWR	General	HI-STORM 32P	2005
Browns Ferry 1, 2, 3	Tennessee Valley Authority	BWR	General	HI-STORM 68B	2005
Quad Cities 1 & 2	Exelon Generation	BWR	General	HI-STORM 68B	2005
River Bend	Entergy Nuclear Operations	BWR	General	HI-STORM 68B	2005
Fort Calhoun	Omaha Public Power District	PWR	General	NUHOMS-32PT	2006
Hope Creek	PSEG Nuclear	BWR	General	HI-STORM 68B	2006
Grand Gulf	Entergy Nuclear Operations	BWR	General	HI-STORM 68B	2006
Catawba 1 & 2	Duke Energy	PWR	General	NAC UMS	2007
Indian Point 1, 2, 3 (Unit 1 – shutdown)	Entergy Nuclear Operations	PWR	General	HI-STORM 32P	2008
Vermont Yankee	Entergy Nuclear Operations	BWR	General	HI-STORM 68B	2008
Limerick 1 & 2	Exelon Generation	BWR	General	NUHOMS 61BT	2008
St. Lucie 1 & 2	FPL Energy	PWR	General	NUHOMS 32PT	2008
Seabrook	FPL Energy	PWR	General	NUHOMS 32PT	2008
Monticello	Xcel Energy	BWR	General	NUHOMS 61BT	2008
Humboldt Bay (shutdown)	Pacific Gas & Electric	BWR	Site-specific	HI-STAR 100	2008
Kewaunee	Dominion Generation	PWR	General	NUHOMS-32P	2009
Diablo Canyon 1 & 2	Pacific Gas & Electric	PWR	Site-specific	HI-STORM 32P	2009

Table 2-3
Expected Dry Spent Fuel Storage Facility Development at U.S. Commercial Nuclear Power Plants 2010- 2020

Plant Name	Company Name	Fuel Type	Approximate Loading Year
Beaver Valley 1	FirstEnergy Nuclear Operating Co.	PWR	2013-2014
Brunswick 1 & 2	Progress Energy	BWR	2010-2011
Braidwood 1 & 2	Exelon Generation	PWR	2011
Byron 1 & 2	Exelon Generation	PWR	2010
Clinton	Exelon Generation	BWR	2016
Comanche Peak	TXU Generating Company	PWR	2014-2016
Cook 1 & 2	Indiana Michigan Power	PWR	2011
Cooper	Nebraska Public Power District	BWR	2010
Crystal River	Progress Energy	PWR	2012
Fermi	Detroit Edison	BWR	2010
Ginna	Constellation Energy	PWR	2010
LaCrosse (shutdown)	Dairyland Power	BWR	2011
LaSalle 1 & 2	Exelon Generation	BWR	2010
Nine Mile Point 1 & 2	Constellation Energy	BWR	2012
Perry	FirstEnergy	BWR	2010
Pilgrim	Entergy Nuclear Operations	BWR	2014-2015
Salem 1 & 2	PSEG Nuclear	PWR	2010
Summer	South Carolina Electric & Gas	PWR	2015-2017
Turkey Point 3 & 4	FPL Energy	PWR	2011
Vogtle	Southern Nuclear Operating Co.	PWR	2013-2014
Waterford 3	Entergy Nuclear Operations	PWR	2011-2012
Watts Bar 1 & 2	Tennessee Valley Authority	PWR	2020

3
SPENT FUEL STORAGE AND TRANSPORTATION REGULATIONS AND GUIDANCE

This section describes the regulations for at-reactor SNF storage under the site-specific and general license requirements contained in 10CFR72; the process for certification of a SNF package for storage under 10CFR72; and the process for certification of a SNF package for transport contained in Title 10, U.S. Code of Federal Regulations, Part 71, *Packaging and Transportation of Radioactive Material* (10CFR71).

In addition to providing a summary of the regulations governing SNF storage and transport, NRC regulatory guidance documents are summarized, including Standard Review Plans, Regulatory Guides, and Interim Staff Guidance for storage and transportation.

3.1 Site-Specific ISFSI License

In order to obtain a site-specific ISFSI license, the applicant must demonstrate to the NRC that issuance of the license, authorizing construction and operation of an ISFSI at a designated site, meets all of the technical, administrative, and environmental licensing requirements. A one-step licensing process is utilized in 10CFR72. The application for a site-specific license must contain general and financial information about the applicant, proposed technical specifications, a Safety Analysis Report (SAR), an emergency plan, an ISFSI decommissioning plan, a security plan, and an Environmental Report (ER). The regulations for a site specific license govern both at-reactor ISFSIs as well as away-from-reactor (AFR) storage facilities.

In the regulations under which the current site-specific licenses were granted, the initial license term for an ISFSI may not exceed 20 years from the date of issuance. However, licenses may be renewed by the NRC at the expiration of the initial license term upon application by the licensee for a period of 20 years. Several site-specific licenses have been renewed for a period of 40 years. The NRC is in the process of changing its regulations to allow an initial license term of up to 40 years and a license renewal term for a period of up to 40 years. [NRC 2009b] After the NRC receives and reviews a license for completeness, notice of the proposed action and opportunity for public hearing is published in the Federal Register to afford the public an opportunity to participate in the licensing process. Procedures associated with public hearings are specified in Title 10, Part 2, Subparts G and K.

An applicant has several options associated with the submittal of a site-specific license. These options include: (a) preparation and submittal of a stand-alone site-specific license application; (b) preparation and submittal of a site-specific license application that references a previously approved SNF storage technology's Topical Safety Analysis Report (TSAR) (described in Section3.8), or CoC (described in Section 3.3; or (c) some combination of (a) and (b).

The time required for the NRC to reach a final decision on an ISFSI license application will depend on whether or not the NRC holds a hearing on the application and the extent to which the SNF storage technology to be used has already been reviewed by NRC. Site-specific review times have ranged from approximately 15 months to approximately 45 months. A period of 36 months for NRC review is considered typical for an application that references a storage technology that has already been approved by the NRC. Longer site-specific license review times would be expected for an ISFSI license application that references a storage technology TSAR that is still undergoing NRC review and approval, or if there is significant intervention in the licensing proceeding by outside parties.

3.1.1 License Application

Subpart B of 10CFR72, "*License Application, Form, and Contents*", specifies the information to be contained in a license application (LA) for an ISFSI. Regulatory Guide 3.50, "*Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste*," outlines an acceptable format for submitting the LA information specified in 10CFR72. NRC review of a license application for a dry storage facility will be based on NUREG-1567, "*Standard Review Plan for Spent Fuel Dry Storage Facilities.*" [NRC 2000a]

A one-step licensing procedure is provided in 10CFR72; therefore, the LA must be complete prior to its initial submission to the NRC. In accordance with 10CFR72, Subpart B, a LA for an ISFSI must contain the following information, using the guidance provided by Regulatory Guide 3.50:

- The LA includes general and financial information required by 10CFR72.22 as well as a summary level of information contained in other documents that are included as attachments to the LA.
- The technical information and safety analysis report required by 10CFR72.24. The SAR is provided as an attachment to the LA. The Applicant's technical qualifications, required by 10CFR72.28, are included in the ISFSI SAR.
- Financial assurance and recordkeeping for decommissioning required by 10CFR72.30. The LA must include a proposed decommissioning plan for the ISFSI after all SNF, and reactor-related greater-than-Class C low-level radioactive (GTCC) waste have been removed.
- Emergency plan required by 10CFR72.32. The emergency plan is provided as an attachment to the LA.
- Proposed technical specifications required by 10CFR72.26. Technical specifications are provided as an attachment to the LA.
- ER required by 10CFR72.34. The ER is provided as an attachment to the LA.

- Security information as required by 10CFR72, Subpart H. The physical security program for an ISFSI is submitted under separate cover and includes a Physical Security Plan, Safeguards Contingency Plan, and a Security Training and Qualification Plan.
- Training program as required by 10CFR72.192. The training program is provided as an attachment to the LA.
- Quality assurance (QA) program description as required by 10CFR72.24(n). The QA plan is generally provided as a revision to the existing site QA plan approved under 10CFR50.
- Licensee contact with the NRC prior to and during the preparation of the LA and related documents, particularly the ER and the SAR, can help the licensee to determine, at an early stage, if its licensing approach is acceptable to the NRC and meets the applicable regulations and regulatory guidance. Additional information regarding the SAR, ER, emergency planning, quality assurance, security plan and safeguards, pre-operational testing, and ISFSI decommissioning planning associated with a site-specific license is summarized in the sections below.

3.1.2 Safety Analysis Report for a Site-Specific ISFSI

A SAR is required as part of a site-specific LA under 10CFR72. The SAR must present a description and safety assessment of the proposed site and ISFSI structures; a plan for the conduct of operation; general design criteria; an emergency plan; a description of the quality assurance program; a description of a detailed physical protection plan; and a description of the decommissioning plan. Regulatory Guides 3.48, "*Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation (Dry Storage)*", supplies guidelines for preparation of a SAR. In addition, Regulatory Guide 3.62, "*Standard Format and Content for the Safety Analysis Report for On-Site Storage of Spent Fuel Storage Casks*" supplies guidelines for SAR preparation for an ISFSI located at a reactor site and utilizing dry storage casks. Information requested in Regulatory Guides 3.48 and 3.62 may be referenced to docketed material previously submitted to the NRC and contained in the licensee's FSAR or in an NRC approved CoC or TSAR submitted by a dry storage system vendor. NRC review of a SAR for a dry storage facility will be based on NUREG-1567 "*Standard Review Plan for Spent Fuel Dry Storage Facilities*." [NRC 2000a]

3.1.3 Environmental Report Requirements

An ER must be submitted as part of the LA for an ISFSI in accordance with 10CFR72.34. The ER, which evaluates any environmental effects of the ISFSI, must meet the requirements of Subpart A of 10CFR51, *National Environmental Policy Act--Regulations Implementing Section 102(2)*. Requirements under 10CFR51.61 state that the ER must contain the information specified in 10CFR51.45 and that it must address the siting evaluation factors contained in Subpart E of 10CFR72. Unless otherwise required by the NRC, an environmental impact statement (EIS) for the storage of SNF at an ISFSI at a licensed commercial nuclear power plant site is not required as part of the ER. Regulatory Guide 4.2, Revision 2, "*Preparation of Environmental Reports for Nuclear Power Plants*", provides guidance on the format and content for an ER.

Following successful review of the ER, the NRC issues an Environmental Assessment and a Finding of No Significant Impact (FONSI). Following completion of these actions by the NRC, site-specific licensees are permitted to begin construction of an ISFSI.

3.1.4 Emergency Planning

For an ISFSI located on the site of a nuclear power reactor licensed for operation by the NRC, the emergency plan required by 10CFR50.47 satisfies the requirements of 10CFR72.32. The existing site emergency plan can be incorporated by reference in Chapter 9, Conduct of Operations of the ISFSI SAR. Revisions would be made to the emergency plan to address off-normal events associated with fuel transfer operations and dry storage of SNF at the ISFSI. A shutdown reactor that is planning to give up its Part 50 license must ensure that a stand-alone emergency plan exists for the ISFSI.

3.1.5 Quality Assurance Program

A NRC-approved QA program which satisfies the applicable criteria of Appendix B to 10CFR50 and which is established, maintained, and executed with regard to an ISFSI will be accepted by the NRC as meeting the requirements of 10CFR72.140(b). Reference to the approved program is made by the licensee in Chapter 11, Quality Assurance, of the site-specific ISFSI SAR. The program should be identified by its date of submittal to the NRC, docket number, and date of NRC approval. A shutdown reactor that is planning to give up its Part 50 license must ensure that a stand-alone quality assurance plan exists for the ISFSI.

3.1.6 Security Plan and Safeguards

Subpart H of 10CFR72, Physical Protection, states that the licensee shall establish a detailed plan for security measures for the physical protection of the ISFSI. The plan is to consist of two parts. Part I must demonstrate how the licensee will comply with applicable requirements of 10CFR73 and during transportation to and from the ISFSI. This must include the design for physical protection and the licensee's safeguards contingency plan and guard training plan. Part II must list tests, audits, inspections and any other means used to demonstrate compliance with 10CFR73 requirements. A shutdown reactor that is planning to give up its Part 50 license must ensure that a stand-alone security and safeguards plan exists for the ISFSI.

Current NRC requirements classify an ISFSI as a "protected area" and as such must be enclosed in a Controlled Access Area with an intrusion detection system, proper illumination, periodic security patrols, physical and visual searches of personnel and vehicles entering the ISFSI, and communications capability with plant security. In order to be certain that all requirements for security are consistent with 10CFR73, licensees should check current regulatory requirements including NRC security orders associated with at-reactor dry storage of SNF. The physical security plan should describe the overall security policies and outline specific criteria to be followed by all personnel entering the protected area. A security plan should be implemented for all phases of a fuel transfer campaign.

3.1.7 Pre-Operational Testing and Initial Operation

A description of the pre-operational testing and operating start up plans are to be included in the SAR with an emphasis on plans which demonstrate that the installation layout, equipment, and planned operations meet the safety and design criteria set forth in the SAR. Test programs that demonstrate the integrity of the installation structures and equipment and that demonstrate the applicability of the safety analysis should be detailed.

In addition to demonstrating that the ISFSI complies with the safety and design criteria set forth in the SAR, pre-operational testing aids the licensee in finding any operational problems during the test run rather than during the actual run with irradiated fuel. Prior to the start of pre-operational testing, NRC will conduct inspection activities as discussed in Section 7.2.

3.1.8 ISFSI Decommissioning Plan

A description of decommissioning plans for the ISFSI should be included in the license application and as part of the SAR. Information should be provided on the decommissioning method that has been selected along with plans for facilitating the decommissioning process.

3.2 General License for Storage of Spent Fuel at Power Reactor Sites

In 1990, the NRC issued a general license under 10CFR72, Subpart K, *General License for Storage of Spent Fuel at Power Reactor Sites*. This rule authorized any Part 50 licensee to store SNF at its reactor site in NRC-certified storage casks without filing licensing documents necessary for a site-specific application and without prior specific NRC approval. At that time, the NRC originally approved four storage casks for use under the general license. Storage casks are issued CoCs in accordance with 10CFR72, Subpart L. Following NRC safety review, a rulemaking process is initiated to add NRC-certified casks to 10CFR72.214, *List of Approved Spent Fuel Storage Casks*. While it was initially thought that use of a general license for dry storage would require fewer utility resources, experience has shown that storage of SNF under either a site-specific or general license will require a significant commitment of personnel, resources, and expertise. Under either licensing process, it is the licensee that is ultimately responsible for storage system quality, safety, and reliability.

Use of a general license does not require submission of an application to the NRC and does not present an opportunity for a formal licensing hearing on the use of the cask at a nuclear power plant site. In order to use a certified cask under a general license, the regulations in 10CFR72 state that a reactor licensee must notify NRC of its intent to do so 90 days prior to the first use of the cask to store SNF at its site.

In order to load SNF into a certified cask under a general license, the licensee must perform written evaluations in accordance with 10CFR72.212, prior to use of the cask, that establish the conditions set forth in the cask's CoC have been met, that the cask storage pad and areas have been designed to adequately support the load of the casks, and that NRC limits for radioactive material effluents and direct radiation will be satisfied. These written evaluations would contain information similar to that required in the SAR and ER for a site-specific license and must be retained until SNF is no longer stored under the general license. The licensee must also confirm that the reactor site parameters are enveloped by the cask design bases contained in the CoC, and determine whether the activities related to use of the general license involve any unreviewed safety question, or a change in the facility technical specification as provided under 10CFR50.59. The licensee's review must also address security, emergency planning, quality assurance, radiation protection, and training programs described in Section 3.1. Although this review must be documented, it does not have to be submitted to the NRC for prior approval. No later than 30 days after using a cask, the reactor licensee must register each cask with the NRC.

In the regulations under which the current storage CoCs were granted, the general license for the storage of SNF in each cask fabricated under a CoC terminates 20 years after the first use of the particular cask, unless the cask's CoC is renewed, in which case the general license then terminates 20 years after the cask's CoC renewal date. The term of the general license continues independent of whether the cask vendor remains in business. In the event that a cask vendor does not apply for renewal of a cask CoC, any cask user or user's representative may apply for renewal of the CoC. NRC is in the process if changing its regulations to allow an initial CoC term of up to 40 years and a CoC renewal term of up to 40 years. [NRC 2009b] The new rule would also revise the regulations to specify that the general license for the storage of SNF in each cask fabricated under a CoC commences upon the date that the particular cask is first used by the general licensee to store SNF and should not exceed the term certified by the cask's CoC, unless the cask's CoC is renewed, in which case the cask can be used under a general license until the cask's CoC expires.

If a shutdown plant that is storing SNF in an ISFSI under a general license wants to surrender its Part 50 license, the company would need to convert to a site-specific license under Part 72. However, a shutdown reactor is permitted to retain its Part 50 license throughout the period that SNF remains on site. To date, all of the shutdown plant sites that have build ISFSIs using the general license provisions have maintained their Part 50 license and plan to do so for the foreseeable future. Thus, general licensees have not yet demonstrated the process to transition from a general license under the provisions of 10CFR72, Subpart K, to a 10CFR72 site-specific license.

Although a reactor licensee is not required to notify the NRC until 90 days prior to first use of casks at its site under a general license, the NRC recommends that licensees notify NRC staff regarding dry storage plans several years in advance of the first cask loading in order to ensure that NRC has adequate resources available for inspection activities associated with ISFSI construction, pre-operational review and testing, and initial operations. It would also be prudent for the licensee to begin its review of cask CoCs and the associated safety analyses several years prior to first use to ensure that site parameters are enveloped, no unreviewed safety questions exist, no changes are needed to technical specifications, and no amendments would be required to the cask CoC for use at the reactor site. Planning horizons should assume that it may be necessary to seek changes to a Part 50 license or to a cask CoC in order to store SNF in certified casks under a general license, as discussed in more detail in Section 5.

3.3 10CFR72 Certificate of Compliance

The approval process used to certify a SNF storage cask for use under a general license is detailed in 10CFR72, Subpart L, *Approval of Spent Fuel Storage Casks*. Dual purpose technologies must also be certified in accordance with the provisions contained in 10CFR71, as described in Section 3.4. The approval processes for the storage CoC and transport CoC can be done separately or in parallel. An applicant for a storage CoC must submit an application to the NRC, including a SAR describing the proposed cask design and the conditions under which the cask should be used to store SNF. NRC will review cask SAR, issue written Requests for Additional Information (RAI), prepare a draft Safety Evaluation Report (SER), a draft environmental assessment (EA), and a draft CoC for the package.

Upon completion of these draft documents, NRC publishes a notice of proposed rulemaking for public comment in the Federal Register to amend 10CFR72.214, *List of Approved Spent Fuel Storage Casks*. Following review of any public comments submitted, the NRC completes a final SER, EA, and CoC. A final rulemaking, including the resolution of any public comments, is published in the Federal Register along with the effective date of the final rule and the CoC. This is typically 30 days after publication.

NRC review of storage systems for certification under 10CFR72, Subpart L, is based on NUREG-1536, "*Standard Review Plan for Dry Cask Storage Systems*." [NRC 2009a][5] Conditions of approval outlined under Subpart L include:

- Design, fabrication, testing and maintenance of SNF storage casks must meet NRC requirements outlined in 10CFR72.236.
- Design, fabrication, testing and maintenance of SNF storage casks must be conducted under a QA program that meets NRC requirements.
- Fabrication of casks under a CoC may begin prior to receipt of the cask CoC; however, such fabrication is done at the cask applicant's risk. A cask that is fabricated prior to the CoC being issued must conform to the final CoC before SNF can be loaded in the cask.
- The cask vendor must ensure that a record is established and maintained for each cask fabricated under the CoC. The record must include:
  - the NRC CoC number;
  - the cask model number;
  - the cask ID number;
  - date fabrication was started and completed;

[5] It should be noted that NRC has issued, for public comment, a draft revision to NUREG-1536, NRC 2009a, "*Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility — Draft Report for Comment*," NUREG-1536. [NRC 2009a] This draft NUREG-1536, Revision 1A, was revised to incorporate interim staff guidance documents ISG 1 through ISG 22. A final version of NUREG-1536 is expected to be published in mid 2010. The final version, as well as any applicable ISGs should be consulted by licensees as there may be changes between the draft and final. This SRP will be revised periodically to reflect current guidance to the staff.

  - certification that the cask was designed, fabricated, tested and repaired in accordance with an NRC-accepted QA program;
  - certification that required inspections under 10CFR72.236(j) were performed and found satisfactory; and
  - the name and address of cask user.
- The original of this record is to be supplied to the cask user and a copy maintained by the cask vendor. These records must be available to the NRC for inspection.
- Cask vendors must also ensure that written procedures and appropriate tests are established prior to use of the certified casks. A copy of procedures and tests must be provided to each cask user.

A summary of the storage casks that have been approved by NRC for use under a general license is included in Section 4, Table 4-2.

3.4 Regulations for a 10CFR71 Certificate of Compliance

SNF transport cask designs must be certified by the NRC in accordance with the regulations contained in 10CFR71. Designers of SNF transport casks submit an application to the NRC for review and approval. The application contains information as described in the *Standard Review Plan for Transportation Packages for Spent Fuel* (NUREG-1617). [NRC 2000b] An application for certification of a transport package must address the safety and operational characteristics of the package, including design analysis for structural, thermal, radiation shielding, nuclear criticality, material content confinement, and analysis of the package under both normal conditions of transport and the hypothetical accident conditions. In addition, the application must contain operational guidance, such as any testing and maintenance requirements, operating procedures, and conditions for package use. After performing an independent review of the application, if NRC determines that the SNF cask meets the 10CFR71 requirements, it issues a SER and a Radioactive Material Package CoC to the cask designer. The CoC allows any licensee to use the cask as long as the licensee has a general or specific NRC license to *"…receive, possess, use, or transfer licensed 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."* In addition, licensees must also have a NRC-approved quality assurance plan that meets the requirements of 10CFR71, Subpart H, Quality Assurance.

The designers of SNF transport casks must demonstrate, either through physical testing or computer analysis, that the casks will meet NRC requirements related to containment of material, radiation control, and criticality control under both normal conditions of transport (as specified in 10CFR 71.71) and hypothetical accident conditions (as specified in 10CFR 71.73). Under normal conditions of transport, the radiation level must not exceed: (1) 200 mrem per hour at any point on the external surface of the package; and (2) 10 mrem per hour at any point 80 inches (2 meters) from the outer surface of the transport vehicle.

The hypothetical accident conditions require that the conditions be sequentially imposed on the transport package and that any damage caused by the sequential accident conditions are cumulated. That is, evaluation of package's ability to withstand any one accident condition must consider the damage that resulted from the previous accident conditions. The 10CFR 71.73 accident conditions require that casks be subjected to all of the following accident conditions in the following sequence:

- Free Drop: A 9 meter (30-foot) free drop of the cask onto a flat, unyielding, horizontal surface. The cask must strike the surface in a position for which maximum damage is expected.
- Puncture: A 1 meter (40-inch) free drop of the cask onto a vertical steel bar, 15 centimeters (six inches) in diameter, mounted on an unyielding, horizontal surface. The cask must strike the steel bar in a position for which maximum damage is expected.
- Thermal: Exposure of the cask in a fully-engulfing, hydrocarbon fuel/air fire with an average flame temperature of at least 800°C (1475°F) for a period of 30 minutes. The regulations specify the physical conditions of the fire including the dimensions of the hydrocarbon fuel source around the cask and the position of the cask relative to the surface of the fuel source.
- Immersion: Immersion under at least 0.9 meters (3 feet) of water.

As a separate accident condition, 10CFR 71.51 requires a deep immersion test for SNF packages with activity greater than 37 PBq (1 million Curies (37 PBq). The regulations require that the package must be designed so that its undamaged containment system can withstand an external water pressure of 2 MPa (290 psi) for a period of not less than one hour without collapse, buckling, or in-leakage of water. The pressure requirement of 2 MPa (290 psi) is equivalent to 200 meters (656 feet) of water submersion and corresponds to the approximate depth of the continental shelf.

NRC regulations allow cask designers to determine cask response to the hypothetical accident conditions either by physical test or by computer analysis. The regulations define the allowable radioactivity release and allowable external radiation dose from a package after being subjected to the hypothetical accident conditions. In addition, the package must be designed such that a criticality event cannot occur under normal conditions of transport or hypothetical accident conditions.

Transportation package CoCs are issued for a period of five years, and may be renewed for a new five year period. In order to renew a CoC, a CoC holder would submit a request to NRC with any necessary supporting information describing the capability of the package design to continue to meet technical requirements. After reviewing this information, the NRC would determine whether to grant a CoC renewal.

Dual-purpose storage and transport systems that have valid storage and transport CoCs are summarized in Table 3-1.

3.5 NRC Acceptance Procedures and Rules of Engagement

The NRC developed Standard Review Plans (SRP) to summarize the regulatory requirements necessary for approval of an application for SNF storage and transportation packages, as well as for development of ISFSIs. The SRPs describes the procedures that NRC staff use to determine that the regulatory requirements have been satisfied by a particular license application. In addition to the SRPs, NRC staff has also issued interim staff guidance documents (ISGs) to identify emerging issues and develop staff positions. These ISGs, identified in Section 3.9, are initially issued in a draft form for public comment and are available on NRC's web site. SRPs will be periodically updated by NRC staff in order to incorporate ISG documents.

NRC staff has also developed guidelines to define its expectations for interactions between NRC staff, ISFSI licensees, and SNF storage and transport package CoC holders during the licensing process. NRC has referred to these guidelines as "rules of engagement." NRC's rules of engagement were updated by NRC in a Regulatory Issue Summary, 2004-20, "*Lessons Learned from review of 10 CFR Parts 71 and 72 Applications*." [NRC 2004c] In 2010, NRC staff from the Division of Spent Fuel Storage and Transportation (SFST) published an SFST Office Instruction regarding the Acceptance Review Process for radioactive materials packages, discussed in more detail below. The rules of engagement include the following provisions:

- **Pre-Application Interactions**: Applicants are encouraged to meet with SFST staff prior to submittal of applications for ISFSIs or storage and transport CoCs in order to discuss potential licensing actions. The purpose of these meetings is to provide applicants with an opportunity to discuss their proposals with staff and solicit feedback regarding regulatory positions.
- **NRC Point of Contact**: NRC will assign a project manager as the primary point of contact between SFST and an applicant.
- **Telephone interactions**: SFST encourages telephone contact with applicants as a means of clear understanding between the parties and efficiency of the review process. However, since these telephone interactions are not public meetings, they only involve a general information exchange.
- **Submittal of applications**: NRC requires applicants to submit complete, high-quality applications that follow the recommended format and content in appropriate Regulatory Guides, Standard Review Plans, and Interim Staff Guidance documents. Deviations from regulatory positions should be clearly identified and include sufficient technical basis. SARs should include a thorough explanation of all issues. Submittals should include sufficient information for a technical reviewer to perform an independent review. NRC also suggests that applicants consider the submittal of a draft CoC license when a new application is filed or a draft mark-up CoC for a license amendment.
- **Review of applications**: The project manager performs an administrative review of an application to determine the completeness of the application and to identify any significant omissions of information in accordance with an Acceptance Review Process that was published as an SFST Office Instruction in April 2010. [NRC 2010d] During the acceptance review, NRC staff conducts an administrative and technical sufficiency review to ensure that the application contains sufficient technical information for the staff to conduct its technical review of the application. According to the Acceptance Review Process, the acceptance review should take no longer than 60 days. Acceptance reviews for simple and routine amendment requests should be completed in 30 days or less. As part of the acceptance review

process, NRC staff may issue a Request for Supplemental Information (RSI) to support the detailed technical review to follow.

- **Requests for additional information**: NRC will provide an applicant with a specific time period in which to respond to RAIs, based on the complexity of the application and its priority. If an applicant cannot meet the NRC's schedule, the applicant must submit a letter to the project manager at least two weeks in advance of the RAI response due date and provide the new submittal date and the reasons for the requested change. If an applicant does not meet the NRC's RAI response due date, this may result in rescheduling of the application because of other previously scheduled casework and competing priorities.
- NRC encourages applicants to meet with the NRC staff in a public meeting to discuss proposed RAI responses in order to ensure that the applicant's responses will address the staff's issues and avoid the need for a second RAI. If an RAI is unclear, an applicant may request clarification, which can be accomplished through a conference call with the appropriate technical reviewer. NRC advises that applicants should not submit partial RAI responses, as NRC staff will not begin a technical review of the RAI response until a complete response has been received. When setting the review schedule for an application, SFST staff will not schedule a second round of RAIs. If one is needed and the responses to the second RAIs are not sufficient for the staff to make a licensing determination, then SFST will identify its positions and concerns in a public meeting, and suspend further technical review.

3.6 Interface with 10CFR50 License

3.6.1 Site-Specific License

When storing SNF under a site-specific license, all fuel and cask handling operations performed in the SNF pool and cask handling areas are done in accordance with the 10CFR50 operating license. Thus, fuel handling and cask handling procedures need to be revised to accommodate cask transfer and loading operations. Technical specifications and fuel pool related safety analysis must include and bound cask handling and loading operations. Changes to the 10CFR50 license to accommodate cask handling operations or modifications to the SNF storage pool may require a 10CFR50 license amendment.

Other interfaces with the 10CFR50 license for a site-specific 10CFR72 license are limited since the 10CFR72 license is a separate license, independent of the reactor operating license. There are two specific references to 10CFR50 in 10CFR72. The emergency plan under Subpart B, Paragraph 72.32(c) allows use of the emergency plan required by 10CFR50.47 for an ISFSI located at a licensed nuclear reactor site. The QA requirements under Subpart G, Paragraph 72.140(d) accept an NRC-approved QA program which satisfies the criteria of 10CFR50, Appendix B and is maintained and executed for an ISFSI.

3.6.2 General License

As stated in 10CFR72.212 (b)(4), prior to the use of certified casks under the general license, licensees are required to determine if activities related to the storage of SNF under a general license involve any unreviewed safety questions or changes in the facility technical specifications, as provided under 10CFR50.59. Section 72.212 (b)(6) requires a review of the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine if there is a decrease in the effectiveness of these programs caused by SNF storage under a general license and requires that the general licensee prepare the necessary changes to these programs. As was needed for a site-specific license, fuel pool handling procedures must be revised to include cask transfer and cask loading operations. The technical specifications and fuel pool related safety analysis must bound cask handling and loading operations. Changes to the Part 50 license to accommodate cask handling operations or modifications to the SNF storage pool may require a Part 50 license amendment.

Section 72.218, *Termination of Licenses*, requires a plan for the removal of SNF stored under a general license from the reactor site as part of the program for SNF management that is required by 10CFR50.54(bb). In addition, an application for termination of a reactor operating license submitted under 10CFR50.82 must describe how the SNF stored under a general license will be removed from the reactor site.

3.7 Topical Safety Analysis Reports

While the majority of the dry storage facilities in operation store SNF under the general license provisions of 10CFR72, there are site-specific licensees that are storing SNF in cask designs that have not been certified by NRC under 10CFR72, Subpart L. In lieu of submitting a LA and SAR for certification of a SNF storage cask, cask vendors may also prepare a TSAR for a specific SNF storage cask design and these TSAR documents may be incorporated into an a site-specific 10CFR72 SAR by reference. Regulatory Guide 3.61 provides guidance on the format and content of a TSAR for a SNF storage cask.

The TSAR would include an analysis of the cask design describing potential hazards and how the design protects against these hazards. Included in this analysis are:

- Cask vulnerability to accidents;
- Radiation shielding;
- Confinement and control of radioactive materials;
- Reliability of safety systems; and
- Radiological impact.

The TSAR would describe all technical information, provide a safety assessment of the design bases, and describe the QA program associated with design and fabrication of the casks. An analysis of anticipated operations would also be included describing preoperational testing, anticipated operations and maintenance, potential limiting conditions on cask use or fuel to be stored, and facility decommissioning considerations. Descriptions of analytical models or calculational methods and technical information to support new design features may also be included in the TSAR.

3.8 Changes, Tests, and Experiments

In accordance with 10CFR72.48, site-specific or general licensees and 10CFR72 CoC holders may make changes to their ISFSIs, SNF storage cask designs, and procedures as described in the ISFSI or storage system Final SAR (FSAR) or may conduct tests or experiments which are not described in the FSAR. These changes, tests, or experiments may be made without prior NRC approval if they do not involve a change in the license conditions, an unreviewed safety question, a significant increase in occupational exposure, or a significant unreviewed environmental impact. A proposed change, test, or experiment constitutes an unreviewed safety question if the probability or the consequences of an accident or malfunction evaluated in the FSAR may be increased; if a possibility for an accident or malfunction not previously evaluated in the FSAR may be created; or if the safety margin of any technical specification is reduced. This provision is especially useful during the license application review process and during ISFSI construction for incorporating unforeseen changes into the FSAR without having to amend the license application or submit a licensing amendment.

The licensee is required to maintain records of any changes, tests, or experiments not described in the FSAR but that are allowed under 10CFR72.48. The records must include a written safety evaluation that provides bases for the determination that the change, test, or experiment did not involve an unreviewed safety question. An annual report describing the changes, tests, and experiments including a summary of the safety evaluation for each is to be sent to the NRC.

Any changes, tests, or experiments not described in the FSAR which do constitute an unreviewed safety question, significant increase in occupational exposure, significant unreviewed environmental impact, or which change the license conditions, require the licensee to submit an application for an amendment of the license pursuant to 10CFR72.56.

Licensees using dual purpose technologies must be aware of the fact that changes to a storage technology licensed under both 10CFR72 for storage and 10CFR71 for transport may require an amendment to the 10CFR71 transport CoC to accommodate the changes made under 10CFR72 prior to the system being readied for transport. This is due the fact that 10CFR71 regulations do not have a provision similar to 10CFR72.48. In 2002, NRC published a proposed rule that considered revisions to 10CFR71 that would allow certificate holders for dual-purpose technologies to make certain changes to the package without prior NRC approval. After evaluating the proposed rule and public comments, NRC determined that implementation of this change would result in new regulatory burdens and costs that could be significant. However, NRC staff also issued Interim Staff Guidance (ISG) -20 in order to clarify the degree of flexibility allowed in transport package design changes, including changes in contents and in package operations, without prior NRC approval.

3.9 Interim Staff Guidance Documents

SFST has issued ISG documents that are used to provide guidance to NRC staff concerning issues not currently addressed in a SRP or issues where clarification of SRP text is necessary. The ISGs are intended to ensure consistent reviews by the SFST staff. NRC staff expect to incorporate the guidance contained in the ISGs into the applicable SRPs during periodic updates of the SRPs.

The current practice is for SFST to issue a draft ISG for comment prior to the ISG becoming final. This is similar practice to publication of a draft NUREG document for public comment prior to being finalized. Since ISGs can and have been amended from time to time, EPRI does not describe the detailed technical issues addressed by these ISGs in this section. EPRI provides a listing of the current ISGs, the revision number, publication date, and a brief description of each ISG. As noted in Section 3.3., NRC incorporated ISG-1 through ISG-22 into a draft SRP, NUREG-1536. [NRC 2009a]

Specific ISGs should be consulted for their applicability to SNF storage and/or transport and to determine if the specific ISG supersedes guidance contained in previously published SRPs associated with SNF storage and/or transport. SRPs associated with transport package applications have not yet been revised to incorporate ISG documents. In addition, as with any NRC guidance documents, the guidance provided in ISG documents is not a regulation or requirement and can be modified or superseded by an applicant with supportable technical arguments.

- [6]SFST-ISG-1, Rev. 2, Damaged Fuel, May 11, 2007: provides guidance on classifying SNF as either (1) damaged, (2) undamaged, or (3) intact, before interim storage or transportation. This is not a regulation or requirement and can be modified or superseded by an applicant with supportable technical arguments.
- SFST-ISG-2, Rev. 1, Fuel Retrievability, February 22, 2010: provides guidance for determining if storage systems to be licensed under 10CFR72 allow ready retrieval of SNF.
- SFST-ISG-3, Rev. 0, Post Accident Recovery and Compliance with 10CFR72.122(l): had been interpreted to mean that a licensee must have a means of retrieving and repackaging individual fuel assemblies even after an accident. NRC reevaluated this interpretation and proposed modification of SRPs to communicate the distinction between retrievability and post accident recovery. The interpretation that will be incorporated into the storage SRPs is that 10CFR72.122(l) applies to normal and off-normal design conditions and not to accidents.
- SFST-ISG-4, Rev. 1, Cask Closure Weld Inspections: According to Appendix C of the draft NUREG-1536, dated March 2009, this ISG has been superseded by ISG-15 and ISG-18.

[6] NRC issued, for public comment, a draft revision to NUREG-1536, NRC 2009a, "*Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility — Draft Report for Comment*," NUREG-1536. [NRC 2009a] This draft NUREG-1536, Revision 1A, was revised to incorporate interim staff guidance documents ISG 1 through ISG 22. A final version of NUREG-1536 is expected to be published in mid 2010. The final version, as well as any applicable ISGs should be consulted by licensees as there may be changes between the draft and final. This SRP will be revised periodically to reflect current guidance to the staff.

- SFST-ISG-5, Rev. 1, Confinement Evaluation: provides NRC staff's recommended approach to confinement evaluation for dry cask storage systems.
- SFST-ISG-6, Rev.0, Establishing Minimum Initial Enrichment for the Bounding Design Basis Fuel Assembly(s): provides NRC staff's recommendation that, in a SNF package SAR, "the minimum initial enrichment should be specified as an operating control and limit for cask use, or the licensee should justify the use of a neutron source term, in the shielding analysis, that specifically bounds the neutron sources for fuel assemblies to be placed in the cask. Absent adequate justification acceptable to the staff, the SAR should not attempt to establish specific source terms as operating controls and limits for cask use."
- SFST-ISG-7, Rev. 0, Potential Generic Issue Concerning Cask Heat Transfer in a Transportation Accident: addresses NRC staff concerns regarding the potential for adverse effects of fission gases to the gas-mixture thermal conductivity in a SNF canister in a post accident environment, including: (1) the reduction of the thermal conductivity of the canister gas by the mixing of fission gases expelled from failed fuel pins and (2) the resultant temperature and pressure rise within the canister.
- SFST-ISG-8, Rev. 2, Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks, September 27, 2002: expands NRC guidance contained in NUREG-1617 regarding burnup credit to:: (1) extend the range of allowed burnup and cooling time; (2) allow loading of assemblies exposed to burnable absorbers; (3) removing the loading offset for initial ^{235}U enrichments between 4 and 5 percent; and (4) indicate an acceptable source for selecting a bounding axial burnup profiles.
- SFST-ISG-9, Rev. 1, Storage of Components Associated with Fuel Assemblies, April 22, 2002: clarifies the technical criteria for types of materials that may be approved for storage along with SNF assemblies, such as PWR control rods.
- SFST-ISG-10, Rev. 1, Alternatives to the ASME Code, November 13, 2000: provides NRC staff recommendations regarding documentation of licensee commitments to the application of ASME Code Section III and proposed alternatives to the Code to the design and fabrication of dry cask storage systems. NRC staff recommends that all codes and standards applied to components important to safety and associated approved alternatives as committed to by the applicant should be documented in the license, certificate of compliance, or technical specification.
- SFST-ISG-11, Rev. 3, Cladding Considerations for the Transportation and Storage of Spent Fuel, November 17, 2003: expands the technical basis for the storage of SNF including assemblies with average burnups exceeding 45 GWd/MTU and specifies the acceptance criteria for limiting SNF reconfiguration in storage casks.
- SFST-ISG-12, Rev. 1, Buckling of Irradiated Fuel Under Bottom End Drop Conditions: provides recommendations regarding analyses of fuel rod buckling performed to demonstrate fuel integrity following a cask drop accident.
- SFST-ISG-13, Rev. 0, Real Individual, May 17, 2000: clarifies the meaning of a real individual as used in 10CFR 72.104; specifies how the applicant may perform dose evaluations beyond the controlled area for site-specific and general licenses; and clarifies storage SRP text regarding dose calculations outside the controlled area.

- SFST-ISG-14, Rev. 0, Supplemental Shielding, November 13, 2000: provides guidance regarding supplemental shielding, such as an earthen berm, that may be installed at an ISFSI to meet the requirements of 10 CFR 72.104(a) regarding the dose limits for normal conditions of operation.
- SFST-ISG-15, Rev. 0, Materials Evaluation, January 10, 2001: provides specific guidance for the review of materials selected by an applicant for its dry cask storage system or transportation package.
- SFST-ISG-16, Rev. 0, Emergency Planning, June 14, 2000: provides specific guidance for review of Emergency Plans for facilities licensed pursuant to 10CFR72.
- SFST-ISG-17, Rev. 0, Interim Storage of Greater Than Class C Waste, November 16, 2001: provides guidance regarding the interim storage GTCC waste at ISFSIs or monitored retrievable storage (MRS) facilities.
- SFST-ISG-18, Rev. 1, The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage, October 3, 2008: provides guidance to address the design and testing of the closure welds associated with the redundant closure of all-welded austenitic stainless steel canisters as an acceptable confinement boundary under 10 CFR 72.236(e).
- SFST-ISG-19, Rev.0, Moderator Exclusion Under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel Under the Requirements of 10 CFR 71.55(e), May 2, 2003: provides review guidance for meeting the fissile material package standards in 10CFR 71.55(e), which require that a fissile material package be subcritical under hypothetical accident conditions assuming that the fissile material is in the most reactive credible configuration consistent with the damaged condition of the package and the chemical and physical form of the contents, and water moderation occurs to the most reactive credible extent consistent with the damaged condition of the package and the chemical and physical form of the contents.
- SFST-ISG-20, Rev. 0, Transportation Package Design Changes Authorized Under 10 CFR Part 71 Without Prior NRC Approval, April 8, 2005: clarifies the degree of flexibility allowed in transport package design changes, including changes in contents and in package operations, without prior NRC approval. Unlike 10CFR72, 10CFR71 does not include specific change authority that would allow licensees to make changes in the design or operation of an NRC-certified package without prior NRC approval.
- SFST-ISG-21, Rev. 0, Use of Computational Modeling Software, April 5, 2006: supplements guidance in the storage and transport SRPs regarding NRC staff's position on what an acceptable analysis using computational modeling software (CMS) *"should include, and what information should be reviewed by the staff when considering a submittal from an applicant using CMS in the design review of a storage cask or transportation package. This ISG applies to both thermal and structural analyses utilizing CMS."*
- SFST-ISG-22, Rev. 0, Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations in LWR or Other Uranium Oxide Based Fuel, May 8, 2006: provides technical review guidance to ensure that operating procedures, technical specification, and associated licensing documentation submitted by applicants,

provide a supportable analysis of the potential for cladding splitting, should fuel rods be exposed to an oxidizing gaseous atmosphere.

- SFST-ISG-23, Draft, Application of ASTM Standard Practice C1671-07 when performing technical reviews of spent fuel storage and transportation packaging licensing actions, June 2009: provides guidance regarding the use of ASTM Standard Designation C1671-07 "*in the evaluation of licensing actions which rely upon boron-based metallic neutron absorber materials for nuclear criticality control in dry cask storage systems and transportation packaging*."
- SFST-ISG-25, Draft, Pressure Test and Helium Leakage Test of the Confinement Boundary for Spent Fuel Storage Canister, September 3, 2009: supplements guidance in the storage SRPs for evaluating the helium leakage testing and ASME Code required pressure (hydrostatic/pneumatic) testing that is specified for the canister confinement boundary.

3.10 Additional Storage and Transport Guidance Documents

The NRC has issued guidance documents related to SNF storage including Regulatory Guides and NUREG documents. Other guidance documents associated with dry storage are also included in the list below.

3.10.1 U.S. Code of Federal Regulations

- 10CFR2, "Rules of Practice for Domestic Licensing Proceedings and Issuance of Orders"
- 10CFR20, "Standards for Protection Against Radiation"
- 10CFR50, "Domestic Licensing of Production and Utilization Facilities," August 15, 1991.
- 10CFR71, "Packaging and Transportation of Radioactive Material"
- 10CFR72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste"
- 10CFR73, "Physical Protection of Plants and Materials"
- 10CFR961, "Standard Contract for Disposal of Spent Nuclear Fuel and/or High Level Radioactive Waste"

3.10.2 Regulatory Guides

- Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water Cooled Nuclear Power Plants".
- Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants"
- Regulatory Guide 3.48, "Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage)".

- Regulatory Guide 3.50, “Standard Format and Content for a License Application to Store Spent Fuel and High-Level Radioactive Waste”.
- Regulatory Guide 3.53, “Applicability of Existing Regulatory Guides to the Design and Operation of an Independent Spent Fuel Storage Installation”.
- Regulatory Guide 3.54, “Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation”.
- Regulatory Guide 3.60, “Design of an Independent Spent Fuel Storage Installation (Dry Storage)”.
- Regulatory Guide 3.61, “Standard Format and Content for a Topical Safety Analysis Report for a Spent Fuel Dry Storage Cask”.
- Regulatory Guide 3.62, “Standard Format and Content for the Safety Analysis Report for Onsite Storage of Spent Fuel Storage Casks”.
- Regulatory Guide 3.66, “Standard Format and Content of Financial Assurance Mechanisms”.
- Regulatory Guide 3.72, “Guidance for Implementation of 10 CFR 72.48, Changes, Tests, and Experiments”
- Regulatory Guide 3.73, “Site Evaluations and Design Earthquake Ground Motion for Dry Cask Independent Spent Fuel Storage and Monitored Retrievable Storage Installations”
- Regulatory Guide 4.2, "Preparation of Environmental Report for Nuclear Power Stations"
- Regulatory Guide 7.6, “Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels”
- Regulatory Guide 7.8, “Load Combinations for the Structural Analysis of Shipping Casks for Radioactive Material”
- Regulatory Guide 7.9, “Standard Format and Content of Part 71 Applications for Approval of Packages for Radioactive Material”
- Regulatory Guide 7.10, “Establishing Quality Assurance Programs for Packaging Used in Transport of Radioactive Material”
- Regulatory Guide 7.11, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Maximum Wall Thickness of 4 Inches (0.1 m)”
- Regulatory Guide 7.12, “Fracture Toughness Criteria of Base Material for Ferritic Steel Shipping Cask Containment Vessels with a Wall Thickness Greater than 4 Inches (0.1 m) But Not Exceeding 12 Inches (0.3 m)”

3.10.3 NUREG Documents

- NUREG-0554, "Single Failure-Proof Cranes for Nuclear Power Plants," May 1979
- NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants,” January 1980
- NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems, Initial Report,” January 1997

- NUREG-1536, Revision 1A, Draft, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility — Draft Report for Comment,” March 2009.
- NUREG-1567, “Draft Standard Review Plan for Spent Fuel Dry Storage Facilities,” March 2000.
- NUREG-1571, "Information Handbook on Independent Spent Fuel Storage Installations"
- NUREG-1609, “Standard Review Plan for Transportation Packages for Radioactive Material,” Initial Publication, March 1999.
- NUREG-1609, Standard Review Plan for Transportation Packages for MOX-Radioactive Material (NUREG-1609, Supplement 1), September 2005.
- NUREG-1609, Standard Review Plan for Transportation Packages for Irradiated Tritium-Producing Burnable Absorber Rods (TPBARs) (NUREG-1609, Supplement 2), February 2006.
- NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel, Initial Report, March 2000.
- NUREG-1617, “Standard Review Plan for Transportation Packages for MOX Spent Nuclear Fuel, Supplement 1, September 2005.
- NUREG-1619, "Standard Review Plan for Physical Protection Plans for the Independent Storage of Spent Fuel and High-Level Radioactive Waste,” July 1998
- NUREG-1714, “Final Environmental Impact Statement for the Construction and Operation of an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians and the Related Transportation Facility in Tooele County, Utah,” December 2001.
- NUREG-1727, “NMSS Decommissioning Standard Review Plan,” September 2000.
- NUREG-1745, “Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance,” June 2001.
- NUREG-1864, “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant,” March 2007.
- NUREG-1927, “Standard Review Plan for Renewal of Independent Spent Fuel Storage Installation Licenses and Dry Cask Storage System Certificates of Compliance”, Draft report for comment, September 2009.

3.10.4 NUREG/CR Documents

- NUREG/CR-6314, "Quality Assurance Inspections for Shipping and Storage Containers," April 1996.
- NUREG/CR-6361, “Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages,” March 1997.
- NUREG/CR-6407, “Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety,” February 1996.

- NUREG/CR-6487, “Containment Analysis for Type B Packages Used to Transport Various Contents,” November 1996.
- NUREG/CR-6700, “Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High-Burnup LWR Fuel,” January 2001.
- NUREG/CR-6701, “Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel,” January 2001.
- NUREG/CR-6716, “Recommendations on Fuel Parameters for Standard Technical Specifications for Spent Fuel Storage Casks,” March 2001.
- NUREG/CR-6759, “Parametric Study of the Effect of Control Rods for PWR Burnup Credit,” February 2002.
- NUREG/CR-6760, “Study of the Effect of Integral Burnable Absorbers for PWR Burnup Credit,” March 2002.
- NUREG/CR-6761, “Parametric Study of the Effect of Burnable Poison Rods for PWR Burnup Credit,” March 2002.
- NUREG/CR-6798, “Isotopic Analysis of High-Burnup PWR Spent Fuel Samples From the Takahama-3 Reactor,” January 2003.
- NUREG/CR-6801, “Recommendations for Addressing Axial Burnup in PWR Burnup Credit Analyses,” March 2003.
- NUREG/CR-6802, “Recommendations for Shielding Evaluations for Transport & Storage Packages,” May 2003.
- NUREG/CR-6835, “Effects of Fuel Failure on Criticality Safety and Radiation Dose for Spent Fuel Casks,” September 2003.

3.10.5 Other References

- NRC Bulletin 95-01, “Quality Assurance Program for Transportation of Radioactive Material,” January 13, 1995
- NRC Bulletin 96-02, “Movement of Heavy Loads Over Spent Fuel, Over Fuel in the Reactor Core, or Over Safety Related Equipment,” April 11, 1996.
- NRC Bulletin 96-04, “Chemical, Galvanic, or Other Reactions in Spent Fuel Storage and Transportation Casks,” July 5, 1996.
- NRC Bulletin 97-02, “Puncture Testing of Shipping Packages under 10 CFR Part 71,” September 23, 1997.
- NRC Information Notice 1995-38, “Degradation of Boraflex Neutron Absorber in Spent Fuel Pool Storage Racks,” September 8, 1995.
- NRC Information Notice 1995-29, “Oversight of Design and Fabrication Activities for Metal Components Used in Spent Fuel Dry Storage Systems,” June 7, 1995.

- NRC Information Notice 1995-28, “Emplacement of Support Pads for Spent Fuel Dry Storage Installations at Reactor Sites,” June 5, 1995.
- NRC Information Notice 1997-39,” Inadequate 10 CFR 72.48 Safety Evaluations of Independent Spent Fuel Storage Installations,” June 26, 1997.
- NRC Information Notice 1999-29, “Authorized Contents of Spent Fuel Casks,” October 28, 1999.
- NRC Information Notice 1999-15, “Misapplication of 10 CFR Part 71 Transportation Shipping Cask Licensing Basis to 10 CFR Part 50 Design Basis,” May 27, 1999.
- NRC Information Notice 2002-35, “Changes To 10 CFR Parts 71 and 72 Quality Assurance Programs,” December 20, 2002.
- NRC Information Notice, 2003-16, “Icing Conditions Between Bottom Of Dry Storage System and Storage Pad,” October 6, 2003.
- NRC Information Notice 2004-13, “Registration, Use, and Quality Assurance Requirements for NRC-Certified Transportation Packages,” June 30, 2004.
- NRC Information Notice 2009-26, “Degradation of Neutron-Absorbing Materials in the Spent Fuel Pool,” October 28, 2009.
- Regulatory Issue Summary, RIS-06-022: “Lessons Learned from Recent 10 CFR Part 72 Dry Cask Storage Campaign,” November 15, 2006.
- ANSI/ANS 57.9-1992-R2000, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)”.
- ANSI N14.6-1993, “Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More”.
- ANSI N45.2-1977, “Quality Assurance Program Requirements for Nuclear Power Plants”.
- ANSI N45.2.15, “Requirements for the Control of Hoisting, Rigging and Transporting of Items at Nuclear Power Plant Sites”.
- Nuclear Energy Institute, “Dry Fuel Storage Generic Action Plan”, NEI 97-01, March 1997
- Nuclear Energy Institute, “Guidelines for 10 CFR 72.48 Implementation,” Appendix B, NEI 96-07

4
OVERVIEW OF DRY STORAGE TECHNOLOGIES

This section provides an overview of dry storage technologies that have been approved by the NRC for storage and/or transport and that are being used for storage of SNF at nuclear power plant sites or are actively being marketed for at-reactor storage in the U.S. These dry storage systems are either approved for use as part of a site-specific ISFSI license or have been certified for storage by the NRC. This overview will include a general description of the dry storage technology, its physical parameters, its current licensing status, and a summary of those plants that have loaded or plan to load these systems in the near term.

4.1 Dry Storage Technologies Approved Under Site-Specific Licenses

As discussed in Section 2, the NRC has issued site-specific licenses for ISFSIs at 11 commercial nuclear power plant sites. A site-specific license may reference an NRC-approved TSAR for one or more dry storage technologies; CoCs for dry storage technologies that were previously approved by NRC; or the dry storage technology may be approved for storage specifically under the site-specific license. Dry storage technologies that are referenced within a site-specific license SAR or that are included in site-specific ISFSI licenses are summarized in Table 4-1.

Several dry storage technologies that were originally approved by NRC through vendor TSARs and referenced in site-specific licenses were later certified by NRC under 10CFR72, Subpart L, "Approval of Spent Fuel Storage Casks." For example, the Surry ISFSI (SNM-2501) references TSARs for the CASTOR V/21, MC-10, NAC-I28 S/T, CASTOR X, and TN-32 metal cask designs. The CASTOR V/21, MC-10 and TN-32 metal cask designs later received COCs under 10CFR72, as shown in Table 4-2.

In addition, the NUHOMS system was first approved for storage as the NUHOMS-07P system in the H.B. Robinson site-specific ISFSI license. The NUHOMS-24P system was first approved in the site-specific licenses for the Oconee and Calvert Cliffs ISFSIs, and the NRC later issued a CoC for the Standardized NUHOMS system (CoC #72-1004). The Standardized NUHOMS-24P is the system that is referenced in the Rancho Seco site-specific ISFSI.

The site-specific license for the Trojan ISFSI is based on a hybrid of two separate storage systems. The BNFL Fuel Solutions (BFS) TranStor system was approved for storage under Trojan's site-specific ISFSI license. Portland General Electric (PGE) experienced problems during TranStor canister loading operations in the Trojan SNF pool. While PGE had already finished construction of the Trojan ISFSI, including the TranStor concrete overpacks, it decided not to utilize the TranStor canisters. Instead, PGE amended its site-specific license to permit the use of the Holtec MPC-24P canister for storage in the TranStor concrete overpacks. This system references the HI-STORM 100 CoC #72-1014.

The Humboldt Bay ISFSI also uses a storage technology that was approved under its site-specific license, the HI-STAR HB, which is a modified version of the HI-STAR 100 system. The ISFSI is designed to store the HI-STAR HB systems in a below-grade storage vault. The cask storage vault is comprised of six below-grade, cylindrical storage cells that are structural units constructed of steel-reinforced concrete with a carbon steel liner. The vault is sized to hold five HI-STAR HB casks with SNF and one GTCC certified cask.

Table 4-1
Summary of Site-Specific Licenses and Referenced Dry Storage Systems

ISFSI	Dry Storage System	Cask Vendor	Site-Specific License
Surry	CASTOR V/21 MC-10 NAC I28 CASTOR X TN-32	General Nuclear Systems, Inc. Westinghouse NAC International, Inc. General Nuclear Systems, Inc. Transnuclear, Inc.	SNM-2501
H.B. Robinson	NUHOMS-07P	Transnuclear, Inc.	SNM-2502
Oconee	NUHOMS-24P	Transnuclear, Inc.	SNM-2503
Fort St. Vrain	MVDS	Foster Wheeler Energy Applications	SNM-2504
Calvert Cliffs	NUHOMS-24P	Transnuclear	SNM-2505
Prairie Island	TN-40	Transnuclear	SNM-2506
North Anna	TN-32P	Transnuclear	SNM-2507
Rancho Seco	NUHOMS-24P (standardized)	Transnuclear	SNM-2510
Trojan	HI-STORM 24P MPC TranStor Concrete Overpack	FuelSolutions Holtec International	SNM-2509
Humboldt Bay	HI-STAR HB	Holtec International	SNM-2514
Diablo Canyon	HI-STORM 100SA (CoC 72-1014)	Holtec International	SNM-2511

4.2 Dry Storage Technologies Approved For Use Under a General License

Table 4-2 identifies dry storage technologies that have been certified by NRC and that are listed in 10CFR72.214, "*List of approved spent fuel storage casks*." These technologies are approved for use under a general license. Table 4-2 includes the cask vendor, storage design model, applicable 10CFR72 CoC and its date of issuance, as well as the 10CFR71 transport CoC for those technologies that are certified for both storage and transport.

In order for a certified storage technology to be added to the list in 10CFR72.214, or for an amendment to a CoC, NRC must undergo a rulemaking process through publication of a notice in the Federal Register to allow for public comment. NRC regulations for transport under 10CFR71 do not require rulemaking in order for NRC to issue a CoC for transportation.

Table 4-2
Summary of Dry Strorage Technologies Approved for Use Under a 10CFR72 General License

Cask Vendor	Storage Design Model	Certificate of Compliance	Date of Issuance
General Nuclear Systems	CASTOR V/21	Storage: 72-1000	8/17/1990
NAC International, Inc.	NAC S/T	Storage: 72-1002	8/17/1990
NAC International, inc.	NAC-C28 S/T	Storage: 72-1003	8/17/1990
Transnuclear, Inc.	TN-24	Storage: 72-1005	11/04/1993
BNG Fuel Solutions	VSC-24	Storage: 72-1007	05/07/1993
	FuelSolutions	Storage: 72-1026	2/15/2001
		Transport: 71-9276	10/31/2007
Holtec International, Inc.	HI-STAR 100	Storage: 72-1008	10/4/1999
	HI-STORM 100	Storage: 72-1014	6/1/2000
	HI-STAR 100	Transport: 71-9261	5/8/2009
NAC International, Inc.	NAC-UMS	Storage: 72-1015	11/20/2000
		Transport: 71-9270	10/29/2007
	NAC-MPC	Storage: 72-1025	4/10/2000
		Transport: 71-9235	6/12/2009
	MAGNASTOR	72-1031	2/4/2009
Transnuclear, Inc.	NUHOMS-24P NUHOMS-52B NUHOMS-61B, -61BTH NUHOMS-32PT, -32PTH NUHOMS-24PHB NUHOMS-24PTH	Storage: 72-1004	1/23/1995
	Advanced NUHOMS-24PT1	Storage: 72-1029	2/5/2003
	NUHOMS-HD	Storage: 72-1030	1/10/2007
	MP 187 MP 197	Transport: 71-9255 Transport: 71-9302	11/25/2008 8/30/2007
	TN-32	Storage: 72-1021	4/19/2000
	TN-68	72-1027	5/28/2000
		71-9293	2/10/2006

Source: U.S. Nuclear Regulatory Commission, http://www.nrc.gov/waste/spent-fuel-storage/designs.html

4.3 NRC-Approved Dry Storage Technologies

This section provides an overview of dry storage technologies that have been approved by the NRC for storage or for storage and transport (that is, dual purpose technologies) and are currently being used for storage of SNF at nuclear power plant sites or are actively being marketed for at-reactor storage in the U.S. The dry storage technologies described below have either been approved for use as part of a site-specific ISFSI license or have been certified under 10CFR72 for storage by the NRC. This overview includes a general description of the dry storage technology, current licensing status, 10CFR72 CoC number (if applicable), and a summary of those plants that have loaded or plan to load these systems in the near term. For those dry storage technologies that are currently being loaded at nuclear power plant sites or that have been certified by NRC and are expected to be loaded in the future, additional information about these technologies is provided including physical parameters of the technologies.

4.3.1 EnergySolutions

EnergySolutions purchased BNG FuelSolutions in 2006, becoming the licensee for the VSC-24 canister-based storage system and the FuelSolutions dual-purpose system. EnergySolutions is no longer marketing these dry storage systems in the U.S.; however, SNF is stored in the VSC-24 system at three nuclear power plant sites and in the FuelSolutions system at a fourth site.

4.3.1.1 VSC-24

The VSC-24 system is a vertical concrete cask system that stores 24 PWR SNF assemblies. Its components include a multi-assembly sealed basket (MSB), a ventilated concrete cask (VCC), and a MSB transfer cask (MTC). The welded MSB provides confinement and criticality control for the storage and transfer of SNF. The VCC provides radiation shielding while allowing cooling of the MSB by natural convection during storage. The VSC-24 CoC #72-1007 expires on May 7, 2013. The VSC-24 system is used for storage of SNF at Entergy's Palisades ISFSI, Dominion's Point Beach ISFSI, and Entergy's Arkansas Nuclear One ISFSI. All three of these sites are now loading DPC-based systems for SNF storage rather than continuing to load SNF into the storage-only VSC-24 design.

The MSB consists of a steel cylindrical shell with a thick shield plug and steel cover plates welded at each end. The shell length is fuel specific and varies from 4.2 meters (m.) to 4.6 m. (13.8 to 15.1 feet [ft.].), the diameter is 1.6165 m (5.3 ft.), and the shell thickness is 2.54 centimeters (cm.) (1 inch [in.]). An internal fuel basket is designed to hold 24 PWR SNF assemblies. The steel basket is a welded structure consisting of 24 square storage locations. Support in the horizontal direction is provided by curved supports located at each end and the center of the basket assembly. The basket is coated with a CarboZinc 11 coating for corrosion protection. The MSB is installed vertically in the VCC, as shown in Figure 4-1.

The VCC is a reinforced concrete cask in the shape of a right circular cylinder. The VCC has four air inlets on the cask bottom and four air outlets located at the top of the concrete cask. The air outlets and inlets are protected from debris intrusion by wire mesh screens during storage.

The internal cavity of the VCC as well as the air inlets and outlets are steel-lined. After the MSB is inserted in the VCC, a shield ring is placed over the MSB/VCC gap and the cask weather cover is installed.

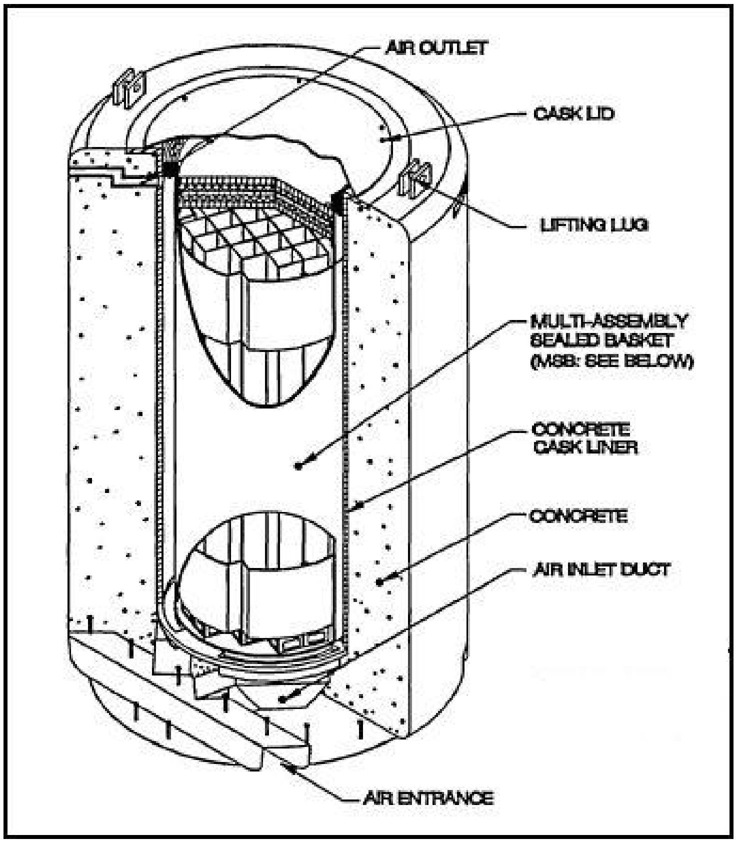

Figure 4-1
VSC-24 Concrete Storage Cask, CoC #72-1007 [BNG FuelSolutions 2005]

The MTC is used to shield, support and protect the VCC during fuel loading and MSB transfer to the VCC. It is a shielded lifting device with inner and outer structural steel cylinders which house lead and solid RX-277 neutron shield cylinders.

4.3.1.2 FuelSolutions

The FuelSolutions DPC system is a vertical concrete cask system that includes the following components: the W21 or W74 canisters; W100 transfer cask used for canister loading, closure and transfer to the concrete storage cask; and the W150 concrete storage cask that provides passive vertical dry storage of a loaded canister. The W21 canister stores up to 21 PWR assemblies and the W74 canister stores 64 BWR assemblies. The FuelSolutions CoC #72-1026 expires on February 15, 2021. A TS125 transportation cask has been certified for transport of the W21 or W74 canister, CoC # 71-9276. The FuelSolutions system (W74) is used for storage of SNF at the Big Rock Point ISFSI – a shutdown nuclear power plant.

The W21/W74 seal-welded DPC consists of a shell assembly, top and bottom inner closure plates, vent and drain port covers, internal basket assembly, top and bottom shield plugs, and top and bottom outer closure plates. All structural components are constructed of high strength carbon or stainless steel. The W21 fuel basket is a right circular cylinder configuration with 21 stainless steel guide tubes for the PWR contents. The W74 fuel basket assembly consists of two right circular cylindrical baskets, with a total of 74 cell locations and a capacity of up to 64 BWR assemblies. The ten unfueled cell locations are mechanically blocked to prevent loading in these positions.

The W150 is the storage overpack for both the W21 and W74 canisters. There is a long and a short version of the cask, both of reinforced concrete with a steel liner. The W150 provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during long-term storage. The storage cask has an annular air passage to allow the natural circulation of air around the canister.

The W100 transfer cask provides shielding during canister movements between the SNF pool and the storage cask. The cask is a multi-wall (steel/lead/steel/water/steel) design. Covers are bolted on each end of the cask to allow access to the cask cavity from either end. The top cover includes a secondary central cover for ram access during horizontal loading and unloading operations. The W100 neutron shield cavity is filled with clean water either prior to placement in or following removal from the SNF pool. To prevent contamination of the annular region between the W100 and the canister, an inflatable annulus seal is used during loading. The TS125 transportation cask is used for offsite shipment of the loaded W21 or W74 canister. It is of stainless steel construction with integral gamma and neutron shielding.

4.3.2 Foster Wheeler Energy Applications

FW Energy Applications, Inc. received an approved topical report for the MVDS for storage of PWR and BWR SNF in 1988. MVDS consists of a concrete vault module with a matrix of 83 PWR or 150 BWR storage positions. The MVDS was later modified to store HTGR fuel from the Fort St. Vrain reactor. Each storage position holds one fuel assembly in a fuel storage container (FSC). The loaded FSC is brought to the MVDS in a transfer cask (TC) from the reactor building and is received in the transfer cask reception bay. The TC is lifted from the TC trailer by the MVDS crane, and positioned vertically in the cask load/unload port. The FSC is then removed by the container handling machine (CHM) and moved by crane over the charge face structure of the storage vault. When the CHM reaches its position, the crane lowers the CHM to the charge face structure in the vault module.

The MVDS civil structure consists of vault modules, the transfer cask reception bay, the charge face structure, foundation structure, and standby and neutron source storage wells. The foundation for the vault modules is a reinforced concrete slab which extends approximately 3.05 m. (10 ft.) beyond each side of the storage facility. Each vault module is shielded by a 1.07 m. (3.5 ft.) concrete wall. Ambient air flows into the vault modules through an inlet duct and passes around the outside of the FSCs. The air exits through a reinforced concrete exhaust stack, which projects above the MVDS and is covered by a steel canopy. The charge face structure forms a shielded cover over the vault modules and is made of carbon steel and filled with non-structural concrete for shielding purposes. The air outlet stack forms one wall of the charge hall and the other three sides are made of reinforced concrete up to 10.37 m. (34 ft.) above ground level. The remainder of the walls and the roof are made of steel sheathing supported by steel framework. The transfer cask reception bay is part of the MVDS civil structure. It allows the transfer cask to be moved laterally by the MVDS crane into the cask load/unload port through an opening above the bay at the FSC level. The Fort St. Vrain MVDS ISFSI and charge face are shown in Figure 4-2.

Figure 4-2
Fort St. Vrain ISFSI, MVDS Dry Storage Facility and Interior Charge Face [DOE 2009]

4.3.3 General Nuclear Systems, Inc.

4.3.3.1 CASTOR V/21

General Nuclear Systems, Inc.'s CASTOR V/21 metal storage cask is designed to store 21 PWR SNF assemblies in a vertical orientation. The cask is approximately 16 feet high and 8 feet in diameter. It weighs approximately 117 tons when fully loaded. The CASTOR V/21 received a certificate of compliance for use under a general license in 1990, CoC #72-1000.

The CASTOR V/21 cask body consists of a thick-walled nodular iron casting. It is sealed with two stainless steel lids bolted to the cask and elastomer and metal seals are used for each lid to provide leak tightness. Gamma and neutron radiation shielding is provided by the cast iron wall of the cask body and additional neutron shielding is provided by polyethylene rods incorporated into the cask wall. The external surface of the cask is covered with heat transfer fins that run circumferentially around the cask. The inside of the cask contains a fuel basket structure comprised of 21 square tubes of welded stainless steel and borated stainless steel plates for criticality control. The inside of the cask has a nickel coating for corrosion protection. An epoxy resin coating protects the outside of surface of the cask. Four trunnions are connected to the cask for lifting and rotating. The cask system uses a pressure-sensing device to monitor the pressure in the interspace between the primary and secondary lids to ensure leaktightness. The CASTOR V/21 is currently in use at Virginia Power's Surry ISFSI under a site-specific license. The CASTOR V/21 cask is shown in Figure 4-3.

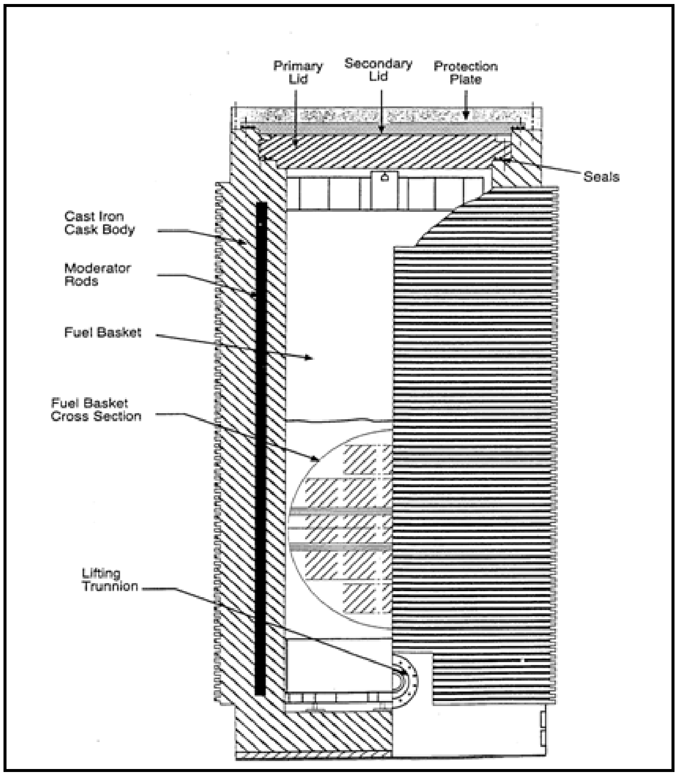

Figure 4-3
CASTOR V/21 Metal Storage Cask, CoC # 72-1000 [NEI 1998]

4.3.3.2 CASTOR X

General Nuclear Systems, Inc.'s CASTOR X/33 metal storage cask is designed to store 33 PWR SNF assemblies in a vertical orientation, as shown in Figure 4-5. The cask is approximately 4.8 m. (15.8 ft.) high and 2.38 m. (7.8 ft.) in diameter. It weighs approximately 107 tonnes [t.] (118 tons) when fully loaded. The NRC approved the CASTOR X/33 TSAR for storage, Docket 72-1018, in April 1994.

The CASTOR X/33 cask body is made of 0.304 m. (12 in.) thick ductile cast iron as shown in Figure 4-4. The cask is sealed with two stainless steel lids bolted to the cask using both metallic and elastomeric O-ring seals. Gamma shielding is provided by the wall of the cask body and neutron shielding by a row of polyethylene rods incorporated into the cask wall. The external surface of the cask is covered with heat transfer fins that run circumferentially around the cask. The inside of the cask contains a fuel basket structure comprised of 33 square tubes of welded stainless steel and borated stainless steel plates for criticality control. The inside of the cask has a nickel coating for corrosion protection. An epoxy resin coating protects the outside surface of the cask. Four trunnions are connected to the cask for lifting and rotating. The cask system uses a pressure-sensing device to monitor the pressure in the interspace between the primary and secondary lids to ensure leaktightness of the seals. The CASTOR X/33 is currently in use at Virginia Power's Surry ISFSI, under a site-specific license.

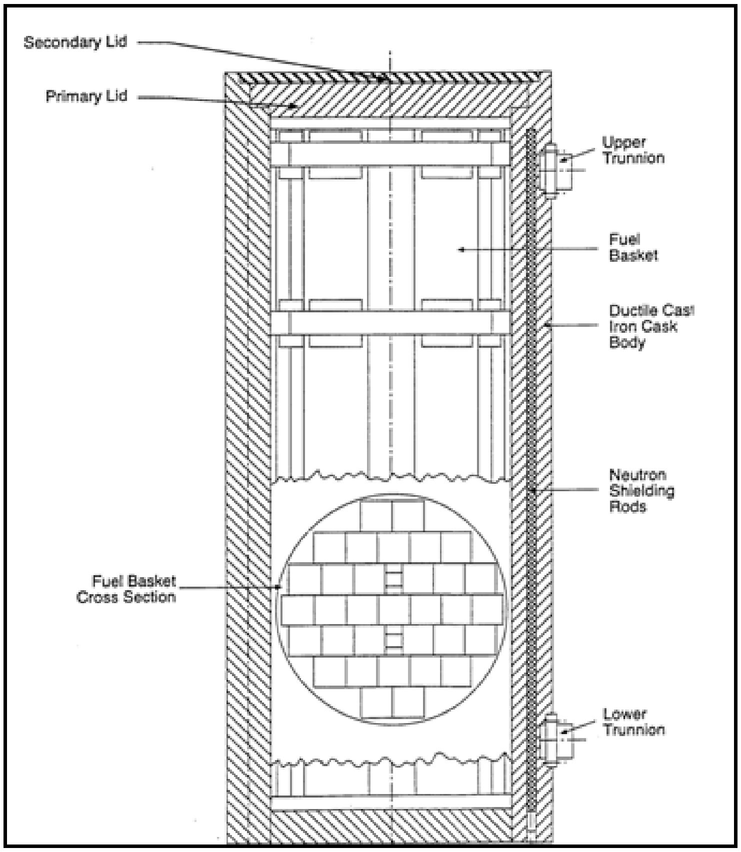

Figure 4-4
CASTOR X/33 Metal Storage Cask [NEI 1998]

4.3.4 Holtec International, Inc.

Holtec International (Holtec) has two related DPC-based system designs that have been approved by the NRC. The HI-STAR 100 is a metal DPC-based system that can be used for both storage and transportation. It utilizes a metal cask for both storage and transport that contains the DPC. The HI-STORM 100 consists of a vertical concrete storage overpack that houses the DPC. The DPC designs can be transferred from the storage system to the HI-STAR transport cask for shipment offsite. A variation of the HI-STORM system that is stored in an underground configuration, the HI-STORM 100U, has been certified by NRC for storage. Table 4-3 summarizes the physical parameters associated with the Holtec HI-STAR and HI-STORM storage technologies that have been certified by the NRC.

Table 4-3
Holtec International - Dry Storage System Parameters

Description	HI-STAR 100		HI-STORM 100			HI-STORM 100U		
Fuel Type	**PWR**	**BWR**	**PWR**		**BWR**	**PWR**		**BWR**
# Assemblies	24	68	24	32	68	24	32	68
Maximum Heat Load (kilowatts)	19	18.5	27	36.9	36.9	27	36.9	36.9
Minimum Cooling Time (Years)	5	5	3					
Maximum Fuel Burnup (GWd/MTU)	42.1	37.6	68.2 (PWR), 65 (BWR)					
Dual Purpose Canister Length [m. (in.)] Outer Diameter [m. (in.)]	4.83 (190.3) 1.74 (68.5)							
Transfer Cask Length [m. (in.)] Outer Diameter [m. (in.)] Loaded Weight [t. (lbs)] (with water) HI-TRAC 100 Hi-TRAC 125 HI-TRAC 125D	4.98 – 5.12 (196.25 - 201.5) 2.32 – 2.4 (91.25 - 94.625) 87.09 – 90.26 (192,000 -199,000) 107.23 – 111.13 (237,500 - 245,000) 103.65 – 107.05 (228,500 - 236,000)							
Storage Cask Length [m. (in.)] Outer Diameter [m. (in.)] Loaded Weight [t. (lbs)]	5.87 – 6.17 (231-243) 3.37 (132.5) 163.29 (360,000)					See note 1. 5.79 (228) minimum 2.18 (86) 66.68 (147,000)		
NRC Part 72 Docket	72-1008		72-1014					
Note 1: The HI-STORM 100U utilizes a Vertical Ventilated Module for underground storage. The dimension length dimensions provided are the minimum dimensions of the Cavity Enclosure Container (CEC) plus the approximately length of the closure lid above the CEC. A general licensee may increase this length provided that a 10CFR72.48 analysis is completed and documented. For more detailed information refer to the licensing dockets associated with these storage systems.								

4.3.4.1 HI-STAR 100

The HI-STAR 100 system consists of the following components: multi-purpose canister (MPC), which contains the fuel; and the HI-STAR 100 overpack, which is used to load, unload, transfer, and store SNF contained in the MPC. The HI-STAR 100 storage system has been certified to store up to 24 PWR fuel assemblies or 68 BWR fuel assemblies. The HI-STAR 100 was certified by NRC for storage under 10CFR72 (CoC #72-1008) effective on October 4, 1999. The HI-STAR metal overpack has also been certified for transport with CoC #71-9261. The HI-STAR 100 system is used for storage of SNF at the Dresden and Hatch ISFSIs. Figure 4-5 presents a diagram of the HI-STAR 100 system.

The MPC serves as the confinement system for the stored fuel and is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. All MPC components that may come into contact with SNF pool water are made of stainless steel or passivated aluminum/aluminum alloys such as the neutron absorbers. The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components. All confinement boundary components are made entirely of stainless steel. The honeycombed basket, which is equipped with neutron absorbers, provides criticality control.

There are three MPC designs that have been approved for storage in the HI-STAR 100: MPC-24, for storage of PWR SNF; and the MPC-68 and MPC-68F, for storage of BWR SNF. The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC. The MPC models all have the same external diameter. The approved contents for each of the MPC designs are specified in Appendix B of the HI-STAR 100 CoC.

The HI-STAR 100 overpack is a heavy-walled steel cylindrical vessel used that has been certified for both storage and transport. The overpack provides a helium retention boundary that is formed by an inner shell welded at the bottom to a cylindrical forging and, at the top, to a heavy main flange with bolted closure plate. The inner surfaces of the HI-STAR overpack form an internal cylindrical cavity for housing the MPC.

The HI-STAR 100 overpack includes inner, intermediate and enclosure shells that form the body of the package. A detailed description of the overpack is contained in the package SAR. The primary shielding for the HI-STAR 100 storage system is contained in the overpack and consists of neutron shielding and additional layers of steel for gamma shielding. Neutron shielding is provided around the outer circumferential surface of the overpack. Gamma shielding is provided by the overpack inner, intermediate, and enclosure shells with additional axial shielding provided by the bottom plate and the closure plate. Lifting trunnions are attached to the overpack top flange forging for lifting and for rotating the cask body between vertical and horizontal positions. Pocket trunnions are welded to the lower side of the overpack to provide a pivoting axis for rotation.

Once the SNF is loaded and the MPC and cask are sealed, the HI-STAR 100 System can be used for at-reactor storage in an ISFSI and for transport of SNF offsite.

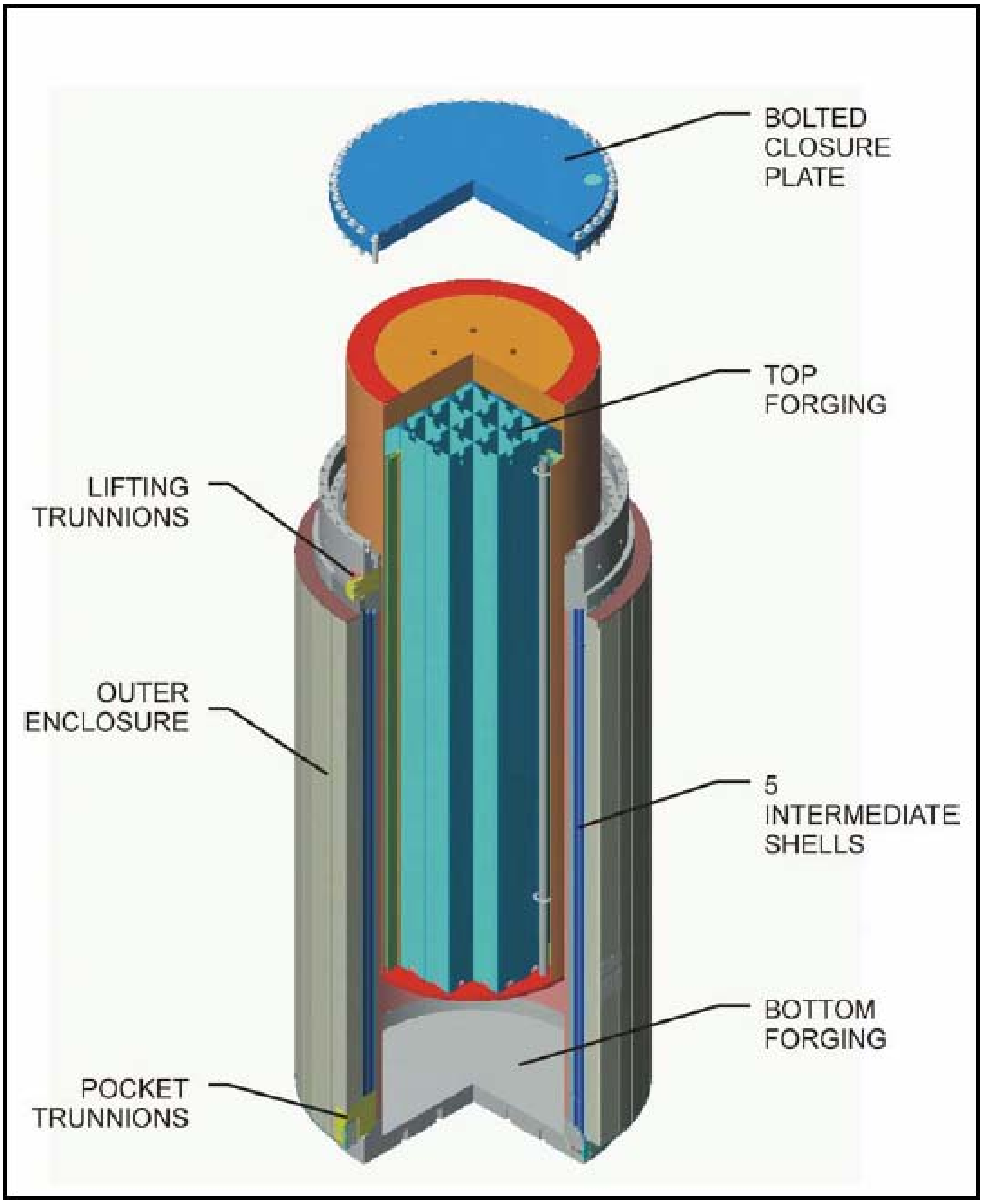

Figure 4-5
HI-STAR 100 Overpack with MPC Partially Inserted, CoC #72-1008 [Singh 2004]

4.3.4.2 HI-STORM 100

The HI-STORM 100 system consists of: MPCs, which contain the fuel; a HI-STORM concrete storage overpack, which contains the MPC during storage; and the HI-TRAC transfer cask, which contains the MPC during loading, unloading and transfer operations. The cask stores up to 32 PWR fuel assemblies or 68 BWR fuel assemblies. The HI-STORM 100 system was certified under 10CFR72 (CoC #72-1014) on June 1, 2000. As shown in Table 2-2, the HI-STORM 100 system is used for SNF storage at the following ISFSIs: Arkansas Nuclear One, Dresden, Hatch, Trojan, Columbia, FitzPatrick, Sequoyah, Farley, Browns Ferry, Quad Cities, River Bend, Hope Creek, Grand Gulf, Indian Point, Vermont Yankee, and Diablo Canyon. Figure 4-6 presents the HI-STORM 100 concrete overpack with a MPC.

As in the HI-STAR 100 system, the MPC serves as the confinement system for the stored fuel and is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. The MPC features are the same as the MPC features identified in Section 4.3.4.1 for the HI-STAR 100 system. There are five PWR MPC designs that have been approved for storage in the HI-STORM 100 storage overpack: the MPC-24, MPC-24E, MPC-24EF, MPC-32, and MPC-32F; and three BWR MPC designs: MPC-68, MPC-68F, and MPC-68FF. All eight MPC models have the same external diameter. The approved contents for each of the MPC designs are specified in Appendix B and Appendix B-100U to the HI-STORM CoC.

The HI-TRAC transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the SNF pool to the storage overpack, and from the storage overpack to a HI-STAR transport cask for transport offsite. The transfer cask is a multi-walled cylindrical vessel with a neutron shield attached to the exterior. There area two HI-TRAC transfer cask sizes: a 125 ton HI-TRAC and a 100 ton HI-TRAC. The weight designation indicates the approximate weight of a loaded transfer cask during any loading, unloading, or transfer operation.

The HI-STORM 100 or HI-STORM 100S storage overpack provides shielding and structural protection of the MPC during storage. The HI-STORM 100S includes a modified lid which incorporates the air outlet ducts into the lid, allowing the overpack body to be shortened. The HI-STORM storage overpack is a vertical, heavy-walled steel and concrete, cylindrical vessel. The overpack side wall is constructed of un-reinforced concrete that is enclosed between inner and outer carbon steel shells. The overpack has four air inlets at the bottom and four air outlets at the top to allow air to circulate naturally to provide cooling to the MPC. There are also two additional HI-STORM overpacks, the HI-STORM 100A and 100SA that include features that allow the storage overpack to be anchored to the concrete storage pad at sites with high seismic conditions.

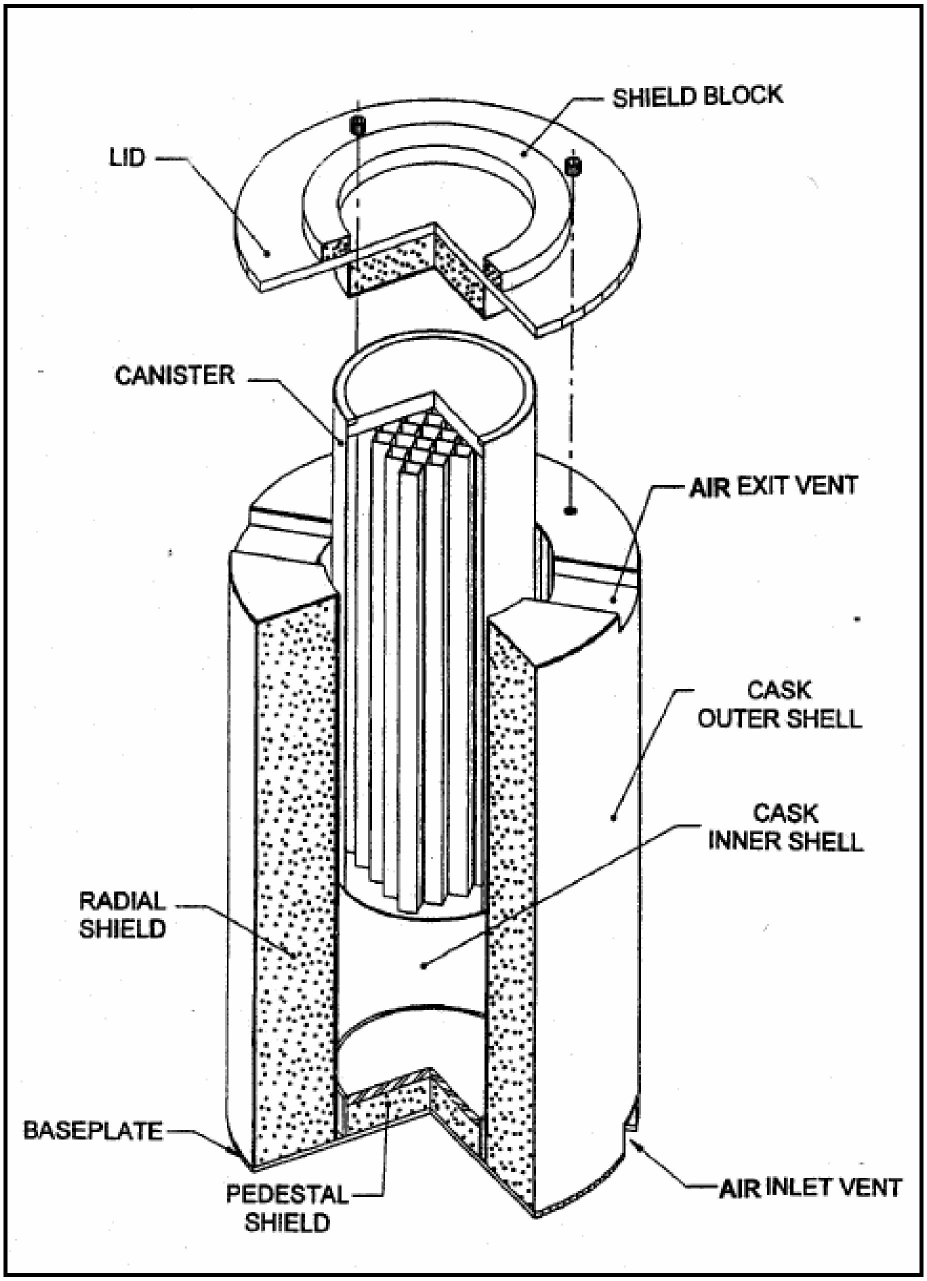

Figure 4-6
Cross Section of the HI-STORM 100 Overpack with MPC [NRC 2001b]

The HI-STORM system also includes a storage system that can be used in an underground configuration, the HI-STORM 100U. The HI-STORM 100U storage module (Vertical Ventilated Module, or VVM) is an air-cooled vault that relies on vertical ventilation and provides structural protection of the MPC during storage. Air inlets and outlets allow air to circulate naturally through the underground cavity to cool the MPC inside. The VVM is a subterranean steel structure that is seal welded to prevent ingress of any groundwater from the surrounding subgrade. The surrounding subgrade and a top surface pad provide radiation shielding. Figure 4-

7 presents a graphic that shows a vertical cask transporter moving a loaded HI-TRAC transfer cask to a HI-STORM 100U storage module for transfer of a loaded MPC.

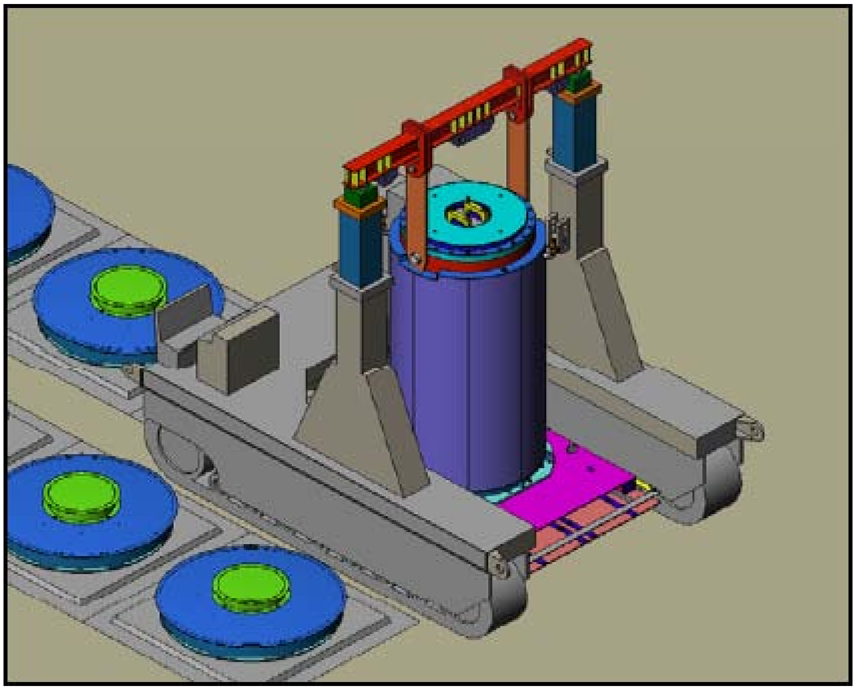

Figure 4-7
HI-STORM 100U Vertical Cask Transporter Moving Loaded HI-TRAC Transfer Cask to HI-STORM 100U Storage Module [Singh 2010]

In September 2009, Holtec submitted an application to the NRC for certification of the HI-STORM FW (flood and wind) system, under a new NRC Docket 72-1032. The HI-STORM FW is designed to store 37 PWR or 89 BWR assemblies in a MPC-37 or MPC-89 canister, respectively. Similar in design to the HI-STORM 100 system, it includes seal-welded MPCs that contain the SNF; a vertically ventilated concrete and steel storage overpack that houses an MPC during storage; and a variable weight transfer cask (HI-TRAC VW) that is used to contain the MPC during loading, unloading and transfer operations. The MPC baskets utilize Metamic-HT as the neutron absorber. The MPC-37 canister is designed with a maximum heat load of 47.05 kW, maximum fuel burnup of 68.2 GWd/MTU, and a minimum cooling time of 3 years. The MPC-89 canister is designed with a maximum heat load of 46.36 kW, maximum fuel burnup of 65 GWd/MTU, and a minimum cooling time of 3 years.

4.3.5 NAC International, Inc.

NAC International (NAC) has developed numerous SNF storage and transport systems that have been approved by the NRC through topical reports and the CoC process. During the 1990s, NAC developed the NAC S/T system of casks, some of the first metal storage casks approved for SNF storage in the U.S. The NAC I28 S/T is discussed below since one of these systems has been loaded with commercial SNF and is currently in storage at an ISFSI.

NAC has three DPC-based technologies that have been certified by NRC under Part 72 for storage. The NAC Multi-Purpose Canister (NAC MPC) system, NAC Universal Multi-Purpose Canister System (NAC UMS) system, and the Modular Advanced Generation, Nuclear All-Purpose Storage (MAGNASTOR) system. The NRC has also approved transportation packages to transport the DPC designs from the NAC MPC and NAC UMS systems. NAC plans to submit an application to the NRC for certification of a transport cask for the MAGNASTOR system.

Table 4-4
NAC International - Dry Storage System Parameters

Description	NAC MPC		NAC UMS		MAGNASTOR	
Fuel Type	**Yankee PWR**	**CY PWR**	**PWR**	**BWR**	**PWR**	**BWR**
# Assemblies	36	24-26	24	56	37	87
Maximum Heat Load (kilowatts)	12.5	16-18	23	23	35.5	33
Minimum Cooling Time (Years)	8	6	5	5	4	4
Maximum Fuel Burnup (GWd/MTU)	36	40	60	45	60	60
Dual Purpose Canister Length [m.] [(in.)] Outer Diameter [m. (in.)]	3.11 (122.5) 1.79 (70.6)		4.45-4.84 (175.1-190.4) 1.70 (67.1)		4.69 – 4.87 (184.8-191.8) 1.83 (72)	
Transfer Cask Length [m. [(in.)] Outer Diameter [m.] [(in.)] Loaded Weight with water [t.] [(lbs)]	3.39 (133.4) 2.20 (86.5) 61.45 (135,473)	4.14 (162.9) 2.26 (89) 78.34 (172,708)	4.5 – 4.89 (177.3-192.6) 2.16 (85.3) 90.6 – 97.20 (199,800 – 214,300)		4.84 (190.62) 2.24 (88) 104.1 (229,500)	
Storage Cask Length [m. (in.)] Outer Diameter [m.] [(in.)] Loaded Weight [t.] [(lbs)]	3.25 (128) 4.06 (160)		3.45 (136) 5.31 – 5.70 (209.2-224.5) 140.0 – 146.9 (308,700 – 323,900)		3.45 (136) 5.54 – 5.72 (218.3-225.3) 145.15 – 145.6 (320,000 – 321,000)	
NRC Part 72 Docket	72-1025		72-1015		72-1031	
For more detailed information refer to the licensing dockets associated with these storage systems.						

4.3.5.1 NAC I28 S/T

NAC International, Inc. has developed four variations of the NAC S/T cask: NAC S/T (CoC #72-1002), NAC-C28 (CoC # 72-1003), NAC-I28, and NAC-STC. NAC S/T CoC #72-1002 and NAC-C28 CoC #72-1003 expire on August 31, 2010. The NRC approved the NAC-I28 cask design for storage in February 1990. The NRC approved the NAC STC TSAR for storage in July 1995 and issued a CoC #71-9235 for transport under 10CFR71.

The NAC-I28 is designed to vertically store 28 intact PWR fuel assemblies, as shown in Figure 4-8. The NAC S/T casks are made of a multi-wall cylinder with a thick inner shell and outer shell both made of stainless steel, separated by lead. The overall dimensions of the cask are 4.6 m. (15.1 ft.) long and 2.4 m. (7.9 ft.) in diameter. The inner and outer shells are connected to each other on both ends by austenitic stainless steel rings and plates. The upper end of the cask is sealed by a stainless steel bolted closure lid. The closure lid uses a double barrier seal system with two metallic O-ring seals. A neutron shield cap encased in stainless steel is placed on top of the cask and may be welded to the cask body. Gamma shielding is provided by the lead wall, and neutron shielding by a layer of a poured-on-place solid borated synthetic polymer which surrounds the outer shell along the cavity region, and is enclosed by a stainless steel shell with end plates that are welded to the outer shell.

The bottom of the cask is sealed by a stainless steel plate with an outer closure plate, separated by lead for gamma shielding. Twenty-four explosively bonded copper/stainless fins are located within the radial neutron shield to enhance heat rejection. Six trunnions can be attached to the cask for lifting and rotating the cask.

The fuel basket is a right circular cylinder configuration with 28 aluminum fuel tubes that are separated and supported by an aluminum and stainless steel grid of spacers and tie bars, with borated neutron poison material for criticality control in the basket assembly.

The NAC-I28 cask is used at Dominion’s Surry ISFSI, under a site-specific license.

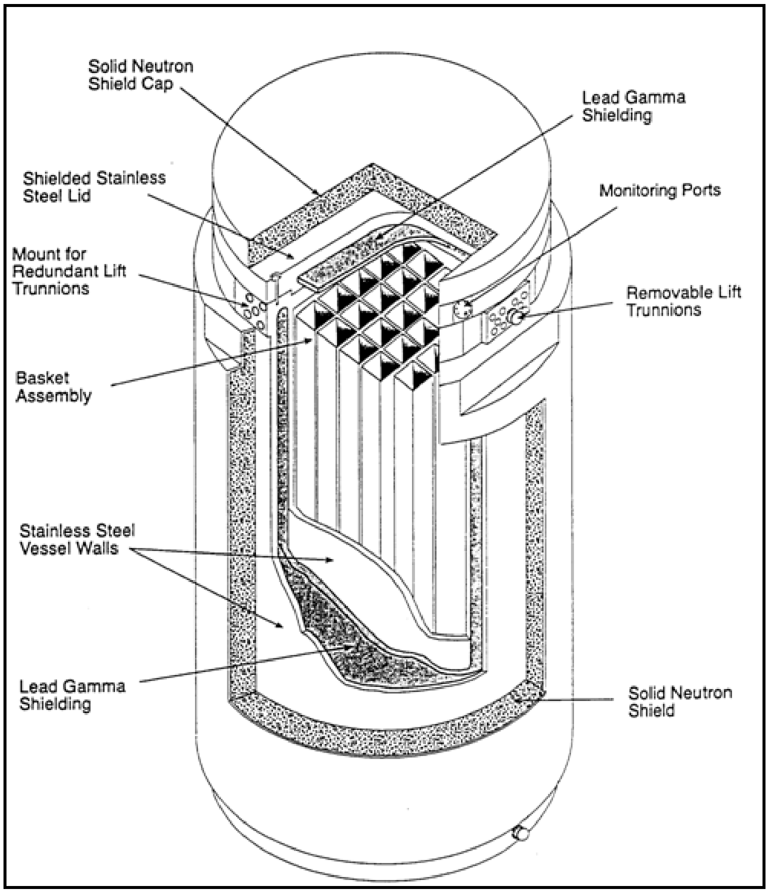

Figure 4-8
NAC S/T Metal Storage Cask [NEI 1998]

4.3.5.2 NAC MPC

The NAC Multi-Purpose Canister DPC system (NAC MPC) consists of a transportable storage canister, vertical concrete cask, and a transfer cask. The NAC MPC system received NRC storage certificate #72-1025, which expires on April 10, 2020. The principal components of the NAC-MPC storage system are the transportable storage canister (TSC), the vertical concrete cask (VCC), and the transfer cask. The dual purpose TSC is licensed for transport in the NAC-STC transportation cask, CoC 71-9235.

The TSC assembly consists of a right circular cylindrical stainless steel shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The cylindrical shell, plus the bottom plate and lids, constitutes the confinement boundary. The stainless steel fuel basket is a right circular cylinder configuration with up to 36 fuel tubes (for Yankee Class fuel) and up to 26 fuel tubes (for Connecticut Yankee fuel) laterally supported by a series of stainless steel support disks, which are retained by spacers on radially-located tie rods. The SNF assemblies are contained in stainless steel fuel tubes. The square fuel tubes are encased with Boral sheets on all four sides for criticality control. An amendment has been submitted to the NRC that would allow storage of SNF from the LaCrosse BWR, owned by Dairyland Power Cooperative.

The VCC serves as the storage overpack for the TSC and it provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the TSC during storage. The storage cask is fabricated from reinforced concrete with a structural steel liner. The concrete wall and steel liner provide neutron and gamma radiation shielding. The storage cask contains an annular air passage to allow the natural circulation of air around the canister. The top of the storage cask is closed by a shield plug and lid. The lid is bolted in place and has tamper indicating seals on two of the bolts.

A transfer cask is used as the lifting device to vertically transfer the loaded canister from the SNF storage pool to the VCC and from the VCC to the NAC STC for transportation off site. The transfer cask has retractable bottom shield doors to facilitate transfer of the loaded canister into the storage or transport casks.

Figure 4-9 is a photograph of the NAC MPC system in storage at the Connecticut Yankee ISFSI.

Figure 4-9
NAC MPC Dual-Purpose Canister System, CoC #72-1025, Connecticut Yankee ISFSI[7]
[Hoedeman 2008]

[7] http://www.connyankee.com/html/fuel_storage.html

4.3.5.3 NAC UMS

The NAC Universal MPC System (UMS) has been certified for the storage and transport of 24 PWR or 56 BWR SNF assemblies. The storage component is designated the Universal Storage System and includes a TSC with a welded closure, a vertical concrete storage cask (VCC), and a transfer cask. The NAC UMS system received a storage certificate #72-1015, which expires on November 20, 2020. The TSC is licensed for transport in the UMS Universal Transport Cask Package, CoC 71-9270.

The TSC is the confinement system for the stored fuel. The TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The cylindrical shell, plus the bottom plate and lids, constitute the confinement boundary. The stainless steel fuel basket is a right circular cylinder configuration with either 24 (PWR) or 56 (BWR) stainless steel fuel tubes laterally supported by a series of stainless steel carbon steel support disks. The square fuel tubes in the PWR basket include Boral sheets on all four sides for criticality control. The square fuel tubes in the BWR basket may include Boral sheets on up to two sides for criticality control. Aluminum heat transfer disks are spaced midway between the support disks and are the primary path for conducting heat from the SNF assemblies to the TSC wall for the PWR basket. There are three TSC configurations of different lengths for PWR and site-specific contents and two TSC configurations of different lengths for BWR contents. BWR SNF rods/assemblies must be intact. PWR and site-specific SNF rods/assemblies may be intact or damaged, with damaged fuel rods/assemblies placed in a fuel can. A canister has also been certified for the storage of GTCC waste.

The storage overpack, designated the VCC, provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the canister during storage. The concrete wall and steel liner provide the neutron and gamma radiation shielding for the storage cask. The concrete cask has an annular air passage to allow the natural circulation of air around the canister to remove the decay heat from the SNF stored in the TSC. The top of the concrete cask is closed by a shield plug and lid which incorporates a carbon steel plate as gamma radiation shielding as well as solid neutron shielding material. A carbon steel lid that provides additional gamma radiation shielding is installed above the shield lid. The lid is bolted in place and has tamper indicating seals on two of the installation bolts. There are three VCC configurations of different lengths for PWR and site-specific contents and two VCC configurations of different lengths for BWR contents.

The transfer cask is used for the vertical transfer of the TSC between work stations and the VCC, or UMS transport cask. The transfer cask incorporates a multi-wall design and a top retaining ring, which is bolted in place to prevent a loaded canister from being inadvertently removed through the top of the transfer cask. The transfer cask has retractable bottom shield doors to facilitate the transfer of the TSC from the transfer cask into the VCC or UMS transportation cask. Figure 4-10 shows the transfer configuration in which a transfer cask transfers a loaded TSC to a VCC for the UMS system.

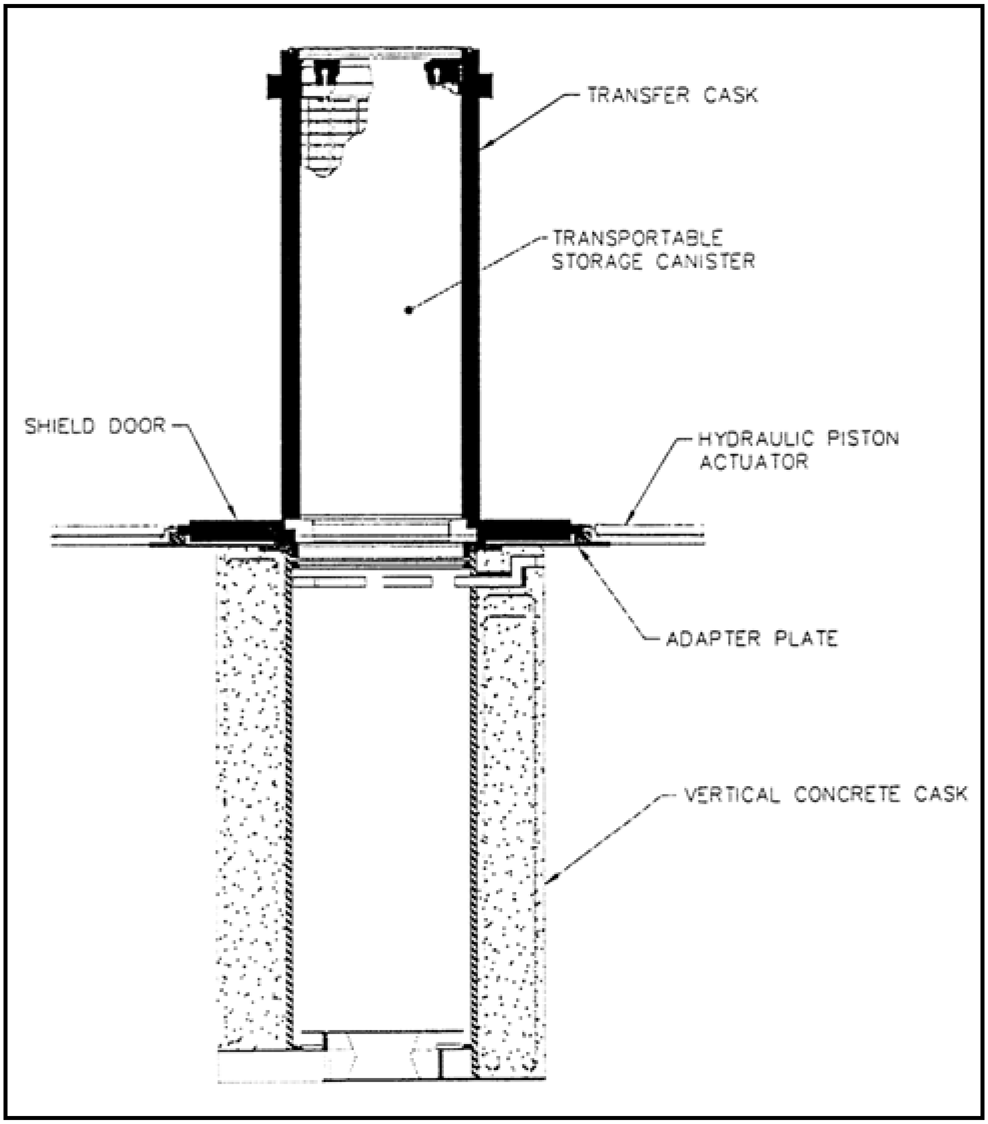

Figure 4-10
NAC UMS Dual-Purpose Storage System, CoC #72-1015 [NEI 1998]

4.3.5.4 NAC MAGNASTOR

The NAC MAGNASTOR System is a DPC-based system with a 37 PWR or 87 BWR assembly capacity. The storage component includes a TSC with a welded closure, a concrete storage cask, and a transfer cask. The NAC MAGNASTOR system received a storage certificate #72-1031, which expires on February 4, 2029. NAC intends to license the MAGNASTOR TSC for transport in a compatible MAGNASTOR transport cask.

The TSC provides the confinement system for the stored fuel. The TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a closure lid, a closure ring, and two sets of redundant penetration port covers. The cylindrical shell plus the bottom plate, closure lid, and welded inner port covers are stainless steel and constitute the confinement boundary. The coated carbon steel fuel basket is a circular cylinder configuration with either 37 PWR or 87 BWR fuel assembly locations. The fuel assembly locations in the PWR and BWR baskets include neutron absorber panels on up to four sides for criticality control. Each neutron absorber panel is covered by a stainless steel sheet to protect the material during fuel loading and unloading, and to maintain it in position.

The concrete storage cask is the storage overpack for the TSC and provides structural support, shielding, protection from environmental conditions, and natural convection cooling of the TSC during long-term storage. The concrete cask is a reinforced concrete structure with a carbon steel inner liner. The concrete cask has an annular air passage to allow a passive convection air flow around the TSC. The top of the concrete cask is closed by a carbon steel and concrete lid bolted in place.

The transfer cask provides shielding during TSC movements between work stations, the concrete cask, or the transport cask. It is a multiwall (steel/lead/NS-4-FR/steel) design with retractable (hydraulically operated) bottom shield doors that are used during loading and unloading operations. A rendering of the MAGNASTOR storage configuration and a cut-away of the storage overpack is provided in Figure 4-11.

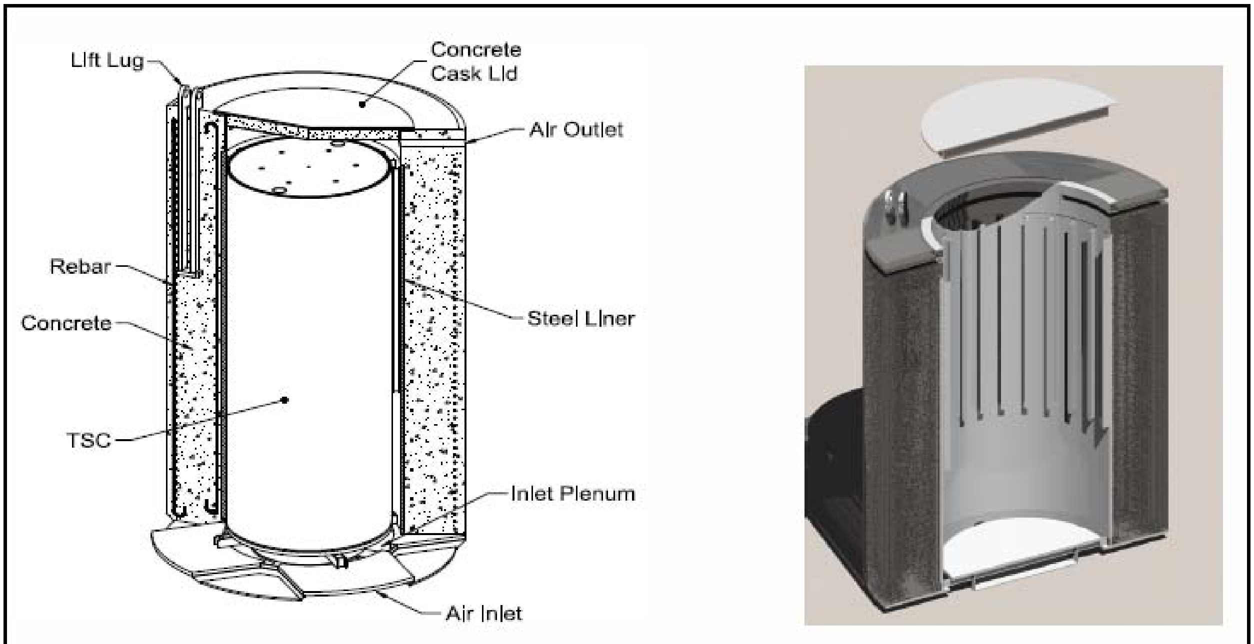

Figure 4-11
NAC MAGNASTOR Dual-Purpose Technology, CoC # 72-1031 [Pennington 2005]

4.3.6 Transnuclear, Inc.

Transnuclear, Inc. has two dry storage product lines for the U.S. market: the family of metal casks including the TN-24P, TN-40, TN-40HT, TN-32, and TN-68; and the NUHOMS system including the standardized NUHOMS system, the Advanced NUHOMS system, and the NUHOMS HD system. Both of these dry storage technologies are summarized below.

4.3.6.1 TN Metal Casks

The Transnuclear family of metal casks, TN-24P, TN-40, TN-32, and TN-68 are metal storage casks designed to store 24 PWR, 40 PWR, 32 PWR, and 68 BWR SNF assemblies, respectively. The TN-24P cask received a CoC (CoC# 72-1005) from NRC for storage of PWR SNF. The TN-40 cask design was approved as part of the SAR for the Prairie Island ISFSI (Docket 72-10). The TN-40HT cask is currently under review by the NRC as part of a license amendment for the Prairie Island ISFSI. The NRC approved the TN-32 cask design for storage (CoC# 72-1021) in April 2000. The TN-68 dual-purpose cask was certified by the NRC in May 2000 (CoC# 72-1027). The TN-68 cask is also licensed for transport (CoC# 71-9293).

The TN-32 cask is in use at Dominion’s Surry and North Anna ISFSIs, and Duke Energy’s McGuire ISFSI. The TN-40 cask is in use at Xcel Energy’s Prairie Island ISFSI. The TN-68 cask is used at Exelon’s Peach Bottom ISFSI. Table 4-4 provides an overview of the TN metal casks that are currently loaded in the U.S. or that are under NRC review. As an example of the TN family of casks, the TN-68 metal cask design described below.

Table 4-5
Transnuclear – TN Metal Cask Dry Storage System Parameters

Description	TN-32	TN-40	TN-40HT	TN-68
Fuel Type	**PWR**	**PWR**	**PWR**	**BWR**
# Assemblies	32	40	40	68
Maximum Heat Load (kilowatts)	32.7	27	32	30
Minimum Cooling Time (Years)	7	10	18	7
Maximum Fuel Burnup (GWd/MTU)	40	45	60	60
Storage Cask				
Length [m] (in.)	4.9 (184.0)	4.4 (175)	4.6 (181.75)	5.5 (215)
Length with protective cover [m. (in.]	5.13 (201.88)	5.13 (202.0)	5.07 (199.6)	
Outer Diameter [m. (in.)]	2.48 (97.75)	2.53 (99.52	2.57 (101)	2.49 (98)
Loaded Weight [t.]	104.8	102.5	109.95	104.3
[(lbs)]	(231,000)	(226,000)	(242,400)	(230,000)
NRC Part 72 Docket	72-1021	SNM-2506	SNM-2506 Under review	72-1027

The TN-68 cask body is a right circular cylinder composed of the following components: confinement vessel with bolted lid closure, basket for fuel assemblies, gamma shield, trunnions, neutron shield, pressure monitoring system, and weather cover. The TN-68 metal dual purpose cask is shown in Figure 4-12.

The confinement vessel consists of an inner shell which is a welded, carbon steel cylinder with an integrally welded, carbon steel bottom closure; a welded flange forging; a flanged and bolted carbon steel lid with an inner metallic seal; and vent and drain covers with closure bolts and inner metallic seals.

The basket consists of an assembly of stainless steel cells that are welded to stainless steel plates. Above and below the stainless steel plates are slotted neutron absorber plates which form an egg-crate structure. The neutron absorber plates provide heat conduction paths from the fuel assemblies to the cask cavity, and the neutron absorber plates provide criticality control.

The gamma shield encloses the confinement vessel and consists of an independent shell and bottom plate of carbon steel which is welded to the closure flange. An optional carbon steel gamma shield ring may be used and is installed above the neutron shield. Gamma shielding is also provided by the confinement lid. The radial neutron shield consists of a borated polyester resin compound which surrounds the gamma shield. The resin compound is cast into long, slender aluminum containers which are enclosed in a smooth outer steel shell. The aluminum containers provide a conduction path for heat transfer from the cask body to the outer shell. Axial neutron shielding is provided by a polypropylene disk placed on the cask lid.

There are four trunnions attached to the cask body. The top trunnions are used for lifting and the bottom trunnions may be used for rotating the unloaded cask. The overpressure monitoring system provides continuous monitoring of the pressure in the interspace between the inner and outer seals on the lid, vent, and drain port covers. The overpressure monitoring system consists of a tank filled with helium, pressure transducers or switches, and associated tubing, fittings, and valves.

The weather cover with an elastomeric seal provides weather protection for the closure lid and seal components, the top neutron shield, and the overpressure system.

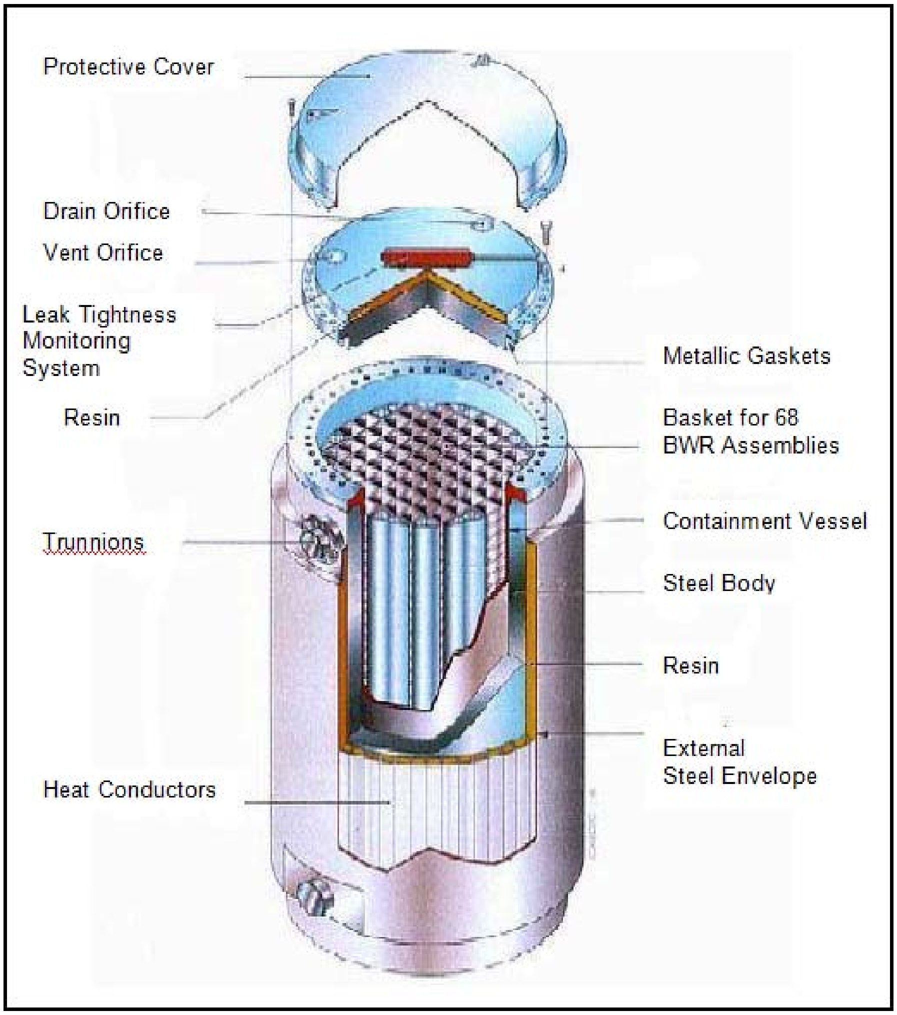

Figure 4-12
TN-68 Transport/Storage Cask, CoC #72-1027 [Bailly 2005]

4.3.6.2 NUHOMS Systems

The NUHOMS family of concrete modular storage systems are designed to horizontally store PWR and BWR SNF assemblies. Its main components include a stainless steel dry shielded canister (DSC) with an internal fuel basket, a concrete horizontal storage module (HSM) that protects the DSC and provides radiological shielding, a transfer cask used to transport the DSC to the HSM and to provide shielding during transfer, and a hydraulic ram system (HRS) used to insert the DSC into the HSM. Table 4-5 summarizes the physical parameters associated with the NUHOMS casks, including parameters for the various DSC designs, transfer cask designs, and HSM designs that have been certified by NRC under the various NUHOMS 10CFR72 dockets.

Table 4-6
Transnuclear – NUHOMS Dry Storage System Parameters

Description	NUHOMS DSC Designs								
	-24PT1	-24PT4	-24PHB	-24PTH	-32PT	32PTH	-32PTH1	-61BT	-61BTH
Fuel Type	PWR	PWR	PWR	PWR	PWR	PWR	PWR	BWR	BWR
# Assemblies	24	24	24	24	32	32	32	61	61
Maximum Heat Load (kilowatts)	14	24	24	40.8	24	34.8	40.8	18.3	31.2
Minimum Cooling Time (Years)	10	5	5	3	5	5	3	4	3
Maximum Fuel Burnup (GWd/MTU)	45	60	55	62	45	60	62	40	62
Dry Shielded Canister Length [m.] [(in.)] Outer Diameter [m.] [(in.)]	4.7 (186.5) 1.71 (67.2)	4.99 (196.3) 1.71 (67.2)	4.73 (186.17) 1.71 (67.2)	4.74, 4.9 (186.5 -193) 1.58 (62.2)	4.74, 4.9 186.5 - 193 1.71 (67.2)	4.9 (193) 1.77 (69.75)	4.7 - 5.04 (185.8 - 198.5) 1.77 (69.75)	4.98 (196) 1.7 (67)	4.98 (196) 1.7 (67)
NRC Part 72 Docket	72-1029	72-1029	72-1004	72-1004	72-1004	72-1030	72-1004	72-1004	72-1004

Table 4-6
Transnuclear – NUHOMS Dry Storage System Parameters (continued)

Description	NUHOMS Transfer Cask and HSM Descriptions		
Transfer Cask Design: Length [m. (in.)] Outer Diameter [m. (in.)] Required crane capacity [t. (tons)] Weight loaded [t. (lbs)]	**OS200** 206.72 92.11 125 Depends on DSC and lifting yoke	**OS197** 207.22 85.5 100 Depends on DSC and lifting yoke	**OS187H** 207.22 85.5 100 196,500

Description	HSM-102		HSM-H/HS
Horizontal Storage Module:	**PWR**	**BWR**	
Length [m. (ft)]	5.80 (19)	6.04 (19.8)	6.2 (248)
Width [m. (ft)]	2.96 (9.7)	2.96 (9.7)	2.9 (9.7)
Height [m. (in.)]	4.57 (180)	4.57 (180)	5.64 (222)
Loaded Weight [t. (lbs)]	159.3 – 165,3 (351,200 -364,400)	159.4 (351,400) (61BT)	180.6 (418,000) (32 PTH1)
DSCs Stored:	24P,24PT2,32PT,24PHB	52B,61BT	24PTH, 61BTH, 32PTH, 32PTH1, 32PTH Type 1

Several NUHOMS systems have been approved by the NRC. The NUHOMS-07P was approved under the site-specific license for the H.B. Robinson ISFSI (Docket 72-1022). The NUHOMS-24P, a storage only system was approved under the site-specific license for the Oconee ISFSI (Docket 72-1004). As shown in Table 4-6, the NUHOMS system has also been approved for use under a general license: the standardized NUHOMS system was certified in January 1995 (CoC #1004) and can be used with the following DSC designs: NUHOMS-24P, -52B, -61BT, -32PT, -24PHB, -24PTH, -32PTH1 and - 61BTH.

An Advanced NUHOMS system (CoC# 72-1029) was certified in February 2003 for storage of the NUHOMS-24PT1 and 24PT4 DSCs. The NUHOMS HD system (CoC# 72-1030) was certified in January 2007 for storage of the NUHOMS-32PTH DSC.

As shown in Table 2-2, the NUHOMS technology is used for storage of SNF at the following ISFSI sites: H.B. Robinson, Oconee, Calvert Cliffs, Palisades, Point Beach, Susquehanna, Rancho Seco, Oyster Creek, San Onofre, Duane Arnold, Millstone, Surry, North Anna, Fort Calhoun, Limerick, St. Lucie, Seabrook, Monticello, and Kewaunee. As an example, the configuration of the Standardized NUHOMS system is described below and shown in Figure 4-13.

The DSC consists of the shell with integral bottom cover plate, bottom shield plug or shield plug assemblies, ram/grapple ring, top shield plug or shield plug assemblies, top cover plate, and basket assembly. The shell length is fuel-specific. The internal basket assembly aids in the insertion of the fuel assemblies, enhances subcriticality during loading operations, and provides structural support during a hypothetical drop accident. The DSC is designed to slide from the transfer cask into the HSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces.

There are several basket assembly configurations associated with the various DSC designs, as described in more detail in the Standardized NUHOMS CoC and SAR. The basket assembly configuration for the 32PT, and 32PTH1 DSC consist of welded stainless steel plates or tubes that make up a grid of fuel compartments supported by aluminum basket rails, and are designed to accommodate 32 PWR assemblies.

The HSM is a reinforced concrete unit with penetrations located at the top and bottom of the walls for air flow. The penetrations are protected from debris intrusions by wire mesh screens during storage operation. The DSC Support Structure, a structural steel frame with rails, is installed within the HSM. An alternate version of the HSM-H design has been provided to allow the use of the system in locations where higher seismic levels exist.

The TC is used for loading and transfer operations within the SNF pool building and to transfer loaded DSCs to or from the HSM. The TC is a cylindrical vessel with a bottom end closure assembly and a bolted top cover plate. Two upper lifting trunnions are located near the top of the cask for downending or uprighting and for lifting of the cask in the SNF pool building. The lower trunnions, located near the base of the cask, serve as the axis of rotation and as supports during transport to the HSM.

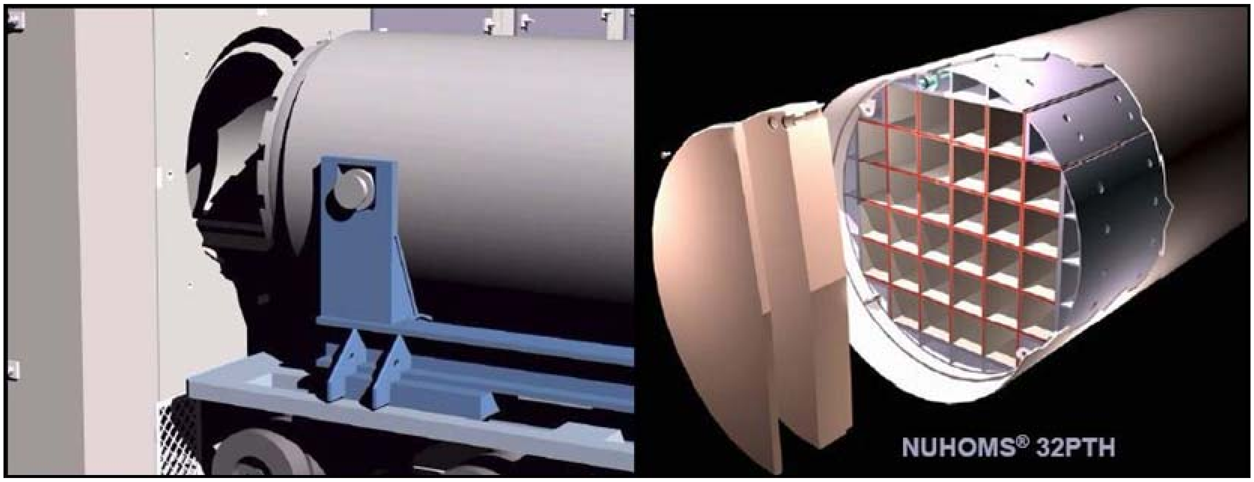

Figure 4-13
NUHOMS' Horizontal Storage Module, Transfer Trailer, and NUHOMS 32PTH DSC [Neider 2005, Neider 2008]

Transnuclear has announced that it plans to submit a license amendment to the NRC for the Standardized NUHOMS® CoC #72-1004 to include higher capacity -37PTH PWR and -69BTH BWR DSCs. The license amendment is expected to be submitted to NRC for review in late 2010. The 37PTH canister is designed with a maximum heat load of 30 kW, maximum fuel burnup of 65 GWd/MTU, and a minimum cooling time of 5 years. The 69BTH canister is designed with a maximum heat load of 26 to 32 kW, maximum fuel burnup of 65 GWd/MTU, and a minimum cooling time of 5 years.

4.3.7 Westinghouse MC-10

As shown in Figure 4-14, the Westinghouse MC-10 cask is a metal cask designed to vertically store 24 PWR SNF assemblies. The cask has a CoC (#72-1000) which expires August 30, 2010. The MC-10 consists of a thick-walled forged steel cylinder. The cask has a cylindrical cask cavity which holds the fuel basket. The overall length is 4.79 m. (15.7 ft.) and the diameter approximately 2.71 m. (8.9 ft.) including fins. The cask body is made of low alloy steel with forged steel walls and a bottom that provide gamma radiation shielding and structural integrity. A low alloy steel shield cover with a metallic O-ring seal provides the initial seal and shielding following fuel loading. A carbon steel cover, with a metallic O-ring seal, provides the primary containment seal. A seal lid provides a secondary containment seal. An additional cover, containing a BISCO NS-3 neutron-absorbing material, is welded over the first two seals to provide seal redundancy.

The inside surface of the cask is thermally sprayed with aluminum to provide corrosion protection. Twenty-four carbon steel heat transfer fins are welded axially along the outside of the cask wall. Carbon steel plates are welded between the fins to provide an outer protective skin. Neutron shielding is provided by a layer of BISCO NS-3 cured in the cavity between the cask wall and outer protective skin. Four trunnions are connected to the cask body for lifting and rotating the cask.

The basket assembly consists of 24 storage locations utilizing a honeycomb-type basket structure. Each of the 24 removable cell storage locations consists of an enclosure, neutron poison material, and wrappers. The upper ends of the enclosure walls are flared to facilitate fuel loading. Neutron absorbing material is attached to the enclosure walls and held in place with a stainless steel wrapper welded to the panel.

The MC-10 cask is in use at Dominion's Surry ISFSI under Surry's site specific license.

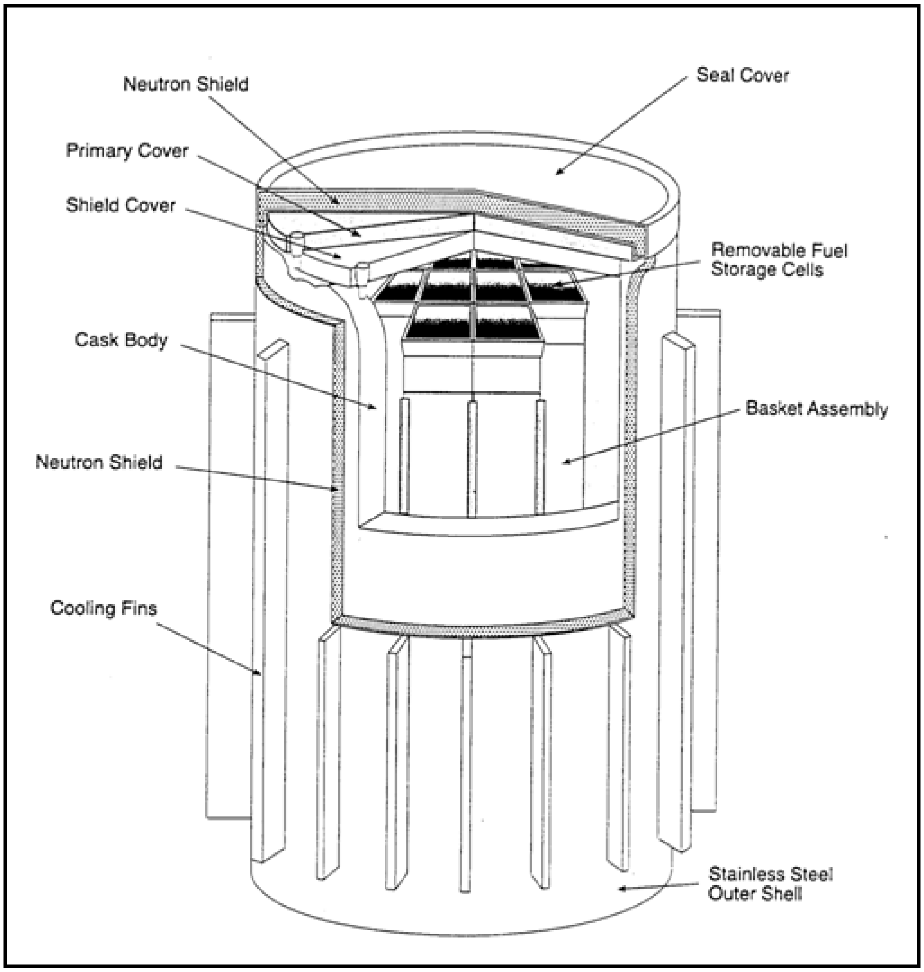

Figure 4-14
Westinghouse MC-10 Metal Storage Cask, CoC #72-1000 [NEI 1998]

5
SPENT FUEL STORAGE PLANNING

This section provides an overview of the process associated with the planning necessary for a SNF storage expansion project – whether an expansion of in-pool storage capacity or development of an at-reactor ISFSI. This overview includes typical steps in the planning process; the organizations that are typically involved in a SNF expansion project; recommended schedules for implementation of in-pool storage expansion, development of an at-reactor ISFSI, certification of a new dry storage technology; and amendment of an existing 10CFR72 CoC. These schedules are based on past industry experience and take into account the licensing status of the storage technologies under consideration.

5.1 At-Reactor Storage Expansion Planning Process

The planning process for increasing at-reactor SNF storage capacity should begin at least six years prior to the need for the additional capacity. A company will typically form a Project Team comprised of appropriate company personnel from areas in the company that will be impacted by a program to increase storage, including: fuel management, nuclear engineering, licensing, plant operations, radiation protection, quality assurance, plant management, purchasing, public affairs, design engineering, construction, security, and environmental engineering. The Project Team should identify a Project Manager to oversee the storage expansion campaign as well as dedicated personnel in each key area of the project. If the company has storage expansion projects planning at multiple sites, it may be beneficial to have one overall Program Manager as well as Project Managers at the individual sites. Adequate resources must be made available to implement the project. Early senior management involvement and support is necessary to ensure the timely outcome of a storage expansion project.

The first step in the planning process will be to identify when additional SNF storage capacity would be required; project the amount of additional storage capacity required to store SNF discharged through the end of the facility's existing (and extended) 10CFR50 operating license; and develop a milestone schedule to meet these requirements. Next, storage expansion alternatives would be identified and evaluated as described in Section 6. The evaluation of storage expansion alternatives would include the identification of site-specific limitations that may impact the various storage options (e.g., crane capacity limits or SNF storage pool structural limits). This evaluation would also include identification of applicable NRC regulations associated with the alternatives being considered; State or Local regulations that pertain to the storage options; and the potential for intervention in the licensing actions being considered.

Upon selection of a storage technology, a dedicated team would be identified to carry out the project, including interface with and review of all storage vendor work product and licensing submittals.

5.2 Scheduling Requirements

NRC's rules of engagement, discussed in Section 3.5, encourage licensees to meet with NRC staff at least two years in advance of submittal of an application for an ISFSI, storage and transport CoC, or for use of certified storage casks under a general license. NRC must ensure that it will have adequate resources in place to perform the required licensing reviews and ISFSI inspections. If an ISFSI licensee plans to utilize a dry storage technology that has not yet been certified by NRC, licensees should inform the NRC of plans to use such technology as early as possible so that appropriate resources can be allocated by NRC to ensure a timely review of the storage system. Typical planning schedules are identified below for: dry storage under a site-specific license; dry storage under a general license; certification of a dry storage technology under 10CFR72 or an amendment of an existing 10CFR72 CoC; and in-pool storage expansion alternatives.

5.2.1 Dry Storage

5.2.1.1 10CFR72 Site-Specific License

Figure 5.1 presents a hypothetical schedule for planning dry storage under a site-specific license. This schedule assumes a total planning horizon of approximately 6 years. Initial planning and procurement to implement dry storage using a site-specific license is expected to take approximately 12 months. Initial planning and procurement activities includes: performance of studies to determine viable storage expansion alternatives, evaluation of site-specific limitations, preparation of a request for proposal for dry storage systems, review and evaluation of proposals and selection of a storage technology.

Design, engineering and preparation of licensing documents for a site-specific license application are expected to take 12 to 18 months. This would include site investigations needed for selection of an ISFSI location, design and engineering activities for the ISFSI, and preparation of a LA, ER and associated documents.

The time required for NRC review of a site-specific license will depend on whether or not the NRC holds a hearing on the application and the extent to which the proposed ISFSI design or storage technology has already been reviewed by the NRC. Site-specific review times have ranged from 17 months to 9 years from submittal a LA for a site-specific license to issuance of a 10CFR72 license by NRC. The average review time is approximately 29 months. It would be prudent to plan for a period of three years for NRC review for an application utilizing a storage technology that has already been certified or that is more than half way through the NRC review process. NRC review times may be longer for a site-specific license referencing a new storage technology or if intervention is expected.

Construction of the ISFSI can begin once NRC has issued a Finding of No Significant Impact (FONSI) following review of the Environmental Report. The FONSI and NRC issuance of an EA are expected approximately 18 months into the NRC's site-specific review process. Construction of the ISFSI would be done at the utility's risk, i.e., it is possible that changes to the ISFSI design could result from NRC's safety review of the site-specific SAR that would require the need to make changes to the ISFSI site.

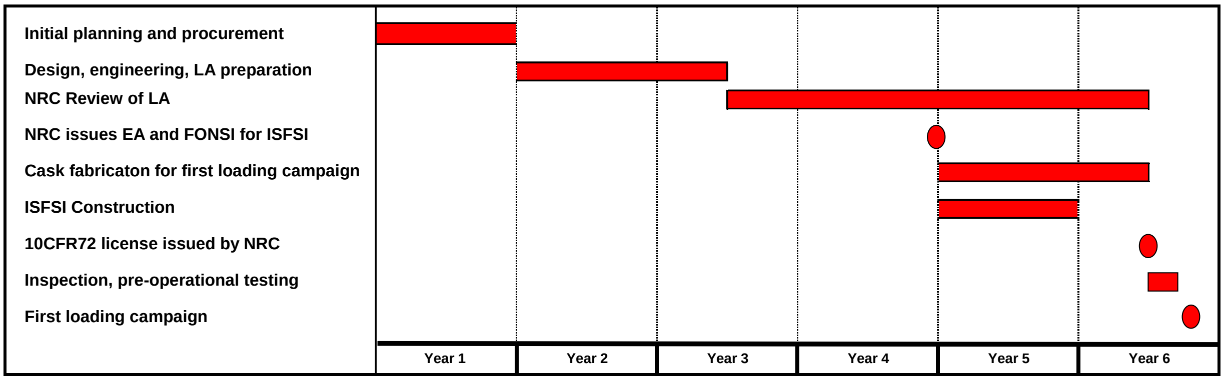

Figure 5-1
Hypothetical Schedule for 10CFR72 Site-Specific License

In this schedule, EPRI assumes that fabrication of the storage technology occurs in parallel with the final 12 to 18 months of NRC review. During the fabrication process, licensees monitor the fabrication of the cask systems and auxiliary equipment. This would include performing periodic audits and surveillance during the fabrication process, and evaluation of any design changes with the cask vendor.

NRC generally performs inspection activities prior to and during construction of the ISFSI storage pad, as well as during pre-operational testing. During the inspection and preoperational testing phase, the licensee would conduct ISFSI start-up testing and conduct functional tests of cask systems and components, and ancillary equipment. During the cask loading dry run, the readiness of people, equipment and procedures are demonstrated.

5.2.1.2 General License

Figure 5-2 presents a hypothetical schedule for planning dry storage using a NRC certified storage technology under a general license. This schedule assumes a total planning horizon of approximately 3.5 years, and assumes that: the general licensee plans to store SNF in a certified dry storage technology; no amendments to the 10CFR72 CoC are needed for the general licensee to utilize the technology; and that no amendments to the plant's 10CFR50 operating license are required.

Initial planning and procurement to implement dry storage under a general license is expected to take approximately 12 months. Initial planning and procurement activities include: performance of studies to determine viable storage expansion alternatives, evaluation of site-specific limitations, preparation of a request for proposal for dry storage systems, review and evaluation of proposals and selection of a storage technology.

Design and engineering for the ISFSI are assumed to take approximately 12 months and can be carried out in parallel with the licensee's 10CFR72.212 evaluation. EPRI assumes that the general licensee's 10CFR72.212 evaluation and its evaluation of existing plant programs, plans and processes will take approximately 18 months. This would include evaluation and modification of QA programs, emergency plans, physical security plans, etc.; as well as development and modification of procedures to support cask loading operations.

The schedule presented in Figure 5-2 assumes that fabrication of storage systems will take 18 months. During the fabrication process, licensees monitor the fabrication of the cask systems and auxiliary equipment. This would include performing periodic audits and surveillance during the fabrication process, and evaluation of any design changes with the cask vendor.

Construction of the dry storage facility including site preparation and construction of the storage pad is assumed to take 12 months. NRC generally performs inspection activities prior to and during construction of the ISFSI storage pad, as well as during pre-operational testing. During the inspection and preoperational testing phase, the licensee would conduct ISFSI start-up testing and conduct function tests of cask systems and components, and ancillary equipment. During the cask loading dry run, the readiness of people, equipment and procedures are demonstrated.

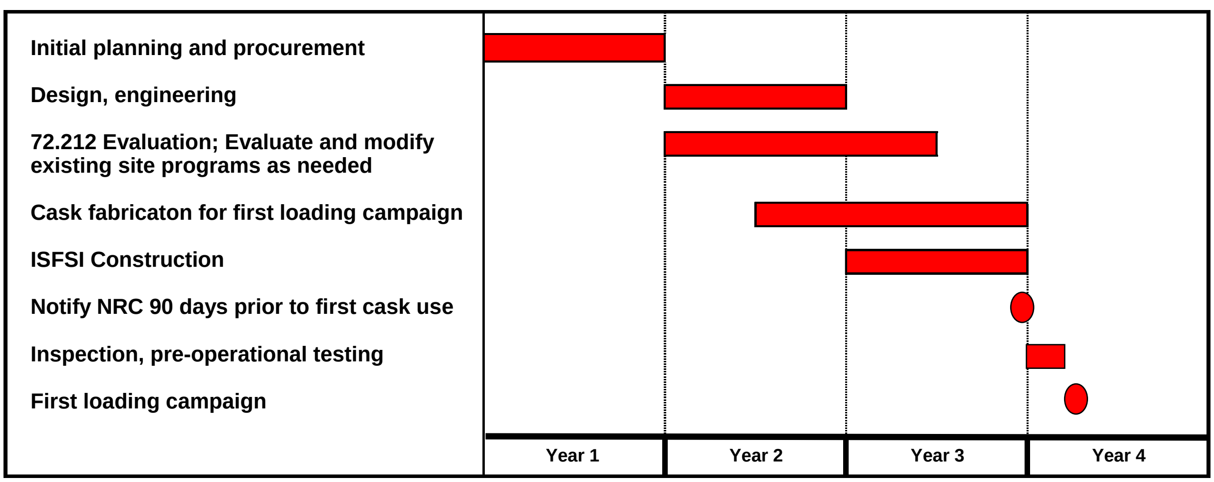

Figure 5-2z
Hypothetical Schedule for Dry Storage Under a General License

While not reflected in the above schedule, if the general licensee plans to utilize a dry storage technology that has not yet been certified, an additional 24 to 36 months should be added to the above schedule depending upon the complexity of the technology to be certified. If the reactor site's 10CFR50 operating license must be amended to address an unreviewed safety question or a change to the plant's technical specifications associated with cask loading operations, a licensee should consider the additional time needed to secure the license amendment and make the associated changes (such as upgrades to a crane). For prudent planning purposes, general licensees should begin dry storage planning at least six years in advance of the need for additional storage capacity to capture any unknown scheduling issues that might occur.

5.2.1.3 10CFR72 Certificate of Compliance

Figure 5-3 presents a hypothetical schedule for certification of a new dry storage technology in accordance with 10CFR72, Subpart L. This schedule assumes a planning horizon of approximately 3½ years, but notes that the NRC review time can be considerably longer if the technology has complex design features, uses new materials or analytical tools that have not been previously reviewed by NRC.

The period for design and development of a LA and SAR for a 10CFR72 CoC is assumed to be 12 months, although more complex designs may take longer. Following submittal of the LA to the NRC, NRC will issue an acceptance letter that includes a schedule for a first NRC RAI, vendor response to the RAI, and NRC issuance of a draft SER and draft CoC following completion of NRC's safety review. NRC's safety review is assumed to be 18 months, but this can be much longer if a second RAI is needed, particularly if the submittal includes the use of new materials or new methodology. Following issuance of a draft SER and draft CoC, the NRC publishes a notice in the Federal Register (FR) that it is amending its regulations to add the storage technology to the list of approved storage casks in 10CFR 72.214. The schedule assumes that the rulemaking period is approximately six months, after which the final rule is published and a final SER and CoC are issued by the NRC.

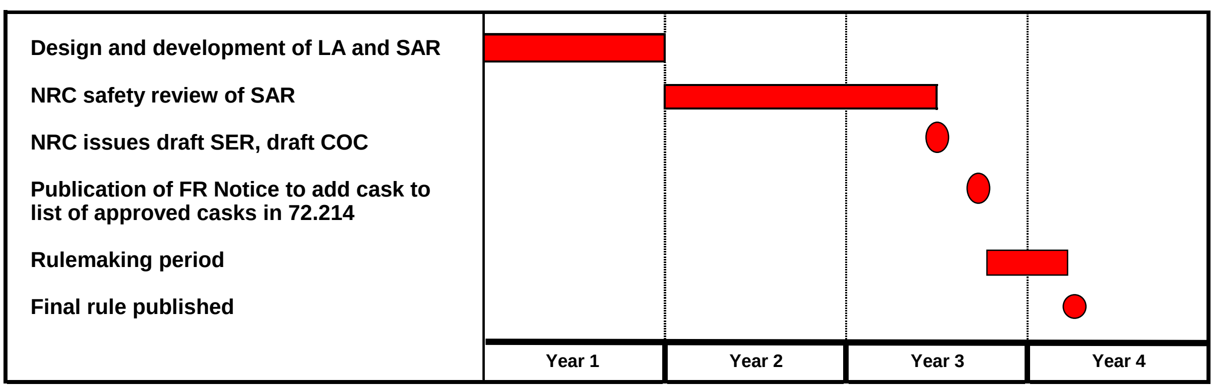

Figure 5-3
Hypothetical Schedule for 10CFR72 Certificate of Compliance Process

Amendments to an existing 10CFR72 CoC can take from 12 months to several years, depending upon the complexity of the amendment. For an amendment to an existing CoC, NRC can publish a "direct final rule" which shortens the rulemaking process to approximately two months from the publication of the FR notice until the rule becomes final. If there are significant public comments following the publication of the FR notice, the NRC withdraws the direct final rule and completes the longer rulemaking process to consider the public comments. NRC's response to any public comments would be included in the FR notice when the final rule is published.

5.2.2 In-Pool Storage Expansion

The time required to rerack a SNF storage pool, including the initial evaluation of storage alternatives will take approximately 2½ years. A hypothetical schedule is presented in Figure 5-4. Initial planning and procurement, design, engineering, and license amendment preparation is expected to take approximately 12 months.

A 12-month period is assumed for NRC's safety review of the license amendment. Typical license amendments for reracking projects, adding additional racks to unracked areas of the SNF pool or adding temporary racks to the cask loading pit, have taken 12 to 18 months for recent license amendments. Recent NRC safety reviews associated with license amendments that would allow the insertion of neutron absorbers in existing SNF storage racks also have taken 12 to 18 months. This assumes that the storage rack designs or in-pool storage expansion proposals do not use new materials or methodology that has not been previously approved by NRC. It also assumes that intervention by outside parties in the NRC proceeding does not occur. Intervention could add one year or longer to this schedule.

The hypothetical schedule assumes that the installation of new racks will take approximately 6 months. Reracking includes shuffling SNF in order to remove existing racks, removal of existing storage racks, and installation of the new storage racks. Depending on the percentage of the existing storage rack locations that are occupied by SNF, this process may need to go through several iterations. The greater the percentage of existing rack locations that are filled, the longer the rack removal and installation process will take. Rack installation for more complex reracking projects may take longer than six months.

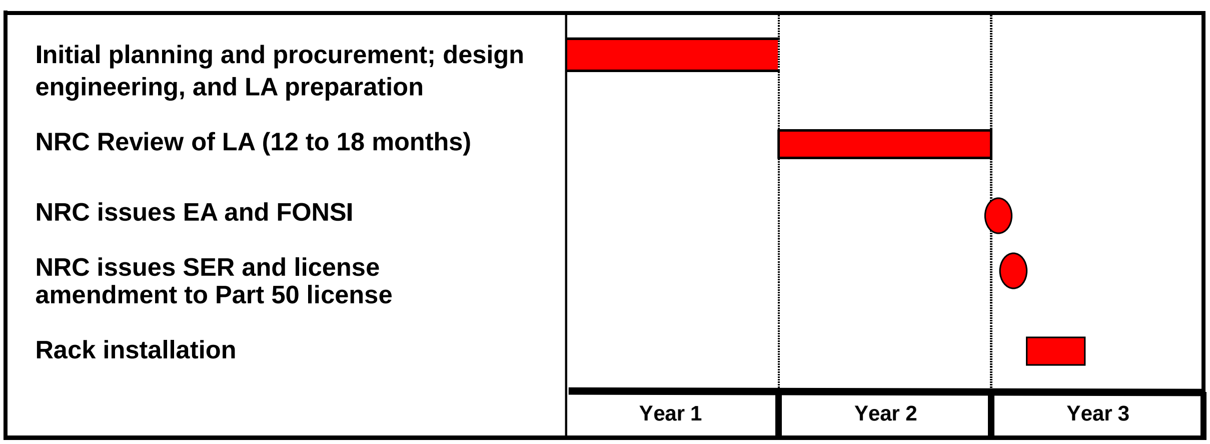

Figure 5-4
Hypothetical Planning Schedule for an In-Pool Storage Expansion Project

5.3 Dry Storage Staffing

On its web site, NRC has published a resource entitled, "*General License Considerations for Spent Fuel Storage in an Independent Spent Fuel Storage Installation at a Reactor Site*". This resource provides a detailed list of activities that must take place prior to a general licensee loading SNF into dry storage under a general license. In addition to the list of activities that must be performed, NRC provides an estimate of the staffing effort required to implement a dry storage project. NRC estimates that approximately 200 staff-months are needed to complete all of the required processes (such as 10CFR72.212 analysis, 10CFR72.48 analyses, 10CFR50.59 analyses, development of procedures, etc.). Figure 5-5 identifies 18 major tasks associated with implementing dry storage under a general license and estimates the staff-months required for each task. For example, NRC estimated that 18 staff-months are required to complete the evaluations required under 10CFR72.212, including evaluation of site characteristics against the cask design criteria contained in the cask's CoC, FSAR and SER; and 22 staff-months are associated with pre-operational start-up and testing activities.

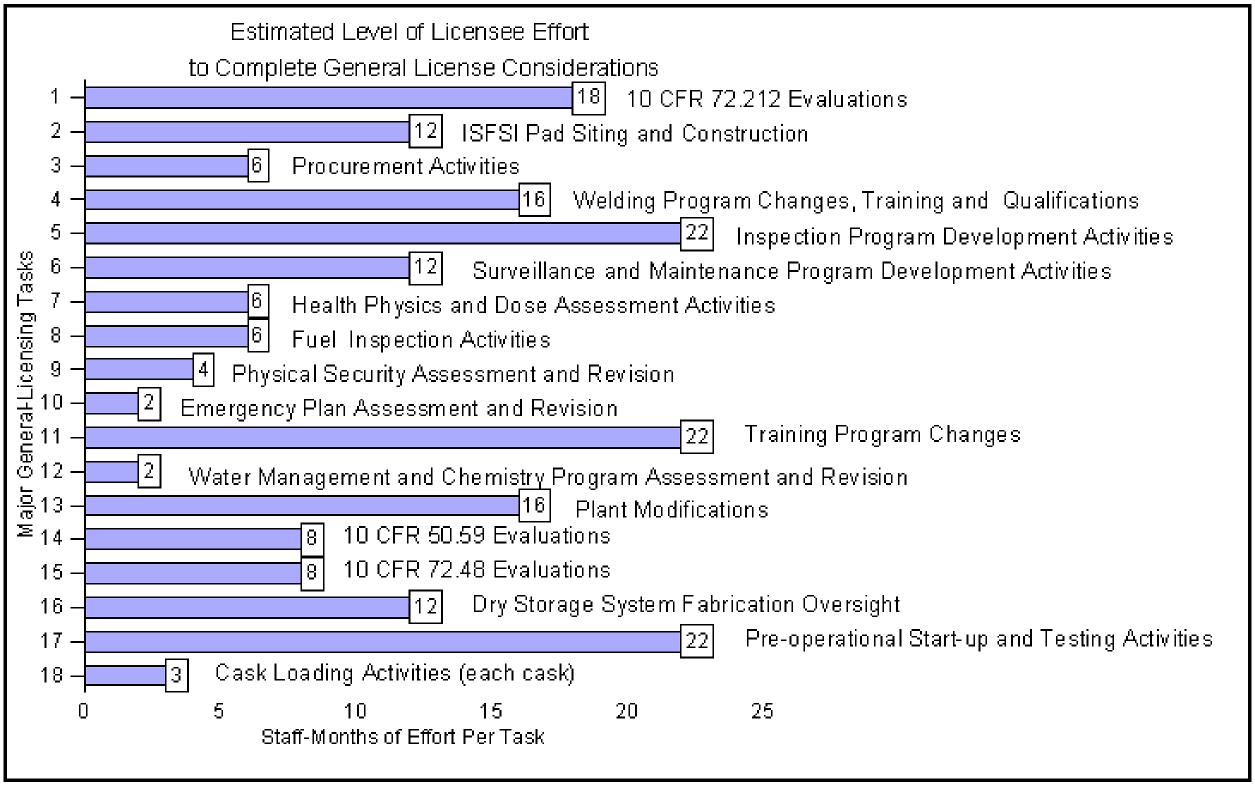

Figure 5-5
Estimated Level of Licensee Effort to Complete General License Considerations[8]

In addition to the project team that is formed to implement a dry storage project, many other individuals (either company employees or contract employees) will be involved at various stages of project implementation.

During the design, engineering, and licensing phase, the major disciplines that will be involved in the dry storage project are: fuel management, nuclear engineering, civil engineering, electrical engineering, radiation protection, regulatory compliance, construction and maintenance, and

[8] http://www.nrc.gov/waste/spent-fuel-storage/sf-storage-licensing/license-considerations.html

plant operations. Staff will be involved in preparation of licensing documents, safety analyses, and facility design.

During storage facility license application review or the 10CFR72.212 analysis required for general licensees, the majority of the staffing will be provided by fuel management, civil engineering, mechanical engineering, nuclear engineering, electrical engineering, radiation protection, regulatory compliance and plant operations. For site-specific licenses, staff will be required to respond to NRC requests for additional information and answer questions related to facility design and operation, as well as supporting NRC inspections. For general licensees, staff will be needed to support NRC inspections associated with implementation of the site's 10CFR72.212 evaluation, 10CFR72.48 evaluations, and 10CFR50.59 evaluations.

During facility construction and storage system fabrication, the following disciplines will be involved in the dry storage project: fuel management, civil engineering, mechanical engineering, nuclear engineering, electrical engineering, radiation protection, regulatory compliance, project control, quality control, purchasing, construction and maintenance, and plant operations. Staff will be involved with oversight of storage system component fabrication, NRC inspections associated with facility construction, facility site preparation and construction, and construction of storage components if applicable.

During preoperational testing and facility operation, the following disciplines will be involved in the dry storage project: fuel management, purchasing, construction, quality control, radiation protection, plant operations and maintenance (reactor engineering, fuel handling operators, welding crew, heavy load crew), chemistry, and system engineering. Staff will be involved with final review of cask loading procedures, delivery of dry storage systems, fuel selection and verification, modification support, cask handling and loading operations, cask transfer operations, NRC inspections, and ongoing scheduling, planning and construction support.

6 TECHNOLOGY EVALUATION CONSIDERATIONS

Due to the uncertainty associated with the schedule for implementation of a Federal SNF management system, all nuclear operating companies are planning to have sufficient on-site storage capacity (in-pool capacity, dry storage, or both) to be able to store all of the SNF discharged over the life of their plants. As discussed in more detail in Section 7.7, the nuclear industry, NRC, and DOE are beginning to consider the technical and regulatory issues associated with storage of SNF at reactor sites for an indefinite time period after nuclear power plants reach the end of their operating licenses and permanently cease operations.

As discussed in Section 5, in performing an evaluation of at-reactor SNF storage alternatives, the first step in the planning process will be to identify when additional SNF storage capacity would be required; project the amount of additional storage capacity required to store SNF discharged through the end of the facility's existing (and renewed) 10CFR50 operating license; and develop a milestone schedule to meet these requirements. SNF storage requirements are affected by future SNF discharges, the characteristics of the SNF discharged, the maximum capacity of reactor SNF pools, and SNF pool thermal management and criticality control requirements.

Next, storage expansion alternatives would be identified and evaluated to determine which storage expansion alternative(s) best meet an individual company's needs. The evaluation of storage expansion alternatives would include the identification of site-specific limitations that may impact the various storage options (e.g., crane capacity limits). This evaluation would also include identification of applicable NRC regulations associated with the alternatives being considered; State or Local regulations that pertain to the storage options; and possible intervention in any licensing actions. The evaluation of dry storage systems will include detailed technical review of the storage system proposed and a commercial review that considers economics and terms and conditions offered. This review should also consider the experience of dry storage vendors and the licensing status of the dry storage system offered. This section will examine the various considerations that are part of an evaluation of SNF storage expansion alternatives.

6.1 Economic Considerations

The economic analysis for storage expansion alternatives should consider both operating and post-shutdown SNF storage for the entire inventory of SNF to be discharged over the operating lifetime of the plant. At this time, it appears to be unlikely that DOE will begin acceptance of SNF prior to existing plants reaching the end of their renewed operating licenses. However, companies may want to estimate how long SNF will remain in dry storage at nuclear power plant sites in order to calculate post-shutdown SNF storage costs to include in decommissioning cost estimates so that sufficient funds can be collected to cover these costs.

6.1.1 In-Pool Storage Expansion Options

As noted in Section 2.1, in-pool storage expansion options may include conventional SNF pool reracking, adding additional racks to unracked areas of the SNF storage pool, licensing soluble boron credit in SNF storage pool, licensing temporary SNF storage racks for areas such as cask loading pits, and replacement of racks or installation of neutron poison inserts to address the neutron absorber degradation in existing racks.

Typical costs components for the various in-pool storage expansion alternatives include design and engineering; licensing costs associated with a license amendment and changes to a plant's technical specifications; rack manufacture, rack installation, and disposal of the old storage racks; manufacture and installation of neutron poison inserts; and movement of SNF within the pool. One factor that will affect rack installation costs is the degree to which current SNF storage racks have been filled. The nearer a SNF pool is to being full, the more SNF handling will be required during rack installation, increasing installation costs.

If a company is evaluating an in-pool SNF storage expansion project that will not provide life-of-plant storage against other alternatives, such as dry storage, it is recommended that the full life-cycle costs associated with SNF storage be considered. For example, if a reracking project will increase storage capacity in the SNF pool for an additional ten years of SNF discharges, but dry storage would be needed after that time, the net present value (NPV) life cycle costs for the reracking costs plus the subsequent dry storage costs should be compared to an alternative that considers near-term dry storage and no reracking project. The life-cycle costs would be considered along with other considerations for the storage expansion alternatives.

6.1.2 Dry Storage

Utility at-reactor dry storage costs are generally classified as upfront costs, incremental costs, decommissioning costs, annual operating costs during reactor operation and following shutdown for decommissioning.

6.1.2.1 Upfront Costs

Upfront costs include engineering, design, and licensing costs; equipment costs; construction costs; and start up and testing costs. Engineering and design costs include the costs for ISFSI siting and evaluation of site against cask design criteria; and design of the storage pad and associated components needed for loading and storage.

Licensing costs for an ISFSI storing SNF under a general license would include the utility internal costs for performing the 10CFR72.212 review of the dry storage technology CoC, technical specifications, FSAR, and NRC SER; performing 10CFR72.48 and 10CFR50.59 analyses to show that the casks can be used at the reactor site and that there are no unreviewed safety questions or changes to plant technical specifications; and updating the site quality assurance plan, emergency plan, training procedures, and security plan. Licensing costs for a site-specific license application would include preparation of licensing documents and NRC fees to review the license application. Fees associated with NRC inspections would also be included as licensing costs.

ISFSI construction costs would be dependent upon the storage site terrain, proximity to local populations and the need for additional shielding, and the size of the facility. Construction costs would include the cost of site preparation, road improvements if necessary, the concrete storage pad(s), electrical system, lighting, and security system. If metal storage casks are used, the casks would likely require monitors to ensure cask seal integrity.

The cask loading equipment required for an ISFSI can either be purchased directly from the storage vendor or, in some cases, leased from the storage vendor. Equipment that may be needed for an ISFSI would include welding equipment for canister-based systems; vacuum drying equipment; helium leak detection equipment; additional cask loading ancillary equipment such as slings and rigging for lifting the transfer cask; tractor equipment for transporting loaded storage casks to the storage pad; transfer casks for moving sealed metal canisters to their concrete storage casks or modules; and any other associated transfer equipment. Storage vendors should provide an estimate of equipment costs.

NRC requires that nuclear operating companies prepare site-specific procedures for ISFSI operations, perform a dry run of cask loading, and transport of a loaded cask to the ISFSI prior to initial operation. In addition, companies will be involved in oversight, audit and surveillance during equipment and cask manufacturing.

6.1.2.2 Incremental Costs

Incremental costs are the costs associated with the purchase and loading of storage systems on a periodic basis. These costs would include capital costs for the storage system including metal casks, concrete overpacks, and DPCs; labor costs associated with loading and unloading a storage system; and consumables associated with storage system loading and unloading such as helium used to fill the cask cavity, welding materials, etc.

Storage vendor bids generally provide costs on a per cask basis may include costs for cask loading operations if requested by the nuclear operating company. Dry storage system costs are typically provided as a base price, with escalation provisions that include some combination of materials and labor indices. Storage vendor bids may also include costs for unloading storage systems if requested as part of the scope of work.

6.1.2.3 Decommissioning Costs

Decommissioning costs for a dry storage facility will depend upon the storage technology selected and whether or not the storage system must be decontaminated. It is unlikely that the concrete storage pads will become contaminated. However, the concrete storage modules or casks may become contaminated or have neutron-induced activation and require disposal as low-level radioactive waste. Since no utility has yet decommissioned a dry storage facility and it is possible that some contamination or activation of the steel-reinforced concrete could take place, there may be a cost associated with low-level radioactive waste disposal for the concrete cask components. Metal storage-only casks and steel storage-only canisters would require decontamination. Since these systems would come into direct contact with SNF, it is possible that the metal components would become activated and may need to be disposed of as low-level radioactive waste.

Decommissioning costs estimates for ISFSIs are generally included as a subset of the decommissioning cost estimate for the power reactors. Also included in these estimates are the any costs associated with transferring SNF from the storage pool to dry storage after the plant permanently ceases operation, annual operating costs for the ISFSI, costs to unload and transfer SNF to transport casks for shipment offsite, and dismantlement costs for the ISFSI.

6.1.2.4 Annual Operating Costs

Annual operating costs for an ISFSI during reactor operation include the costs associated with NRC inspections, security, radiation monitoring, personnel costs for SNF management and fabrication surveillance activities, electric power usage for lighting and security systems, road maintenance to the ISFSI site, and miscellaneous expenses associated with ISFSI maintenance. NRC license fees for dry storage are included as part of the 10CFR50 operating license fee.

Annual operating costs for an ISFSI at a shutdown nuclear power plant include security, license fees (if the dry storage facility has a site-specific license and has not retained its 10CFR50 license), taxes, insurance, personnel costs, monitoring costs, electric power usage, and miscellaneous expenses associated with ISFSI maintenance.

6.2 Technical Considerations

Technical considerations that should be included in the evaluation of SNF storage alternatives include: site-specific physical limitations associated with storage alternatives; compatibility of alternative with existing and projected fuel inventory; licensing status of alternatives; vendor experience including customer base, vendor quality, vendor relationship with regulator, fabricators utilized by vendor, etc.; ALARA considerations; and timing for implementation of storage alternative.

6.2.1 Site-Specific Limitations

As part of the evaluation of SNF storage expansion options, site-specific limitations should be determined for all alternatives under consideration. For example, a site-specific limitation associated with reracking would be the seismic or structural constraints of a SNF pool.

Site-specific limitations associated with dry storage include:

- Siting characteristics such as potential for flooding and seismological characteristics must be assessed.
- Availability of near-site and at-reactor facilities (rail or barge) to facilitate the delivery of large, heavy components that are fabricated off-site. The availability of such facilities may affect the delivery of metal cask systems and concrete overpacks fabricated off site, as well as the eventual transport of SNF offsite.
- The capacity of the main and auxiliary reactor building overhead cranes that service the SNF pool must be adequate to support dry storage loading operations or be able to be upgraded.
- Cask handling activities must comply with a site's design basis for handling heavy loads.
- The effect of radiation from the storage facility on worker dose estimates and off-site dose estimates must be assessed.
- The floor loading capability of the reactor building cask access area and support capability of the decontamination or other cask lay down area must be able to support the system selected.

6.2.2 Licensing Status

The licensing status of a storage alternative will be an important factor in the evaluation. Recent experience has shown that amendments to 10CFR72 CoCs that include methodologies that have not previously been approved by the NRC (such as those for high heat load packages) can take 24 months or longer for approval. Concerning reracking, an important consideration is whether the vendor has successfully designed, licensed, and fabricated similar technology previously. This would include the successful use of any methodology in previous licensing analyses approved by NRC and previous approval of burnable absorbers or other design features. Storage rack designs with features that have not been previously approved by the NRC may require a more lengthy NRC review of the license amendment.

The evaluation of licensing status for dry storage systems will depend upon whether the system will be used under a site-specific or a general license. Storage system licensing schedules should be compatible with utility plans for loading SNF. If the storage system selected has not yet been certified by the NRC, the schedule for certification must be factored into the nuclear operating company's schedule for implementation of dry storage. As discussed in Section 5.2.1.3, a minimum 3 ½ year schedule should be assumed for certification of a new dry storage technology. If a license amendment is needed to an existing 10CFR72 CoC, the schedule to obtain the license amendment must be considered. As discussed in Section 5, the timing associated with an amendment to a CoC can range from 12 months to several years depending upon the degree of complexity of the amendment.

A utility planning to store SNF under a site-specific license can reference 10CFR72 CoCs or previously-approved TSARs in its site-specific license. A utility may also include the storage system design and safety analysis directly in its site-specific SAR.

6.2.3 Vendor Experience

Along with the detailed technical review of the storage system proposed and review of the proposed economics and terms and conditions, the experience of the vendor should also considered. The experience and proven technical capabilities of a vendor to design, license and deliver quality SNF storage systems are important factors in evaluating storage system alternatives – whether existing technology or new technology. Evaluation criteria to assess a particular vendor's experience might include:

- Successful licensing of a storage/transport system – has the vendor previously received NRC approval for a similar storage system? Have key vendor personnel been involved in the successful licensing of a storage system?
- Vendor storage/transport application successfully passes NRC's acceptance review process. Vendor receives limited RAIs associated with the application. This is an indication of vendor's capability and attention to detail.
- Vendor has performed testing and analyses to support new features and methodologies in advance of submitting the license application to NRC. Supporting data and analysis to demonstrate new methodologies is increasingly important.
- Demonstrated use of storage system – are any of the vendor's storage systems presently in use (either the same design or a different design) or are there other reactors that plan to use the same design?
- If the vendor's storage system is in use at reactor sites, have any concerns been identified? If so, how have these concerns been resolved?
- Relationship between the vendor and NRC – both positive interactions and any NRC concerns should be considered as well as the disposition of NRC concerns.
- What has been the outcome and disposition of any findings of vendor QA audits by NRC? Have there been ongoing quality problems?
- Relationship between the vendor and its fabricators; and interface between design and fabrication. Have any concerns been identified concerning fabricator interface? If so, how have they been resolved?
- Vendor resources available to resolve potential problems – does the vendor have an adequate resource base (e.g., management, engineering, operational, financial) to address and resolve, in a timely manner, any design, quality, or other problems that might arise? Has there been significant turnover over in key vendor personnel?

6.2.4 Shutdown Reactor Issues

Shutdown reactor sites planning a dry storage facility must address a number of issues that are not of immediate concern to an operating reactor site. These issues include:

- Storage of GTCC waste may need to be addressed if the licensee plans near-term dismantlement and decommissioning of the reactor and associated reactor internals. NRC has amended the 10CFR72 regulations to allow storage of GTCC waste at ISFSIs. Reactor-related GTCC waste may not be stored in a cask that also contains SNF. ISG-17, Interim Storage of Greater-than-Class C Waste provides NRC guidance regarding the storage of GTCC waste.

- The licensing method to be used for the shutdown reactor ISFSI: a 10CFR72, Subpart K general license that requires the Part 50 license remain active or a 10CFR72 site-specific license.
- A method for transfer of loaded DPCs from dry storage to transport casks for transfer off-site must be considered once the SNF storage pool as been dismantled and decommissioned.

6.3 Institutional Considerations

Some very important factors in the choice of SNF storage technologies are the institutional interfaces. A nuclear operating company should determine how the following types of issues would apply to the storage options under consideration:

- Rate treatment of the storage options, if applicable.
- State environmental regulations and the need to obtain State approvals for the storage expansion alternative;
- Local building or zoning regulations; and
- Activity of local stakeholder and interest groups.

A nuclear power plant site that is located in an area with very active intervention by local stakeholder or interest groups may find that dry storage under a general license provides a more certain schedule than an alternative that necessitates licensing action and the associated opportunity for public hearings. Some states require state approval as well as NRC approval of certain nuclear operating company actions. Section 9 discusses institutional considerations associated with expanding onsite storage in more detail.

7
DRY STORAGE TECHNICAL ISSUES

This section identifies technical issues associated with dry storage of SNF, of which licensees should be aware prior to implementing a dry storage project, including:

- Licensing related issues, such as renewal of site-specific ISFSI licenses and storage CoCs, implementation of a new CoC amendment, and implementation of NRC regulations contained in 10CFR72.48 and 10CFR72.212;
- NRC inspection procedures for dry storage projects;
- Cask and canister loading issues, such as canister loading and transfer operations, water removal during canister/cask loading, increased cask thermal load issues, and loading SNF that may be susceptible to top nozzle stress corrosion cracking;
- The process for characterizing damaged fuel;
- Industry activities associated with very long term storage of SNF; and
- Issues associated with quality assurance, design control and fabrication surveillance activities.

7.1 Licensing-Related Issues

7.1.1 Renewal of Site-Specific ISFSI Licenses and Storage CoCs

In April 2002, Dominion submitted an application to the NRC for renewal of its site-specific license for the Surry ISFSI. In the renewal application, Dominion requested an exemption from the 20-year license renewal term specified in 10CFR72.42(a) and sought approval for a 40-year license renewal term. In February 2004, Progress Energy submitted an application to the NRC to renew the site-specific license for the H.B. Robinson ISFSI. Progress Energy also requested an exemption that would allow the ISFSI license to be extended for an additional 40 years.

The NRC staff determined that approval of a 40-year renewal exemption request was a policy decision, not a technical one. As such, in September 2004, NRC staff submitted a paper to the Commission to request approval of the Dominion exemption request. [NRC 2004a] In November 2004, the Commission authorized the NRC staff to approve 40-year license renewal terms for the Surry ISFSI, with appropriate license conditions to manage the effects of aging. In addition, the Commission directed the NRC staff to: (1) Initiate a program to review the technical basis for future rulemaking; (2) provide recommendations on the license term for 10CFR72 CoCs for SNF storage; and (3) apply the Commission-approved guidance to future applications for site-specific exemption requests for 10CFR72 renewals without further

Commission approval. In response to this direction, the staff submitted a paper to the Commission that provided recommendations regarding rulemaking to address the dry storage license renewal issues identified previously. [NRC 2006a] Subsequently, the Commission directed NRC staff to proceed with rulemaking on this topic. [NRC 2006b] On September 14, 2009, NRC published a proposed rulemaking that would clarify the term limits for dry storage cask CoCs and ISFSI site-specific licensees. [NRC 2009b]

For site-specific ISFSI licenses, NRC has proposed to codify the approach taken when it granted a license renewal period of 40-years for the Surry and H. B. Robinson site-specific ISFSIs. Under the proposed regulations, all site-specific ISFSI licensees will have the flexibility to request up to 40-year initial and renewal terms while ensuring safe and secure storage of SNF.

For 10CFR72 CoCs, the proposed rules would allow the flexibility for applicants to request initial and renewal terms up to 40 years. Under this proposed change, applicants would be required to demonstrate that design and support/operational programs are suitable for the requested term.

For both site-specific licenses and 10CFR72 CoCs, the proposed rule adds a requirement that renewal applicants must provide time-limited aging analyses (TLAAs) and a description of an aging management program to ensure that storage casks will perform as designed under extended license terms. A time limited aging analysis would have the following attributes:

- Involve structures, systems and components (SSC) within the scope of license or CoC renewal;
- Consider the effects of aging;
- Involve time-limited assumptions defined by the current operating term; for example, 40 years;
- Were determined to be relevant by the licensee or CoC holder in making a safety determination;
- Involve conclusions or provide the basis for conclusions related to the capability of the SSCs to perform their intended functions; and
- Are contained or incorporated by reference in the design basis.

An aging management program would be designed to address aging effects and it may include prevention, mitigation, condition monitoring, and performance monitoring.

The NRC staff published a draft standard review plan for renewal applications for comment in September 2009. [NRC 2009d] This standard review plan will provide NRC staff with guidance on the renewal of site-specific licenses and CoCs.

To renew a site-specific license, an applicant must submit a license renewal application to the NRC at least two years prior to the expiration of the license, in accordance with the requirements of 10CFR72. To renew a 10CFR72 CoC, an applicant must submit an application for renewal of the CoC not less than 30 days before the expiration date of the CoC. According to NRC's proposed rule, if "*the applicant has submitted a timely application for renewal, the existing CoC will not expire until the application for renewal has been determined by the NRC.*" [NRC 2009b] Both site-specific license and CoC renewal applications must contain revised technical requirements and operating conditions (fuel storage, surveillance and maintenance, and other requirements) for the ISFSI or dry storage system that address aging effects that could affect the safe storage of the SNF and must specify what the licensee of an ISFSI, or the holder of a CoC, is authorized to store. Figure 7-1 presents a diagram of the license renewal process under 10CFR72.

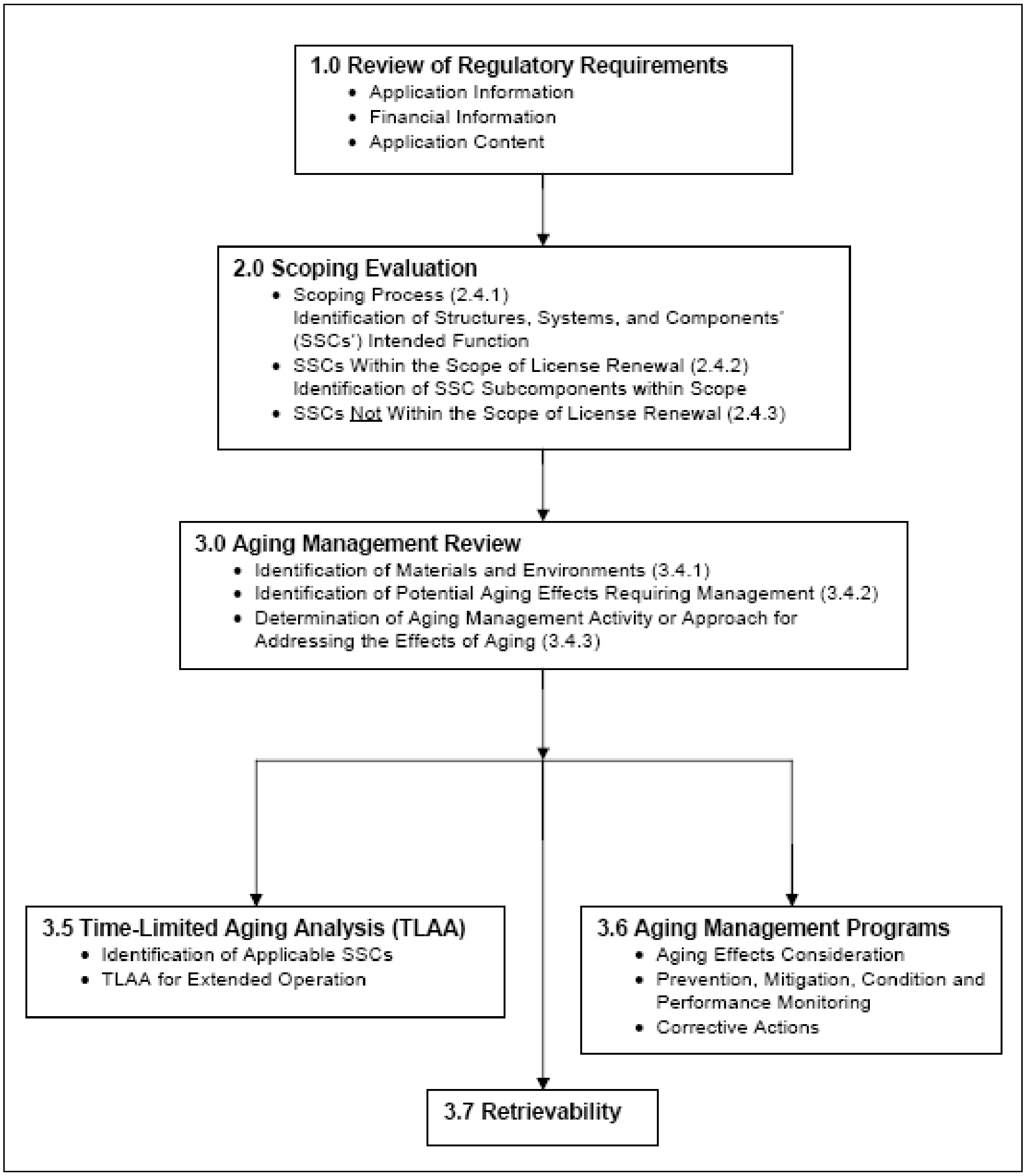

Figure 7-1
License Renewal Process [NRC 2009d]

7.1.2 Implementation of 10CFR72.48 Provisions

NRC regulations contained in 10CFR72.48, "*Changes, tests and experiments*," provide requirements for the process by which site-specific and general licensees and 10CFR72 CoC holders may make changes to their ISFSIs, SNF storage cask designs, and procedures, and conduct tests or experiments not described in the FSAR without prior NRC approval. The 10CFR72.48 process may be used if no changes are needed to the technical specifications associated with the site-specific license; no changes are needed to the terms, conditions or specifications of a CoC; and the changes do not meet the criteria in 10CFR72.48(c)(2), discussed in more detail below. General licensees would also use the 10CFR72.48 evaluation process to evaluate any changes to the written evaluations that are required by 10CFR72.212.

The provisions in 10CFR72.48 work much in the same way as 10CFR50.59 works for a Part 50 license. The licensee is required to maintain records of changes made to the ISFSI, cask or procedures, including a written safety evaluation that provides the bases for the determination that the change, test or experiment does not involve an unreviewed safety question.

In June 1997, NRC issued Information Notice 97-39, *Inadequate 10CFR72.48 Safety Evaluations of Independent Spent Fuel Storage Installations*. [NRC 1997a] This notice was issued to alert addressees to inadequate safety evaluations that have been performed under 10CFR72.48. NRC has developed inspection procedures associated with the review of licensee and CoC holder 10CFR72.48 evaluations. The Inspection Procedure (IP) 60857, *Review of 10 CFR 72.48 Evaluations* [NRC 2007c], provides guidance to inspectors in assessing the effectiveness of licensee or CoC holder performance of 10CFR72.48 evaluations, and in ensuring that any required license or CoC amendments have been obtained. The IP also provides guidance for review of the licensee or CoC holder's procedures and training programs associated with implementation of 10CFR72.48.

NRC has issued Regulatory Guide 3.72 (RG 3.72), *Guidance for Implementation of 10 CFR 72.48, Changes, Tests, and Experiments,* March 2001. [NRC 2001a] NRC's RG 3.72 notes that the Nuclear Energy Institute (NEI) submitted to the NRC "*Guidelines for 10 CFR 72.48 Implementation*," Appendix B to a prior NRC guidance document regarding 10CFR50.59 evaluations (NEI 96-07). [NEI 1996] The 10CFR72.48 evaluation process, shown in Figure 7-1, includes the following steps:

- **Applicability**: Determine if a 10CFR72.48 evaluation is required.
- **Evaluation**: Apply the evaluation criteria of 10CFR72.48(c)(2) to determine if a license amendment or CoC amendment must be obtained from NRC.
- **Documentation and reporting**: Document and report to NRC and to licensees and certificate holders, activities implemented under 72.48.

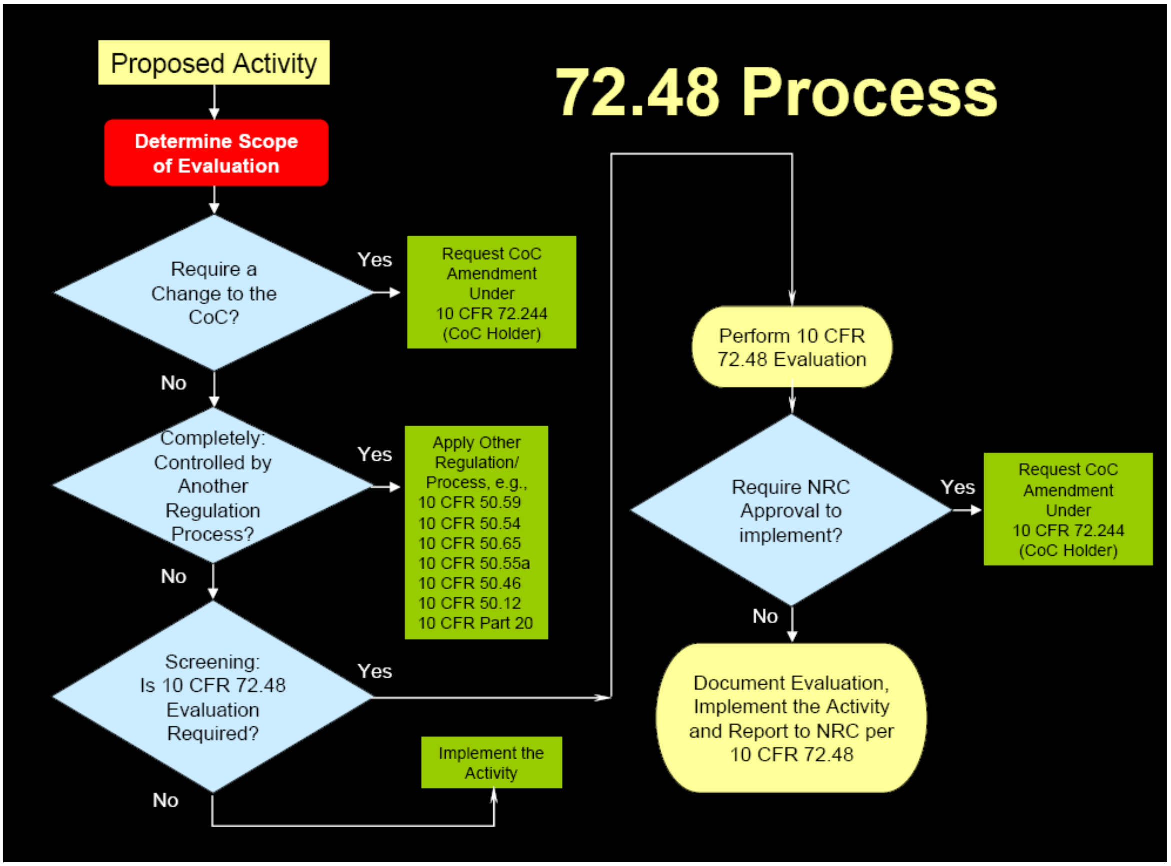

Figure 7-2
10CFR72.48 Process for Site-Specific Licensee, General Licensee and CoC Holder [NEI 1996, Sides 2006]

During the evaluation step, the evaluation criteria from 10CFR72.48(c)(2) are to be applied to determine if a license amendment or a CoC amendment is needed. These evaluation criteria require that site-specific licensees obtain a license amendment or general licensees and CoC holders obtain an amended CoC prior to implementing a proposed change, test, or experiment if the change, test, or experiment would:

- Result in more than a minimal increase in the frequency of occurrence of an accident previously evaluated in the FSAR;
- Result in more than a minimal increase in the likelihood of occurrence of a malfunction of a SSC important to safety previously evaluated in the FSAR;
- Result in more than a minimal increase in the consequences of an accident previously evaluated in the FSAR;
- Result in more than a minimal increase in the consequences of a malfunction of an SSC important to safety previously evaluated in the FSAR;
- Create a possibility for an accident of a different type than any previously evaluated in the FSAR;

- Create a possibility for a malfunction of an SSC important to safety with a different result than any previously evaluated in the FSAR;
- Result in a design basis limit for a fission product barrier as described in the FSAR being exceeded or altered; or
- Result in a departure from a method of evaluation described in the FSAR used in establishing the design bases or in the safety analyses.

While both NRC and industry have developed guidance documents associated with implementation of the 10CFR72.48 provisions, licensees and CoC holders continue to periodically apply these provisions when a license amendment or CoC amendment is necessary. When a licensee or CoC holder is unsure regarding whether it must seek an amendment from NRC, the best course of action may be to discuss the proposed changes, tests, or experiments with NRC staff well in advance of implementing the proposed change such that a license amendment can be obtained if it is necessary. This will avoid potential inspection violations and the need for possible exemption requests while a license or CoC amendment is sought. NRC and industry are examining whether changes are needed to the regulations for 10CFR72.48, the guidance associated with it, or some combination in order to address issues associated with implementation of the 10CFR72.48 process.

During the certification and amendment of 10CFR72 CoCs, both the CoC holder and NRC reviewers should take considerable care to ensure that parameters that are not required to be in the CoC and technical specifications are not inadvertently included in these documents, since any changes to either the CoC or technical specifications will require a CoC amendment.

7.1.3 Implementation of New CoC Amendment

On September 15, 2009, the NRC published a proposed rulemaking in the FR entitled, “License and Certificate of Compliance”. [NRC 2009b] The new rule, which is expected to be finalized in 2010, proposes to modify 10CFR72.212 to allow a general licensee to apply changes authorized by an amendment to a CoC to a previously loaded cask. The licensee must demonstrate through a written evaluation that the loaded cask meets the terms and conditions of the CoC amendment that is being applied to it. The regulations will also be revised to require general licensees to submit a cask registration letter no later than 30 days after applying the changes authorized by an amended CoC to a previously loaded cask. The registration letter must include the cask certificate number, the amendment number to which the cask will conform, the cask model number, and the cask identification number.

As noted in the September 15, 2009, FR notice, under current regulations, general licensees cannot apply new CoC amendments to previously loaded casks. A license cannot utilize 10CFR72.48 provisions to apply CoC amendment changes to previously loaded casks. This is due to the fact that 10CFR72.48 does not allow licensees to make changes that result in a change in the terms, conditions, or specifications incorporated in the CoC. A previously loaded cask is bound by the terms, conditions, and technical specifications of the CoC applicable to that cask at the time the licensee loaded the cask. The proposed rule would not amend 10CFR72.48; but would amend 10CFR72.212 requiring licensees to evaluate that the previously loaded cask meets the terms and conditions of the CoC amendment that is being applied to it.

7.1.4 Implementation of 10CFR72.212 Analysis

The general license provisions of 10CFR72 require that general licensees perform written evaluations to ensure that the conditions established by the storage cask CoC have been met; that concrete storage pads have been designed to support the static and dynamic loads of the stored casks and that the requirements of 10CFR72.104, regarding onsite and offsite radiation exposure have been met. Licensees will prepare a 10CFR72.212 report that outlines their compliance with the regulations. Updated 10CFR72.212 reports must be completed for each loading campaign to ensure that the licensing basis of each loaded cask is properly documented.

As an example, in its first SNF dry storage loading campaign, General Licensee A loaded a dry storage cask with CoC, Amendment 0 and FSAR Amendment 1. The written evaluations performed under 10CFR72.212 would have been based on these documents, the related technical specifications and SER for CoC Amendment 0, along with any changes allowed under 10CFR72.48. In its second loading campaign, General Licensee A loaded the same dry storage cask design, but under CoC, Amendment 1 (and related technical specifications and SER) and FSAR Amendment 2. The 72.212 evaluations for the casks loaded in the second campaign must be based on these later amendment documents.

Thus, as ISFSI operations continue over time, the applicable CoC amendment and/or FSAR revision may change, requiring that the 10CFR72.212 evaluation be revised, at a minimum, for each dry fuel storage campaign. This would include identifying the applicable CoC amendment, FSAR revision, and any changes performed in accordance with 10CFR72.48 associated with the casks loaded in that campaign. General licensees may include an appendix to the 10CFR72.212 evaluation report that lists the fuel campaign loading dates, plant name, cask and component model number and serial number, cask CoC amendment number, cask FSAR revision number, and a list of any approved interim changes not yet included in the cask FSAR, made through the 10CFR72.48 change process or under another process (e.g., editorial or administrative change, or program controlled under 10CFR50.54).

The requirements of 10CFR72.212(b) specify that the general licensee must evaluate the following:

- 10 CFR 72.212(b)(2)(i)(A) - Certificate of Compliance Conditions: the general licensee shall perform written evaluations, prior to use, that establish that the conditions set forth in the CoC have been met. The resulting evaluation would typically be referenced in the 10CFR72.212 Report.
- 10 CFR 72.212(b)(2)(i)(B) - Cask Storage Pad Design: the general licensee shall perform written evaluations, prior to use, that establish that "*cask storage pads and areas have been designed to adequately support the static and dynamic loads of the stored casks, considering potential amplification of earthquakes through soil-structure interaction, and soil liquefaction potential or other soil instability due to vibratory ground motion*." The general licensees' analyses would be referenced in the 10CFR72.212 report including the results of

geotechnical investigations, storage cask design analysis, cask tipover evaluation, cask sliding and overturning evaluation, and evaluation of soil liquefaction.

- 10 CFR 72.212(b)(2)(i)(C) - Dose Analyses Pursuant to 10 CFR 72.104: the general licensee shall perform written evaluations, prior to use, that establish that the requirements of 10CFR72.104 have been met. 10CFR72.104 provides criteria for radioactive materials in effluents and direct radiation from an ISFSI. The analyses will typically describe the controlled area boundary for the ISFSI; the doses due to normal operations and anticipated occurrences; source term assumptions, including a description of the source term and dose rate analysis; and any operational restrictions needed to meet ALARA objectives.
- 10 CFR 72.212(b)(3) - Review of the Cask FSAR and SER: general licensees are required to review the cask FSAR referenced in the CoC and the related NRC SER, prior to use of the general license, to determine whether or not the reactor site parameters, including analyses of earthquake intensity and tornado missiles, are enveloped by the cask design bases considered in these reports. The results of this review are required to be documented. This will include review of the cask environmental parameters to ensure that the site parameters are bounded, review of the heavy haul path; identification of any changes or deviations to the cask's FSAR; and CoC holder approval of site cask operating procedures.
- 10 CFR 72.212(b)(4) - Review of Part 50 Facility Impact (10 CFR 50.59): prior to use of the general license, general licensees are required to determine whether activities related to storage of SNF under this general license involve a change in the facility Technical Specifications or require a license amendment for the facility pursuant to 10CFR50.59(c)(2). The results of this determination must be documented in the evaluation made in accordance with 10CFR72.212(b)(2).
- 10 CFR 72.212(b)(5) - Security Plan: requires general licensees to ensure protection of the SNF against the design basis threat of radiological sabotage in accordance with the same provisions and requirements as are set forth in the licensee's physical security plan pursuant to 10CFR73.55. The licensee would review the site physical security plan and procedures and modify them, as necessary, to reflect SNF cask loading and transport operations on site, as well as storage operations at the ISFSI.
- 10 CFR 72.212(b)(6) – Programs: general licensees are required to review the reactor emergency plan, quality assurance program, training program, and radiation protection program to determine and evaluate an impacts associated with ISFSI operations. The programs are to be modified, as necessary, to include changes to reflect ISFSI operations. The details of those changes are typically maintained in the program plan documents and associated change packages (i.e., evaluations pursuant to 10CFR50.54)…

7.1.5 Control of General Licensee ISFSI Licensing Basis

The licensing basis for a site-specific ISFSI is contained in the ISFSI FSAR and technical specifications. A site-specific licensee has direct control over the ISFSI FSAR and technical specifications.

The licensing basis for a general licensee is more complicated in that the licensing basis includes the applicable CoC (and applicable revision) which describes the technical specifications, approved contents and dry storage technology design features, the applicable FSAR, and the licensee's 10CFR72.212 Report. As discussed in Section 7.1.4, as a general licensee loads additional storage systems, the licensing basis for each storage system must be maintained, and it is likely that the licensing basis for each cask or set of casks loaded will be unique. Therefore, a general licensee will need to maintain multiple CoC amendments, multiple FSAR revisions, and revise the 10CFR72.212 Report to track the licensing basis for each system loaded. This will include the identification of any cask-specific changes made using the 10CFR72.48 process that have not been captured in the cask FSAR. In addition, a general licensee only has direct control over its 10CFR72.212 Report. The CoC Holder controls revisions to the package FSAR and amendments to the CoC.

7.2 NRC Inspections

The requirements for NRC activities associated with inspection of ISFSIs are contained in NRC Inspection Manual Chapter 2690, "*Inspection Program for Dry Storage of Spent Reactor Fuel at and Independent Spent Fuel Storage Installation,*" [NRC 2001c] and in Appendix A to Manual Chapter 2690, "*Inspection Program for an ISFSI Located at a Reactor Site.*" [NRC 2001d] This inspection manual chapter (IMC) covers all activities related to dry storage of SNF in an at-reactor ISFSI, including: operations, maintenance, surveillance testing, preoperational testing, design control, fabrication, and construction. Appendix B to the IMC covers the requirements for an inspection program for an away-from-reactor ISFSI.

When a new at-reactor ISFSI is planned, the NRC Region in which the facility will be located is responsible for developing an Integrated Inspection Plan (IIP) in accordance with the Inspection Manual. Regarding the development of an IIP for a new ISFSI, NRC's goal is to issue an IIP at least 12 to 24 months before the licensee or applicant intends to begin storage of SNF in the ISFSI. Inspection activities will include those associated with the ISFSI: design, construction, fabrication, preoperational testing, and operations; and those associated with existing plant programs that are modified to support ISFSI operations: quality assurance, security, emergency preparedness, or radiation protection. The IPP should indicate the planned licensee milestones, the planned inspection dates, and any linkages between the two (e.g., the relationship between dates for inspecting the ISFSI support pad and the licensee's planned pad construction and concrete placement schedule). The guidance suggests that the NRC Region base its inspection plans on the cask technology FSAR, SER, previous inspections, vendor or fabricator prior performance, and lessons learned from previous IIPs. The SFST staff is expected to identify any technical, regulatory, or performance issues that should be included as specific elements in the IIP, such as inspections of vendors or fabricators.

Appendix A to IMC 2690 provides more detailed information regarding guidance for at-reactor ISFSI inspections including guidance on the scheduling and conduct of inspections during various phases of ISFSI activities: design, fabrication, and construction; preoperational testing; loading and unloading; and storage monitoring. Guidance is also provided on the frequency of performing periodic inspections once SNF has been placed in the ISFSI for storage. Appendix A separates inspection activities into four phases:

- Phase 1 - Design, fabrication, and construction
- Phase 2 - Preoperational testing, including dry runs
- Phase 3 - SNF loading and unloading operations
- Phase 4 - Storage monitoring of the loaded ISFSI

Table 7-1 summarizes the Appendix A milestones (and the “not later than” milestone) for completion of inspection activities associated with Phase 1, Phase 2 and Phase 3. The guidelines allow certain inspection procedures (IPs) to be omitted if the licensee has recently demonstrated satisfactory performance in a specific inspection area. An example of this would be if a licensee has recently completed some dry runs at one ISFSI site, and will be using the same crews and the same equipment to load SNF into an ISFSI at a second site. The Appendix A guidance also notes that selected IPs shown in Table 7-1 would be re-performed if a licensee plans to use a new model or type of dry storage system.

Table 7-1
Appendix A to IMC 2690, Inspection Procedures and Milestone Dates to Support Initial ISFSI Operation

IP Number	IP Subject	“Not Later Than” Milestone
60851	Design control of ISFSI components	Beginning of fabrication
60852	ISFSI component fabrication by outside fabricators	Completion of fabrication
60853	On-site fabrication of components and construction of and ISFSI	Completion of construction
60854	Pre-operational testing of an ISFSI	Completion of pre-operational testing
60855	Operation of an ISFSI (other than initial fuel loading, unloading, and surveillance)	Before loading begins (1)
60856	Review of 10CFR 72.212 (b) evaluations	Completion of pre-operational testing (2)
60857	Review of 10CFR 72.48 evaluations	As needed to support above IPs
(1) All loading and unloading procedures should be reviewed before initial loading of SNF into the ISFSI. (2) Review of the licensee's 10 CFR 72.212(b) evaluations of the ISFSI support pad should be completed before the licensee begins construction of the support pad. IP 60856 should be used to accomplish this review.		
NRC 2007c, NRC 2008a, NRC 2008b, NRC 2008c, NRC 2008d, NRC 2008e, NRC 2008f,		

As shown in Table 7-2, after the initial cask loading campaign, Phases 3 and 4 inspection activities focus on loading and unloading activities, modifications to the facility, 10CFR72.48 evaluations, 10CFR72.212 evaluations if a new storage technology design is used, and surveillance monitoring of ISFSIs.

Table 7-2
Inspection Activities and Frequency to Support Ongoing ISFSI Operation, Appendix A to IMC 2690

IP Number	Inspection Activity
60851	Modifications to the ISFSI
60855	Loading additional casks (each occurrence), performing surveillance, and unloading casks (each occurrence)
60856	First use of different dry storage cask design
60857	Modifications to the ISFSI or dry storage cask design

7.3 SNF Loading and Transfer Issues

The preparation for and loading and transfer of SNF from SNF storage pools to an at-reactor ISFSI is a complex process involving multiple areas of plant support including support staff from plant operations, maintenance, training, regulatory compliance, security, warehouse and spare parts, engineering, quality assurance, radiation protection, and nuclear engineering. The process for loading and transfer of SNF to a canister-based dry storage technology is described below. Also discussed below are issues that have been encountered by licensees during ISFSI operations that relate to SNF loading, handling and transfer activities. These are detailed below to aid licensees in future ISFSI operations.

7.3.1 Typical Canister Loading and Transfer Operations

While each canister-based or cask storage technology will have specific procedures associated with SNF loading operations, and cask handling and transfer activities, many of the steps in the process to transfer SNF from the storage pool to an at-reactor ISFSI are similar, as described below.

The process begins with the receipt of the storage canisters at the reactor site and inspection of the canisters in accordance with the canister inspection procedure. Canister loading and transfer operations occur in accordance with site-specific procedures that have been developed for loading, handling and storing SNF in the specific dry storage technology being utilized. Prior to the start of loading operations, the transfer cask is cleaned and/or decontaminated and inspected in accordance with procedures. [Holtec 2007, NAC 2004, Transnuclear 2007],

Handling empty DPC and transfer cask

- Lift the DPC by crane and lower it into the transfer cask cavity using the DPC's lifting lugs.
- Lift the transfer cask containing an empty MPC by crane and move it to the SNF storage pool.

- Fill the annulus between the DPC exterior surface and the transfer cask with demineralized water. If the transfer cask design calls for it, an inflatable cask annulus seal is connected to the transfer cask to prevent contamination of the exterior surface of the DPC.
- Fill the DPC with either pool water or demineralized water, in accordance with procedures.
- Attach the transfer cask lifting yoke to the transfer cask lifting trunnions.
- Lift the transfer cask and empty DPC and lower it into the SNF storage pool in the cask loading pit. As the transfer cask is lowered into the pool, the exterior surface of the cask is sprayed with demineralized water to minimize contamination of the exterior surface of the transfer cask.

DPC fuel loading operations

- Disengage the lifting yoke from the transfer cask. Remove the yoke from the fuel pool and spray the yoke and crane cables with clean demineralized water, if it is removed from the fuel pool.
- Load SNF assemblies that were previously selected in accordance with the storage system CoC into the DPC.
- Visually verify each SNF assembly serial number, in accordance with procedures, using an underwater camera prior to being loaded into a DPC. The assembly serial numbers are verified against the previously prepared cask loading plan (development of a cask loading plan is discussed in more detail in Section 7.5).
- Transfer SNF assemblies from the fuel storage racks into the DPC until the DPC is fully loaded in accordance with procedures.

Removal of loaded DPC and transfer cask from SNF storage pool

- Lift the DPC closure lid to the SNF pool and lower the lid until it is seated in the top of the DPC.
- Attach the lifting yoke to the transfer cask trunnions and raise the loaded transfer cask to the pool surface.
- Inspect the DPC closure lid to ensure that it is properly seated.
- Raise the cask from the pool while spraying exterior surfaces of the transfer cask and DPC lid that were exposed to pool water with demineralized water.
- Survey the DPC closure lid and perimeter of the transfer cask for surface contamination. Decontaminate the bottom of the transfer cask prior to setting the cask in the designated area for DPC closure operations.
- Move the loaded transfer cask to the designated area for DPC closure operations. Note that some storage system designs allow the DPC closure operations to be performed while the transfer cask is partially submerged in the SNF storage pool.
- For transfer cask designs that include a liquid neutron shield, the neutron shield is filled with water or other approved liquids in accordance with procedures.

MPC moisture removal and helium backfill

- In the designated area for DPC closure operations, install supplemental shielding to minimize personnel exposure.
- Disengage the transfer cask lifting yokes and remove them from the area.
- Connect a drain line to the transfer cask and remove water from the transfer cask annulus until the water line is six to twelve inches below the top of the DPC.
- Install the welding system to the DPC closure lid.
- Connect a water pump to the DPC drain port and remove water from the DPC to support welding operations. Procedures will specify the amount of water to be removed.
- Weld the DPC closure lid to the DPC in accordance with procedures.
- During DPC welding operations, monitor combustible gas levels in the interior of the DPC to ensure that they are within the levels specified in storage system FSAR.
- If required for a specific DPC design, purge the space below the DPC lid with inert gas prior to and during welding of the DPC closure lid to remove combustible gas that develops in this space during closure operations.
- Perform visual and dye penetrant examinations of the welds in accordance with procedures.
- Drain water from the DPC cavity.
- Remove remaining moisture from the DPC cavity using either a vacuum drying system or a forced helium/vacuum drying system in accordance with the procedures.
- Backfill the DPC cavity with helium to provide an inert atmosphere.
- Perform helium leak testing of the DPC closure lid in accordance with procedures.

DPC closure

- If the DPC design has an outer and inner closure lid, pull a vacuum between the inner and outer lids and perform a helium leak test of the inner closure lid. The outer closure lid is welded to the DPC and the welds are examined in accordance with procedures.
- Weld closure plates to the drain port and vent ports and examine the welds in accordance with procedures.
- Remove remaining water from the transfer cask and seal transfer cask drain ports.
- Install the transfer cask lid.

Transfer of DPC to concrete storage cask

- **Vertical transfer**
  - Position an empty vertical concrete cask in the designated DPC transfer location and inspect the interior and openings for debris.
  - Using a crane, install the device used to align the transfer cask to the top of the concrete storage cask.

- Attach the transfer cask lifting yoke to the crane hook and the transfer cask lifting trunnions.
- Lift the transfer cask and move it above the empty concrete storage cask and the alignment device.
- Lower the transfer cask until it engages the alignment device.
- Attach the DPC lifting system to the DPC. Lift the DPC slightly to remove the DPC weight from the bottom transfer doors of the transfer cask.
- Open the transfer cask doors and lower the DPC into the concrete storage overpack, until the DPC is seated in accordance with procedures.
- Disconnect the lifting slings and lower them through the transfer cask to the top of the DPC.
- Close the transfer cask bottom transfer doors and lift it from the top of the concrete cask. Remove the alignment device.
- Remove the slings and lift yoke from the DPC.
- Using a crane, install the concrete cask lid.
- Using appropriate transfer equipment, move the loaded concrete cask along the designated transfer route to its designated storage location

- **Horizontal transfer**
  - Attach the transfer cask lifting yoke to the crane hook and the transfer cask lifting trunnions.
  - Lift the transfer cask over the cask support skid onto the transfer trailer and lower the transfer cask and tip horizontally onto the support skid.
  - Transfer the loaded transfer cask from the fuel building to the ISFSI along the designated transfer route.
  - Using a portable crane, remove the shield door to the storage module and inspect the module cavity.
  - Using a portable crane, unbolt and remove the transfer cask top closure lid.
  - Position the trailer in front of the storage module and connect the skid positioning system. Align the cask and the storage module.
  - Using the skid positioning system, dock the transfer cask into the storage module access opening docking collar.
  - Position the hydraulic ram behind the cask in horizontal alignment with the cask and level the ram. Remove the bottom ram access cover plate from the transfer cask and extend the ram through the bottom cask opening into the DPC grapple ring.
  - Using the hydraulic ram, insert the DPC into the storage module.
  - Disengage the ram grapple mechanism and retract the ram from the cask.

- Replace the cask bottom cover plate and remove the transfer cask from the storage module opening.
- Using a portable crane, install the storage module door.

7.3.2 Water Removal During Canister/Cask Loading

To remove water and residual moisture from a DPC or metal storage cask following fuel loading activities, licensees utilize either vacuum drying systems or forced helium systems. Several dry storage technologies require the use of a forced helium system for residual moisture removal if high-burnup SNF will be loaded into the storage technology. The requirements for using this system are described in the storage technology CoCs and FSARs, including any operational limits imposed such as drying times.

A vacuum drying system connected to a DPC or cask is used to remove water from the system in a stepped evacuation process. The cavity pressure is typically reduced in steps to prevent the formation of ice in the DPC or cask and the vacuum drying system. After each stepdown in cavity internal pressure the cavity pressure is held for some period of time and monitored. The process continues until the cavity pressure in the DPC or cask stabilizes and can be held for the time required in the technical specifications for the storage system. Dryness is confirmed by ensuring that the cavity pressure rise is less than the acceptance criteria specified by the technical specifications. Time limits are set for the vacuum drying process to ensure that the fuel cladding and DPC or cask structural components remain below the allowable limits analyzed in the package FSAR.

In using a forced helium system, helium is circulated through the DPC to evaporate and remove moisture. The helium absorbs residual moisture in the DPC and the humidified helium is removed through the vent port. The absorbed water is removed by condensation and/or mechanical means. Dry helium is recirculated through the DPC until the dry helium dew point temperature acceptance criteria for water vapor pressure are achieved and maintained, as specified in the package FSAR and CoC.

7.3.3 Increased Cask Thermal Load Issues

As SNF storage cask designs are certified to store high burnup, high heat-load SNF, licensees should be aware of potential cask loading issues associated with higher thermal loads. This includes an increase in possible hydrogen generation during cask loading; the potential for water thermal expansion; higher package and canister lid temperatures; and increased worker dose rates during cask loading operations.

Hydrogen gas generation can occur due to oxidation of aluminum neutron absorber panels in the canister basket while the canister is filled with water. Additionally, radiolysis of the water in the canister during loading operations can occur in high flux conditions creating additional combustible gases. Cask operating and monitoring procedures include procedures for monitoring for combustible gas concentrations prior to and during canister lid welding operations. One methodology that has been used is to purge the space below the canister lid with inert gas prior to and during lid welding operations to provide additional assurance that flammable gas concentrations will not develop in this space. Hydrogen gas ignition is also discussed in Section 7.3.7.

There is also an increased potential for thermal expansion of water in the canister containing higher decay heat SNF during welding operations. While some amount of water is typically removed from a loaded canister to allow welding operations to proceed, the amount of water is minimized in order to provide shielding during welding operations. Cask loading procedures have numerous steps and warnings to ensure that a sufficient amount of water is removed to allow room for thermal expansion of the water during welding operations. Several sites have experienced thermal expansion of water with sufficient force to cause water to "burp" from the canister.

Increasing the thermal load of a SNF storage cask can result in a rapid increase in the temperature of the storage canister lid after the canister has been backfilled with helium. Lid temperatures can be as high as 300°F, presenting personnel safety issues for staff involved in the cask loading operations. This phenomenon can be mitigated through the connection of a supplemental cooling system through the transfer cask annulus prior to helium backfill.

7.3.4 Top Nozzle Stress Corrosion Cracking

NRC issued Information Notice 2002-09, "*Potential for Top Nozzle Separation and Dropping of a Certain Type of Westinghouse Fuel Assembly*", on February 13, 2002, to alert licensees to fuel handling events that resulted in the separation of the top nozzle from a fuel assembly and the subsequent drop of the assembly during fuel movement. [NRC 2002] The fuel assemblies were identified as Westinghouse fuel assemblies supplied in the early to mid 1980s that contained materials that were susceptible to intergranular stress-corrosion cracking (IGSCC). Licensees have utilized several methods that allow this fuel to be loaded into dry storage.

Licensees that have loaded SNF with top nozzle degradation have done so using special fuel assembly handling tools to grapple the assemblies and consider this to be "normal" handling such that the SNF is not considered to be damaged. If this methodology is used, licensees need to maintain records that indicate the specific assemblies loaded into specific dry storage casks to ensure that future handling of these assemblies will be done with special handling tools.

Another method has been approved in the technical specifications for some 10CFR72 CoCs that allows the use of an instrument tube tie rod (ITTR) developed by Westinghouse to repair these types of assemblies. The ITTR is a stainless steel tube inserted into the central instrument tube. It extends from the top nozzle adapter plate to the bottom nozzle and reinforces the connection between the top nozzle and the rest of the fuel assembly. It is designed to carry the weight of the entire assembly to allow the fuel assembly to be handled with the standard fuel handling tool.

7.3.5 ALARA Considerations

As SNF burnup and heat loads that are approved for storage increase, there is a commensurate increase in radiation dose, requiring additional shielding for storage overpack designs and the potential for increased worker dose during cask loading operations. Several storage overpack designs have been optimized to reduce on-site and off-site radiation dose through additional shielding. Worker dose during package loading operations can be minimized through the use of supplemental shielding. Industry experience has shown that worker dose during cask loading operations typically decreases as the cask loading team gains experience in loading operations during a campaign.

As discussed in ISG-14, Supplemental Shielding, certain dry storage technologies have identified the possible installation of supplemental shielding by general or site-specific licensees to meet the requirements of 10CFR72.104(a) regarding the dose limits for normal conditions of operation. Supplemental shielding, such as an earthen berm or concrete wall surrounding the general licensee's ISFSI, is not a part of the cask system design approved under 10CFR72, Subpart L. However, since NRC staff would consider engineered features for shielding purposes, such as berms or shield walls, to be important to safety, NRC staff provides guidance in ISG-14 regarding the classification as an important to safety component. Since the CoC holder analyzes a "typical" array of storage casks and some distance to the controlled area boundary to calculate off-site dose in its SAR, it is possible that a general licensee's actual site conditions may have a controlled area boundary distance that is less than that assumed in a SAR. A general licensee may include supplemental shielding to ensure compliance with 10CFR72.104(a) for normal operations and anticipated occurrences. General licensees would evaluate the use of supplemental shielding in 10CFR72.212(b) evaluation. A site-specific licensee would include its evaluation of supplemental shielding in its site-specific SAR.

7.3.6 Storage Canister and Transfer Cask Contamination During Loading

Cask design features and loading procedures minimize the potential that the exterior surface of a DPC will become contaminated during SNF loading and canister transfer operations. The exterior surface of the canister is protected from contamination by SNF pool water through the use of clean, demineralized water in the annulus between the canister exterior and the transfer cask. Some designs will utilize an inflatable seal that is installed in the upper end of the annulus between the DPC and transfer cask to prevent SNF pool water from contaminating the exterior surface of the DPC.

Repeated submersion of a SNF transfer cask into the SNF storage pool may result in the exterior of the transfer cask becoming contaminated during repeated fuel transfer campaigns. Despite attempts to decontaminate the transfer cask, radioactive contamination may "weep" from the cask when out of the SNF pool. There are several actions that can help to minimize cask weeping:

- Cask finish: Consideration can be given to buffing the cask to a high finish to minimize the amount of contamination that will migrate into crevices in the cask surface.

- Minimal submersion time: During transfer cask loading operations, the length of time that the cask remains submerged in the SNF storage pool should be minimized. This will help to minimize the amount of contamination that can migrate to the surface of the cask.
- Decontamination: The transfer cask should be decontaminated as soon as practicable after removal from the SNF storage pool. Thorough decontamination can help to reduce the amount of contamination buildup on the cask surface.
- Contamination control: The amount of contamination in the SNF pool or the level of contamination to which the cask surface is exposed should be reduced to the maximum extent practicable. Contamination control measures may include use of an “anti-contamination plate” on the bottom surface of the cask and pre-wetting the transfer cask surface with deionized water prior to immersion in the SNF storage pool.

7.3.7 Hydrogen Gas Ignition

On May 28, 1996, after loading a VSC-24 ventilated storage cask with SNF, an unanticipated hydrogen gas ignition occurred inside the cask during welding of the shield lid at Wisconsin Electric’s Point Beach plant. There was no damage to the SNF in the cask resulting from the gas ignition. The NRC concluded that there were no measurable releases of radioactivity from the cask, no unanticipated exposures to the staff, and no off-site radiological consequences as a result of the event. Wisconsin Electric concluded that the source of the hydrogen was an electrochemical reaction between zinc in the zinc coating used on the storage canister when in contact with the borated water in the SNF pool. The zinc coating was used to prevent corrosion of the multi-assembly sealed basket. The interaction resulted in the production of free hydrogen gas.

On June 4, 1996, the NRC issued confirmatory action letters to the facilities using or planning to use the VSC-24 storage system (Palisades, Point Beach, and Arkansas Nuclear One). The potential for a hydrogen burn to occur during canister closure welding operations was first communicated by the NRC to licensees in 1996, using NRC Bulletin 96-04, *Chemical, Galvanic or other Reactions in Spent Fuel Storage and Transportation Casks,* [NRC 1996b] and Information Notice 96-34, *Hydrogen Gas Ignition During Closure Welding of a VSC-24 Multi-Assembly Sealed Basket*. [NRC 1996c] Hydrogen can be produced during loading operations by radiolysis of the water surrounding the loaded fuel assemblies and the passivation of the aluminum neutron absorbers exposed to water in the canister. Most systems that involve canister welding require monitoring for combustible gases during welding operations. Once the canister cavity is drained, dried, and backfilled with helium, the source of the hydrogen gas is eliminated.

Despite the fact that cask loading procedures are in place to monitor hydrogen generation during the welding process, several recent hydrogen ignition incidents occurred during the loading of casks at a general licensee site. In one incident, after the first pass of the lid-to-shell weld was completed and work was being done on the next weld layer, the licensee elected to suspend combustible gas concentration monitoring. In the second incident, the position of the licensee’s explosive gas monitor relative to the canister vent port did not detect combustible gas concentrations. Both of these situations resulted in hydrogen ignition events. The licensee implemented changes to its cask loading procedures to introduce an inert gas, argon, beneath the canister lid during the lid-to-shell welding process and to relocate the explosive gas monitor sampling point.

7.4 Fuel Loading Campaigns

7.4.1 Cask Loading Strategies

There are primarily two cask loading campaign strategies that are typically used by nuclear operating companies to manage cask loading, full core reserve (FCR) margin, and SNF inventories. The first strategy is “just-in-time” cask loading campaigns in which casks are loaded with a goal of maintaining FCR, or some “prudent operating reserve” in the SNF storage pool. The licensee does not plan to load additional casks to provide additional margin in the SNF storage pool, and there may even be years in which FCR is lost for a short time, but recaptured during the next cask loading campaign. The second type of cask loading campaign that has been used by nuclear operating companies is one that employs larger loading campaigns with a goal of not just maintaining FCR in SNF storage pools, but to gain sufficient space in the pool such that cask loading campaigns can be spaced further apart. For example, a plant might load 10 to 12 casks every three years rather than 5 to 6 casks every eighteen months.

Benefits of just-in-time cask loading are that it minimizes near-term capital and operating expenditures since only enough casks to maintain (or almost maintain) FCR are loaded. Cask loading crews also do not have long periods of time between cask loading campaigns and may result in shorter learning curves for the next cask loading campaign mobilization. Risks associated with a just-in-time loading strategy include unexpected maintenance that requires off-loading the reactor core at a time when the SNF storage pool has less than one FCR; unexpected delays in delivery of storage casks due to licensing issues or fabrication delays that might affect FCR capability; and increased outage times due to space limitations in the SNF storage pool.

Benefits associated with larger loading campaigns include fewer cask loading campaigns over the life of the plant (although the same number of casks would be loaded over the life of the plant) resulting in cost savings associated with mobilization/demobilization for cask loading, training, and dry runs. If a company owns multiple sites with operating ISFSIs and cask loading equipment is shared between sites, this also results in fewer shipments of equipment between sites and subsequent cost savings. Larger loading campaigns would also provide more margin in SNF storage pools over FCR, such that unexpected maintenance requiring off-loading of the reactor core can be accomplished and unexpected delays in delivery of storage casks are more likely to be accommodated. Costs associated with large loading campaigns include increases in near-term capital and operating budgets due to purchasing and loading casks sooner than in a just-in-time loading scenario. Risks associated with larger loading campaigns include much longer loading cycles (months rather than weeks) to complete a loading campaign and possible impacts on plant maintenance activities or other SNF pool activities, and impacts on workers involved in cask loading operations. Shutdown nuclear operating plants have loaded between 15 and 60 casks in extended campaigns with reasonable schedules.

7.4.2 Fleet Loading Team, Unit Loading Team, Contractor Loading

There are three primary ways in which a nuclear operating company can organize its dry storage loading operations, including the use of a fleet loading team, unit loading teams or contractor loading (which may also be referred to as “pool-to-pad” services).

Fleet loading teams have been used by nuclear operating companies that operate ISFSIs at multiple sites. Each site contributes members to the fleet loading team and the team travels to each site, as needed, to perform dry storage loading operations. This includes performing equipment mobilization efforts, dry runs, cask loading operations, and demobilization of equipment including readying equipment for transport to the next ISFSI site. Some companies have used fleet welding teams, in which each site supplies a welder for canister welding operations, and the welding team is deployed to all sites. Other cask loading operations are performed by site personnel. Benefits of using fleet loading teams include shorter learning curves for loading operations mobilization since team members would likely have performed cask loading within the prior 12 to 18 months; coordination of loading activities among sites including sharing of lessons learned on prior loading operations. Depending upon the timing of loading operations at the various sites, certain NRC dry run activities such as welding and unloading activities may be able to be performed one time to meet the requirements for several sites since the same personnel, equipment and procedures are employed at all sites. This requires forward planning to ensure that there is sufficient time to gain NRC approval to allow the company to take credit for NRC dry run activities performed at one site for multiple sites.

Unit loading teams have also been used by nuclear operating companies in which all members of the cask loading team come from the specific unit’s site and are not generally shared with other ISFSI sites that the company may operate. Unit loading teams would perform all cask loading operations including equipment mobilization, dry runs, cask loading operations, and demobilization of equipment. Benefits include having a dedicated team at the site to perform activities. However, unit loading teams will likely have a greater learning curve prior to the start of new cask loading campaigns due to the fact that loading operations might only take place every other year.

Nuclear operating companies have also employed contractor loading teams for a range of cask loading activities. Some nuclear operating companies have contracted for pool-to-pad services in which a contractor team will perform all activities associated with cask loading including equipment mobilization, assistance in procedure development and training, dry runs, cask loading operations, including fuel handling, and welding, and demobilization. Others have contracted for specific contractor assistance including development of procedures and training for cask loading operations, and performing welding operations. Use of contractor loading teams is particularly beneficial for companies that operate only one ISFSI and cannot share loading resources among sites. Additional benefits include a broad range of experience by contractor teams in loading a range of dry storage systems designs, familiarity with different ancillary and support equipment such as welding equipment, rigging, cranes systems, vacuum drying equipment, etc. Contractor teams can also bring lessons learned from other sites as well as shorter preparation times for equipment check out, training, and preparation for dry runs.

7.5 Long-Term Strategic Dry Storage Inventory Management

Due to the long-term prospects for dry storage at nuclear power plant sites that arise from the uncertainties associated with DOE's acceptance of SNF for permanent disposal, it is recommended that nuclear operating companies put in place long-term strategic dry storage loading plans. These long term plans should include a review of current SNF inventories as well as long term projections of SNF discharges (fuel now in core or not yet loaded).

The first step in developing strategic loading plans for SNF dry storage is to determine the characteristics of SNF pool inventories and classify those assemblies that are available for dry storage. Nuclear operating companies have procedures that govern the selection of fuel assemblies for dry storage. Fuel assembly characteristics are typically maintained in a database that is used to verify compliance with the approved contents for the dry storage cask CoC. The development of a long-term strategic loading plan would have a broader range than selection of fuel assemblies for an upcoming dry storage loading campaign in accordance with plant procedures.

A database should be used to track SNF inventories by cycle discharge date; assembly burnup, initial enrichment, fuel design, and classification (damaged fuel, undamaged fuel, top nozzle susceptible, etc.), along with projected dry storage loading dates and the SNF inventory cooling time (watts per assembly) at the projected loading date.[9] Fuel assembly inserts and control components should also be tracked as part of the inventory that might require dry storage. This database would also track characteristics of SNF assemblies that can be qualified to load into dry storage, including the number of damaged fuel assemblies that can be loaded per cask and the characteristics of damaged fuel (burnup, fuel age, enrichment, fuel designs); number of undamaged SNF assemblies and associated characteristics; cask system regional loading requirements and fuel characteristics; allowed fuel assembly inserts, etc. Regular interactions between a nuclear operating company's SNF management staff and nuclear fuel management staff will help to ensure that plans for future fuel designs, fuel management (number of assemblies, batch average burnups, etc.) are taken into account for long-term dry storage loading plans.

Nuclear operating companies must also consider whether there are strategic SNF storage goals that should be factored into its long-term planning. For example, there may be strategic benefits to continuing to store longer-cooled SNF inventories in the SNF storage pools rather than transferring this SNF to dry storage in the near term. These benefits include pool thermal management issues; and consideration of low decay heat SNF inventories as a "resource" that can be mixed with high burnup SNF in the future to support regional loading in dry storage.

Once a nuclear operating company has determined whether to identify strategic SNF storage goals (such as keeping longer-cooled SNF inventories in the SNF storage pools) and the actual and projected SNF inventory and cask allowable contents have been placed in a database, the

[9] One such database is EPRI's *Cask Loader* software. Data loaded into EPRI's *Cask Loader* software includes: bundle/assembly as-built data, exposure, core location, and failure status; core data, including cycle dates, exposures; and cask data including as-built data. *Cask Loader* populates the chosen casks based on cask technical specification requirements and fuel available that meet the requirements of the cask. [EPRI 2009c]

company can determine the specific SNF assemblies that can be qualified for dry storage for specific loading campaigns. Once all approved assemblies are identified, long-term storage requirements can be examined to determine strategic SNF loading schedules. A long-term strategic loading plan can be used to evaluate various scenarios and the long-term impact on future dry storage loading plans.

7.6 Storage and Transport of Damaged Fuel

7.6.1 Standards and Regulatory Guidance for Storage and Transport of Damaged Fuel

The NRC definition of damaged fuel was expanded and revised in ISG-1, Revision 2, Damaged Fuel, September 2005, and incorporated into the draft NUREG-1536. The draft NUREG-1536 definition of damaged fuel states that "*spent nuclear fuel is considered damaged for storage or transportation purposes if it cannot fulfill its regulatory or design function*." As discussed in Section 8.4.17.2, Fuel Classification, of Draft NUREG-1536, previous definitions of damaged fuel identified specific characteristics of the fuel that classify it as damaged, not taking into account whether the fuel was being stored or transported and independent of the design of the storage or transportation system. NRC staff's position regarding classification of damaged fuel, outlined in ISG-1, Rev 2 and Draft NUREG-1536, is that:

> "*damaged fuel is defined in terms of the characteristics needed to perform the fuel-specific and system-related functions. The materials properties, and possibly the physical condition, of a fuel rod or assembly can be altered during irradiation or storage. If this alteration is large enough to prevent the fuel or assembly from performing its fuel-specific or system-related functions during storage, then the fuel assembly is considered damaged.*
>
> *To determine whether a fuel assembly is undamaged, the following should be stated in the SAR:*
>
> *1) The functions the applicant has imposed on the fuel rods and assembly by either fuel specific or system-related functions to meet a regulatory requirement for the designated phase (storage, transportation, or both);*
>
> *2) The mechanisms of change (alteration mechanisms) or the characteristics of the fuel that could potentially cause the fuel to fail to meet its fuel-specific or system-related functions;*
>
> *3) An acceptable analysis showing that the fuel with the designated characteristics will meet the fuel-specific and system-related functions when the mechanisms considered in item #2, above, are evaluated; and*
>
> *4) The physical characteristics of the fuel, based on item #3, above, that could cause the fuel or assembly to be classified as "damaged."*

Draft NUREG-1536 states that licensees may also use a "default" definition of damaged SNF, derived from ANSI N14.33-2005, if they do not want to perform the assessment outlined in item numbers 1 through 4 above. The default definition, however, may not take full advantage of the flexibility of the performance-based definition of damaged fuel provided in NRC's guidance documents. This default definition may be more restrictive than necessary, depending on the design of the storage or transportation cask. For example, the default definition of damaged SNF indicates that SNF must be classified as damaged if an individual fuel rod is missing from an assembly. However, if an analysis shows that all fuel-specific and system-related functions will be met (e.g., subcriticality will be maintained, that the SNF assembly will be retrievable and that the structural properties of the assembly are not compromised by the missing rod) the assembly may be classified as undamaged.

In September 2005, ANSI published ANSI N14.33-2005, *Storage and Transport of Damaged Spent Nuclear Fuel*. [ANSI 2005] The standard focuses on identification and classification of damaged fuel, and preparation and handling of damaged fuel for storage and transport. ANSI N14.33-2005 classifies damaged fuel into the following categories:

- Cladding Damage, Level 1: Cladding defects greater than pinholes or hairline cracks but the fuel assembly still remains intact as a fuel assembly.
- Cladding Damage, Level 2: Fuel that is no longer in the form of a fuel assembly and consists of debris, loose pellets and particles, rod segments, etc.
- Fuel Assembly Mechanical Damage: Fuel assemblies that have structural damage such that they can not be handled by normal methods.
- Pinhole and Hairline Crack: Cladding defects of such a nature that they are very small and tight and do not have the potential for any significant amount of fuel particulate escaping.

ANSI N14.33-2005 provides a fuel classification methodology that ANSI recommends be followed for classifying damaged fuel. A flow chart that outlines the ANSI recommended fuel classification methodology is shown in Figure 7-3.

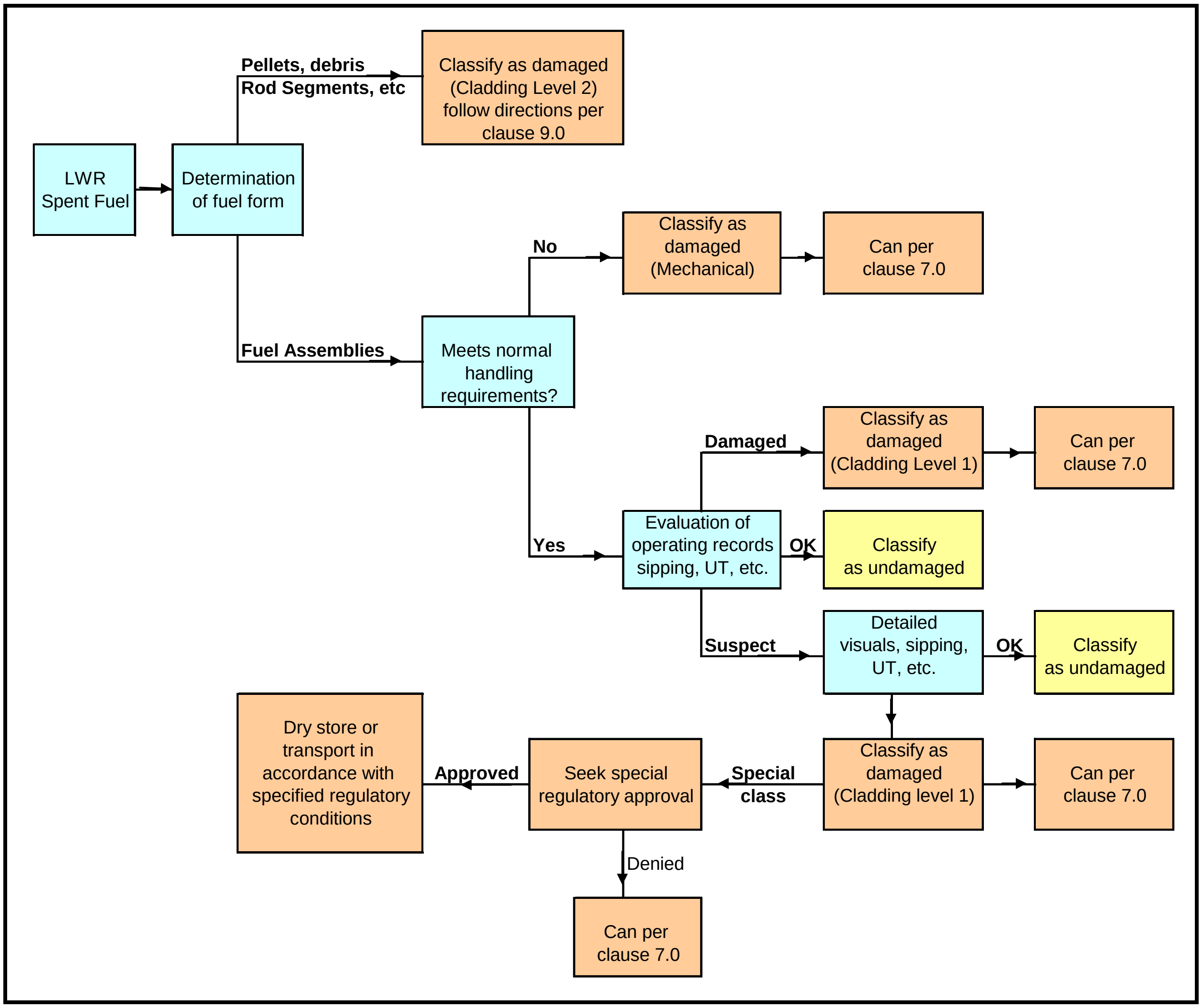

Figure 7-3
Flow Chart for Damaged Fuel Classification [ANSI 2005]

7.6.2 Approved Methods of Storing Damaged Fuel

There are two basic methods that NRC has approved to provide confinement for damaged fuel in storage. The most common method is to place damaged fuel assemblies or fuel debris into damaged fuel containers or cans. Draft NUREG-1536 defines a "can for damaged fuel" as "*a metal enclosure that is sized to confine one damaged spent fuel assembly. A fuel can for damaged spent fuel with damaged spent-fuel assembly contents must satisfy fuel-specific and system-related functions for undamaged SNF required by the applicable regulations*." The can must be individually removable from the cask using normal fuel handling methods (crane and grapple). The can may use a mesh screen to achieve gross particulate confinement but allow water drainage during wet loading operations. The purpose of a damaged-fuel can is to confine gross fuel particles, debris, or damaged assemblies to a known volume within the cask to facilitate criticality, shielding, and thermal requirements and permit normal handling and retrieval from the cask. The damaged-fuel can may need to contain neutron absorbing materials to prevent criticality if many damaged assemblies are co-located in a single cask.

A second method for storage of damaged fuel has been approved by NRC. Instead of “canning” a damaged SNF assembly, Transnuclear has gained approval to confine the damaged fuel bundle with the addition of top and bottom “end caps” – stainless steel caps installed at either end of the basket cell into which the damaged fuel bundle is loaded. The end caps close the ends of the cell and isolate that cell from all the others of the fuel basket, preventing loose particulates from the damaged fuel from migrating into any other cell. This addresses potential concerns regarding criticality, thermal, or shielding that could result from the migration of particles from damaged fuel.

7.7 Very Long Term Storage Issues

With the prospect of SNF being stored at reactor sites for the foreseeable future due to the expected cancellation of the proposed Yucca Mountain repository program, the nuclear industry, NRC and DOE are beginning to examine the issues associated with very long-term storage (defined by NRC as being more than 120 years) and deferred transportation after long-term storage.

7.7.1 EPRI Extended Storage Collaboration Program

Recognizing the likelihood that used nuclear fuel will have to be stored at reactor sites for many decades, EPRI has embarked on an extended storage collaborative research program to define the research and analysis needed to ensure very long-term safe storage, transportation, and monitoring. EPRI held a workshop in November 2009 that brought together representatives from EPRI, nuclear operating companies, the regulatory community, government agencies, used fuel storage vendors, and other stakeholders began defining critical gaps and research needs. [EPRI 2010]

EPRI’s Extended Storage Collaboration Program includes the formation of three committees: a Steering Committee that will provide guidance to the subcommittees and provide continuity to the collaborative research efforts over a multi-year timeframe; a Methodology Subcommittee that will develop model bases, perform gap analysis, and provide guidance to the Experimental Subcommittee; and an Experimental Subcommittee that will provide data collection activities needed to accomplish the goals of the program.

Very long-term SNF management must address multiple components in developing a defensible technical basis: wet and dry storage, as well as both storage and transportation. While the pools are licensed for at least the life of the reactor (typically 40 to 60 years), used fuel in dry storage is stored under separate certificates for an initial 20-year period. These license and certificate time periods will have to be extended. Several site-specific licensees have already extended their licenses for an additional 40 years, and the first 10CFR72 CoC for the dry storage systems loaded under a general license will need to be extended by 2013.[10]

At some unknown time in the future, SNF will need to be transported away from reactor sites after very long term wet or dry storage. It will be necessary to demonstrate that the SNF can be safely transported in accordance with NRC regulations. Dry storage safety related functions must be maintained during very long term storage to ensure that SNF can later be transported. These safety functions include SNF thermal performance, radiological protection, confinement, sub-criticality, and retrievability.

EPRI envisions that due to the expected long-term durations of SNF storage, detailed investigation may be required into a host of potential issues, including:

- Condition of the fuel in dry casks and of the fuel baskets in sealed canisters
- Condition of the fuel and pools in wet storage
- Environmental and handling conditions that could compel repackaging
- Repackaging at sites where reactor decommissioning has taken place (loss of wet pool storage, requirements for dry transfer)
- Long-term lead cask testing of high burnup fuel
- Long-term monitoring requirements
- Effect of long-term storage on transportability

The program has been split into three phases. During Phase 1, participants will review the current technical basis for SNF storage and perform a gap analysis to gain an understanding regarding the time periods covered by existing analysis of storage systems; identification of existing data and operational issues; identification of open items (i.e., the "gaps"), and provide suggested pathways for filling in the gaps. During Phase 2, the program will identify and coordinate experiments, field studies, and additional analyses needed to address the gaps that were identified. During Phase 3, the program will coordinate the collaborative research program that results in a demonstration involving at least one licensed dry storage system loaded with high burnup fuel.

7.7.2 Developing a Regulatory Framework for Long-Term Storage and Deferred Transportation

In anticipation of the Administration's expected termination of the Yucca Mountain repository licensing effort, then-NRC Commissioner Dale Klein proposed, in an internal NRC document written in August 2009, that the Commission direct NRC staff to conduct a thorough review of the NRC's regulatory programs for SNF storage and transportation. [NRC 2009c] The review would evaluate the adequacy of these programs for ensuring safe and secure storage and

[10] As shown in Table 4-2, the first 10CFR72 CoCs were issued in 1990, but none of these packages were loaded under a general license at reactor sites. Thus, the VSC-24 system, which was loaded at three ISFSI sites, will be the first10CFR72 CoC requiring renewal.

transportation for extended time periods beyond the 120 year timeframe that NRC has considered in its existing regulatory framework.

In a February 18, 2010 memorandum to the NRC's Executive Director for Operations, NRC staff were directed to review NRC's regulatory programs for SNF storage and transportation in order to evaluate their adequacy for time periods beyond 120 years. NRC staff was also directed to undertake research to "bolster the technical basis of the NRC's regulatory framework to support extended periods." [NRC 2010a]

In early 2010, NRC staff began to incorporate this new direction with activities that were already under way. NRC staff is performing a regulatory analysis to identify regulatory gaps in the storage and transportation regulations associated with very long-term storage. This "gap" analysis will be used to form the technical basis for any changes to existing regulations.[11]

NRC's budget request for 2011 reflects further planning on this effort. NRC has allocated resources for the development of the technical basis to address safety and security related regulatory gaps in long-term storage and deferred transportation that were identified during the NRC staff review in FY 2010. NRC will identify the need for rulemaking and will evaluate regulatory standards and guidance documents. In addition, research resources have been identified by NRC to support the development of the regulatory framework for extended long-term storage and deferred transportation. Research activities may address such issues as aging management for SNF storage systems, storage and transport of higher burn-up fuels, and long-term cask demonstration programs. NRC's research activities associated with long-term waste management are being coordinated with similar programs being conducted by other organizations, such as DOE, national laboratories, utilities, fuel and storage system vendors and the Extended Storage Collaboration Program. In June 2010, NRC staff requested Commission approval of a project plan that it developed in response NRC 2010a for implementation of an extended storage and transportation program. [NRC 2010b]

7.7.3 Integrated Strategy for SNF Regulatory Activities

In June 2010, the NRC also published a plan for integrating its SNF regulatory activities. The plan states that the reexamination of the U.S. policy for long term management of SNF and HLW may result in the need for extended SNF storage, transportation of older SNF, reprocessing, and possible revision to the regulatory framework for SNF and HLW. The plan is meant to put in place an integrated strategy to assist the NRC "*in addressing ongoing revisions to the national strategy for ensuring public health and safety and the environment in managing SNF and HLW. The purpose of the Plan is to assure that the NRC treats SNF and HLW regulation as a system of interrelated activities so that decisions made about one component or area of the back end of the nuclear fuel cycle adequately consider and integrate related components or areas*." [NRC 2010c]

[11] Similar gap analyses have been performed by EPRI [EPRI 1998; 2002b; 2005a] and are anticipated to be published by DOE later in 2010.

7.8 Quality Assurance, Design Control, and Fabrication Surveillance

The licensee's quality assurance personnel should be prepared to independently assess and inspect activities associated with the design and fabrication of storage components to ensure compliance with regulatory requirements. A well-planned and systematic approach to implementing a QA program should provide adequate assurance that the dry fuel storage project will meet its licensing basis for the system as well as the industry's and public's high expectations for quality. Specific activities conducted by QA personnel include:

- Project-related audits
- Fabrication surveillance
- Quality control inspections
- Participation in resolution of non-conformance reports
- Monitoring of fabricator's QA program implementation
- Monitoring industry related issues

NRC issued Information Notice 95-29, "*Oversight of Design and Fabrication Activities for Metal Components Used in Spent Fuel Dry Storage Systems*", in June 1995, to alert addressees to observed shortcomings in oversight of design and fabrication activities for metal components used in SNF dry storage systems. [NRC 1995] NRC performs inspections of licensees and CoC holders to verify adequate implementation of the regulatory requirements of 10CFR72. NRC conducts QA inspections of designers and fabricators of metal components for dry storage systems. Metal components are normally fabricated at vendor facilities offsite from a power reactor facility.

Licensees are expected to perform surveillance during component fabrication; licensee involvement is recommended to complement the dry storage vendor staff and to ensure a high degree of quality during fabrication. Design control regarding changes to certified storage systems, as well as ISFSI designs, is important to ensure the viability of the technology's design basis. Dry storage vendors have instituted "user groups" through which its technology user can share best practices with other users of a particular storage technology and can collaborate to provide oversight of design and fabrication of storage systems.

In 1999, EPRI published, *Guidelines for Fabrication, Examination, Testing and Oversight of Spent Nuclear Fuel Dry Storage Systems*. The objective of the guidelines is to provide standard guidance to ensure that the licensing/design basis requirements are met during the fabrication, examination, testing, and oversight process for dry storage systems. The guidelines are meant to provide consistency in the application of construction standards; to optimize the use of resources, in particular for oversight functions; and to ensure safety and quality of the dry SNF storage system. [EPRI 1999]

8
AWAY-FROM-REACTOR CENTRAL INTERIM STORAGE

As noted in Section 2, as SNF storage pools were re-racked to the maximum extent possible, nuclear operating companies began to employ interim dry storage technologies to store SNF in certified casks and canister-based systems outside of the storage pool in ISFSIs. Since it is likely that there will be a need for additional SNF storage capacity for many decades into the future, an alternative to storing SNF at reactor sites would be to store SNF at away-from-reactor (AFR) interim storage facility, hereinafter referred to as an AFR generic interim storage facility (GISF). This section provides an overview of the timing and regulations associated with the design, licensing, construction, and operation of an GISF . EPRI Report, 1018722, "*Cost Estimate for an Away-From-Reactor Generic, Interim Storage Facility (GISF) for Spent Nuclear Fuel*," provides additional information regarding generic cost estimates for construction and operation of a GISF. [EPRI 2009a] Other economic studies have been performed by Private Fuel Storage as documented in NRC [2001b].

8.1 Regulations of AFR ISFSI

A GISF that is sited at a greenfield location (meaning that there are no NRC-licensed facilities at the site) would require a site-specific license under 10CFR72. Under current regulations, the initial license term for a GISF may not exceed 20 years from the date of issuance, and licenses may be renewed by the NRC at the expiration of the initial license term upon application by the licensee. In July 2008, NRC released preliminary draft language for public comment that would make changes to 10CFR72 to allow for longer initial and renewal terms for 10CFR72 licenses. That is, the NRC proposes to change the initial license term to 40 years from the date of issuance of a license. In addition, the NRC proposes to allow licenses to be renewed for a period up to 40 years. [NRC 2009b]

In order to obtain a site-specific license, the applicant must demonstrate to the NRC that issuance of the license, authorizing construction and operation of an ISFSI at a designated site, meets all of the technical, administrative, and environmental licensing requirements. Section 3.1 describes the site-specific licensing process. Since an AFR GISF would not have site specific information associated with an operating nuclear power plant, as an at-reactor site-specific licensee would have, the licensee must develop a stand-alone ER, emergency plan, quality assurance plan and physical protection plan to support the LA.

8.2 Schedule for Development of an AFR ISFSI

Based on recent license applications to the NRC for other fuel cycle facilities (such as enrichment facilities), EPRI has developed an estimated schedule for the siting, design, licensing and construction of a GISF. [EPRI 2009a] As shown in Figure 8-1, three phases are associated with the development of a GISF including: a Pre-License Application Phase, a License Application Review Phase, and an Initial Construction/Pre-Operations Phase. During the Pre-License Application Phase, the GISF applicant would develop a program management infrastructure, perform siting studies and geotechnical investigations associated with sites under consideration, and would begin interactions with stakeholders in the areas of the potential sites. Once a site has been selected, the GISF applicant would complete preliminary designs for the GISF, balance of plant facilities, and transportation infrastructure to support completion of the ER and SAR that accompany the facility LA. EPRI assumed the Pre-License Application Phase would take eighteen months to complete.

During the License Application Review Phase, the NRC would review the application for the GISF and would prepare a SER and an Environmental Impact Statement (EIS) to support the licensing decision. An EIS is needed if there is not a NRC-licensed facility at the proposed site for the AFR GISF. During this phase, the applicant would continue its project management functions and stakeholder interactions. In addition, this phase would include technical and legal support to answer NRC requests for additional information (RAIs) regarding the application and to participate in the hearing process. Detailed designs would be completed for the facility, balance of plant facilities, and transportation infrastructure. This phase would also include any state or local review required, such as reviews associated with obtaining building permits. Once NRC has completed its Safety Evaluation Report (SER) and a final EIS has been published, hearings would be held regarding any admitted contentions on the safety and environmental impacts of the facility. Following the completion of hearings on the licensing action, if any, NRC would issue a license under 10CFR72 permitting the construction and operation of the facility.

It should be noted that the time required for the NRC to reach a final decision on a LA for an AFR GISF will depend on the extent to which the SNF storage technology to be referenced in the license has already been certified by NRC as well as whether or not there are intervenors to the licensing proceeding. Based on review times associated with other recent fuel cycle facility licensing actions, EPRI assumes an NRC review time of three years. This assumes that the dry storage technology referenced in the facility license has already been certified by the NRC under 10CFR72.

During the Initial Construction/Pre-Operations Phase, the applicant would continue its project management functions to oversee construction operations and would begin building its staff to operate the facility. Interactions with stakeholder would continue. In addition, this phase would include any engineering or legal support required during construction. This phase would conclude with system start up and dry-run testing which would precede facility operations. EPRI estimates that this phase would take approximately eighteen months.

As presented in Figure 8-1, EPRI assumes that it could take a total of six years to develop a AFR ISFSI from the time that siting studies begin until the facility is ready to begin operation. It should be noted that the schedule for siting, design, construction and licensing for a GISF could take longer than six years. The schedule will be dependent upon the quality of the LA submitted, the extent to which certified dry storage technologies are referenced in the facility design, and whether or not there is intervention in the NRC hearing process.

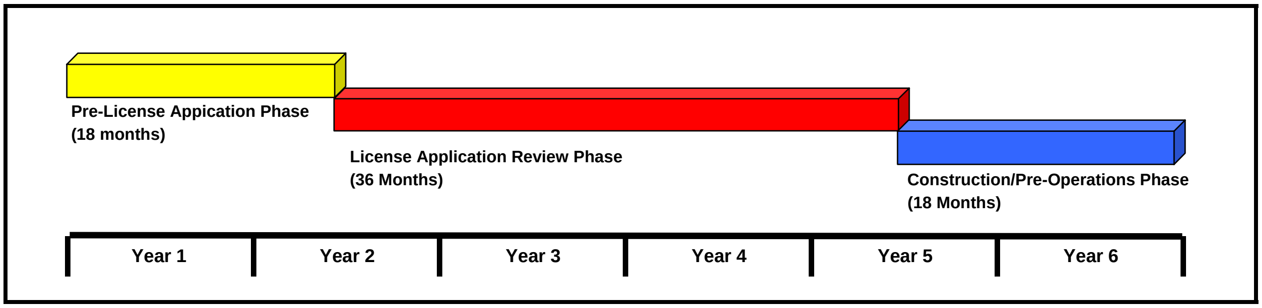

Figure 8-1
Estimated Schedule for Siting, Design, Licensing and Construction of a Away-From Reactor ISFSI [from EPRI 2009a]

8.3 Site Description for a GISF

A generic site plan for a GISF is presented in Figure 8-2. For the purposes of this analysis, EPRI relied upon the site plan, types of facilities, and facility sizes assumed in the Private Fuel Storage LLC Final Environmental Impact Statement (NUREG-1714). [NRC 2001b]

NUREG-1714 was utilized as it provides a recent example of the types of facilities that would be required for an away-from-reactor SNF storage facility. The Owner Controlled Area (OCA) would be bounded by a fence. Within the OCA, there would be a Restricted Area that would contain the Fuel Storage Facility including the storage pads, the Canister Transfer Facility, and a Security/Health Physics building. Other buildings on the GISF site, such as an Administration Building, Concrete Batch Plant, and Operations and Maintenance Building would be located within the OCA, but outside of the Restricted Area security fences. EPRI utilized the description of the various buildings and building specifications assumed in NUREG-1714 as proxies for the buildings for a GISF.

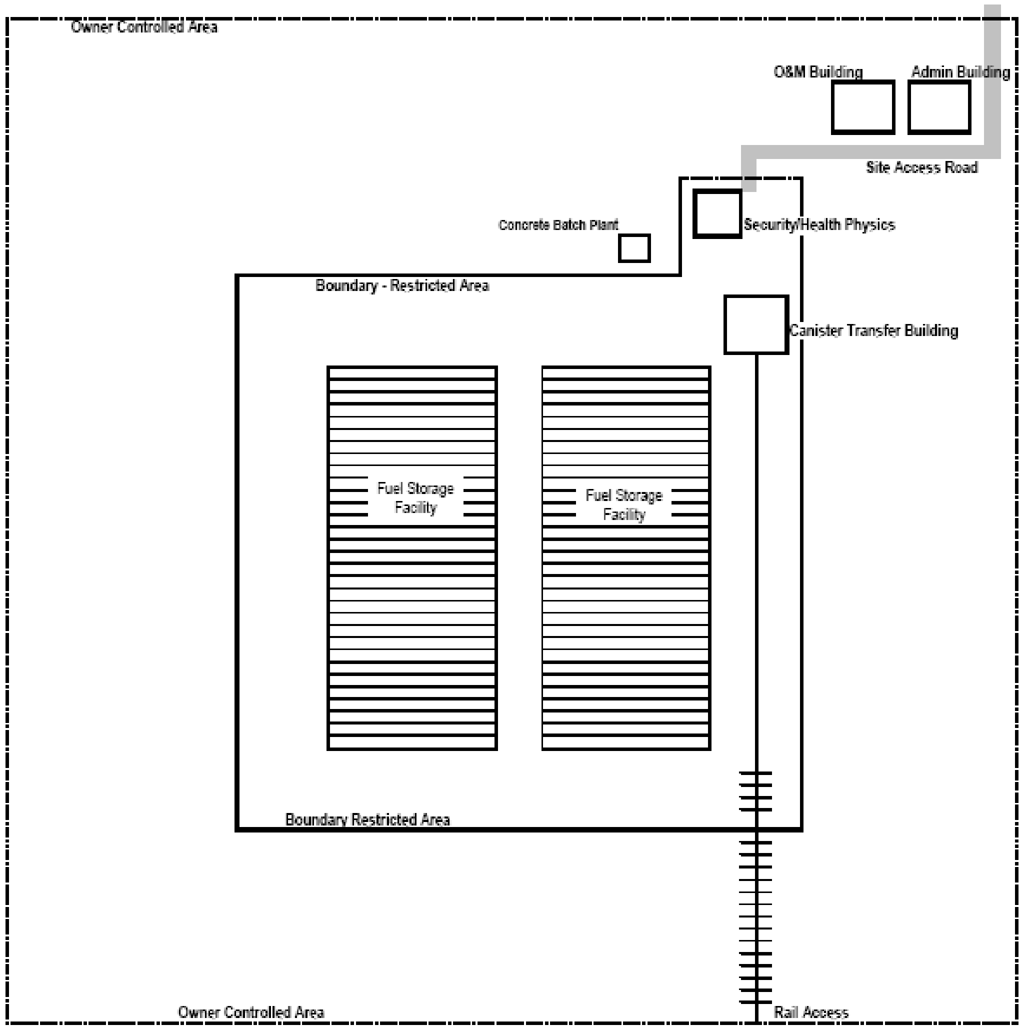

Figure 8-2
Basic Site Plan for a Generic Interim SNF Storage Facility [EPRI 2009a]

9
INSTITUTIONAL CONSIDERATIONS

In addition to the technical and licensing considerations associated with expanding at-reactor SNF storage, there are a number of institutional issues that should be considered early in the licensee's evaluation process. It should be recognized that significant delays to project implementation can occur due to intervention by external parties. An action plan should be developed early in the planning process to address communicating with state and local officials, the media, and members of the public. This action plan would define the company's "messages" regarding the storage expansion project, designate a technical spokesperson and professional spokesperson, and define a schedule for communicating with the defined audiences. The specific actions taken by a utility for communicating with outside parties will be dependent upon an individual utility's location in the country; relationship with local officials, media, and the public; and the plant operating history.

9.1 Early Agency Involvement

Involving regulatory agencies – federal, state and local – at an early stage of planning will often help in establishing a cooperative mode between the nuclear operating company and the agencies. Informing agency management of the company's plans will assist the agency in appropriately allocating its resources and may generate useful feedback in the planning phase of the project. The same considerations apply to notification of political officials. Obviously, agencies and officials prefer to learn of significant plans directly from a nuclear operating company, rather than through the media or as a result of constituent inquiries.

9.2 State Agencies

Each state has its unique regulatory structure that will govern the interface with at-reactor SNF storage projects. In general, there are three components common to most states – public convenience and necessity, environmental, and rates. There are, of course, significant interrelationships among these three components. As the electric utility industry was restructured, some of these relationships changed – particularly for independent power producers.

9.2.1 Public Convenience and Necessity

Some states may require state regulatory approvals before a new at-reactor SNF facility may be sited. This permitting authority is typically under the jurisdiction of the state public service or public utilities commission, and would typically apply to regulated utilities. In many cases, these requirements combine determinations of public convenience and necessity with environmental reviews. The former category may involve consideration of the need for expanded at-reactor storage, alternative methods and sites for providing that storage, consideration of the alternative of not increasing storage (i.e., plant shutdown), and cost comparisons among these alternatives. Achieving consistency between the NRC safety review (both general and site-specific) and those performed by state agencies should be considered.

Two examples of states with a requirement that a company seek a certificate of public convenience and necessity prior to building on-site storage facilities are discussed below. The first example, summarizes the process in Vermont and the outcome for Vermont Yankee. The second example, summarized the process in Minnesota for approval of the Prairie Island ISFSI in the early 1990s, the legislative efforts to expand the Prairie Island ISFSI in the early 2000s, and recent efforts for a certificate of need to build an ISFSI at the Monticello plant.

9.2.1.1 Vermont Certificate of Public Good

Legislation passed by the State of Vermont requires the owners of Vermont Yankee to seek a certificate of public good from the Vermont Department of Public Service prior to the construction of any new SNF storage facility at Vermont Yankee. In April 2006, the Vermont Public Service Board (PSB) issued an order that found construction of dry cask storage would promote the general good of the state of Vermont and that approval of the dry storage facility would result in an economic benefit to Vermont since Vermont Yankee provides one-third of the electricity used in Vermont. However, in granting the certificate of public good, the Vermont PSB required Entergy, the owner and operator of Vermont Yankee, to:

- Submit additional financial assurances for managing Vermont Yankee SNF as long as it is stored on site.
- Update the Vermont Yankee SNF Management Plan to address plans for removal of the SNF in the event the federal government fails to meets its obligations to assume title and possession of this SNF.
- Perform a study that addresses the stability of the river bank adjacent to the proposed storage pad.
- Limit the amount of SNF to be stored at Vermont Yankee to the amount derived from operation of the facility through the end of its existing operating license in 2012.

In addition to the above conditions, Entergy is not permitted to store SNF at the Vermont Yankee ISFSI that is derived from any other source. Storage of SNF from operation of Vermont Yankee after its original March 2012 license expiration date will require the approval of the State of Vermont’s general assembly.

9.2.1.2 Minnesota Certificate of Need

Minnesota law requires that the owners of nuclear power plants in the State submit an application for a Certificate of Need to the Minnesota Public Utilities Commission (MPUC) for additional SNF storage capacity. In support of the certificate of need process, the MPUC develops an EIS regarding the SNF storage expansion project. The applicant must show that the proposed facility meets the criteria established by Minnesota law that governs granting of a certificate of need in that: the facility is needed to ensure future adequacy, reliability, safety and efficiency of the energy supply; a more reasonable and prudent alternative is not available; the consequences of granting a certificate of need outweigh the consequences of denying one; and the design, construction, operation and retirement of the facility will be in compliance with relevant local, state and federal policies, rules and regulations.

In April 1991, Northern States Power Company (now Xcel Energy) submitted an application for a certificate of need to the MPUC. Hearings were held in front of an administrative law judge (ALJ) in late 1991, and in 1992, the ALJ recommended that the PUC deny the certificate of need, stating that "*The likelihood that the dry cask storage would become permanent is so great that it is appropriate to require legislative authorization if the project must go forward immediately*." Despite these recommendations, the MPUC approved the storage of 17 SNF storage casks. Several intervenors appealed the MPUC decision to the Minnesota Court of Appeals, arguing that the additional storage should be classified as “permanent” and that under the 1977 Minnesota Radioactive Waste Management Act, legislative authorization was needed before the MPUC could rule on the matter. Subsequently, the Minnesota Court of Appeals ruled that legislative authorization was needed. After extensive debate in the Minnesota legislature, the 1994 legislature passed a law that permitted NSP to use 17 casks for SNF storage. The law also mandated that NSP install a specific capacity of wind and biomass electrical generating capacity by certain dates in order to be able to load the 17 casks permitted by the legislation.

Since the approval of 17 SNF storage casks were not sufficient to allow Prairie Island to continue to operate for the duration of its operating license, the Minnesota legislature had to address this issue again. In 2003, the legislature authorized the use of up to 48 SNF storage casks, the number specified in the Prairie Island site-specific license. The law also included provisions for renewable energy development and required Xcel Energy to give the Prairie Island Indian Community up to $2.5 million per year for, among other purposes, the acquisition of land away from the Prairie Island facility.

In 2004, Xcel Energy filed an application for a certificate of need to build an ISFSI at its Monticello station. In September 2006, the MPUC approved Xcel Energy's request to build the Monticello ISFSI.

9.2.2 Environmental

Two types of environmental reviews are likely to be relevant. First, many states have created environmental review processes modeled after the National Environmental Policy Act. A state-prepared EIS (sometimes preceded by an applicant's environmental report) may analyze the existing environment, the impacts of the proposed action, mitigation measures, and alternatives to the proposal and set forth a cost-benefit analysis. The preparation of a state EIS typically is in addition to any environmental study (whether an EIS or an EA) that NRC prepares. Consideration should be given to suggesting a coordinated environmental review process between NRC and the state, perhaps with the State adopting portions of the NRC environmental review documents.

The second category of environmental reviews involves state environmental permits. These are usually issued by the state environmental or natural resources department, and they may include air quality, water quality, surface disturbance and land uses. The state may seek to impose radiological conditions on the facility or to review its radiological safety. Nuclear operating company may argue that any limitations involving radiological issues which the state may impose are preempted by federal law and are solely within the jurisdiction of the NRC. The state may assert that, under the federal Clean Air Act, it has authority to set limits on airborne radiological releases.

9.2.3 Rates

The rate treatment given to SNF storage expansion project is an important issue for regulated utilities. In the past, regulated utilities had relatively little difficulty including the capital cost of such projects in the rate base. While not all companies have specifically sought approval to cover SNF storage costs in their electric rates, consideration of rate issues should be addressed during an evaluation of SNF storage alternatives. Two considerations should be borne in mind. First, the rate base disallowances imposed on nuclear plant construction raise the possibility of similar "prudence" disallowances for SNF storage expansions, particularly now that DOE has defaulted on its statutory requirement to begin SNF acceptance by January 31, 1998. State rate regulatory agencies are becoming increasingly concerned that customers are “paying twice” for the back end of the nuclear fuel cycle by paying for at-reactor interim SNF storage beyond 1998 and by paying the 1 mill/KWH Nuclear Waste Fund fee. To date, no commission has barred the utilities from recovering both costs, but it is possible that this will occur in the future.

In states where the electric markets were deregulated, gaining approval from state agencies to recover SNF storage costs in the rates is no longer necessary. Instead, the issue will be the increase in the cost of electricity production due to SNF storage costs and controlling those costs as closely as possible.

9.3 Local Authorities

9.3.1 Permits

Many local jurisdictions require the issuance of building or occupancy permits for the construction or use of major structures. The issuance of state siting, need or environmental permits may not supersede the obligation to obtain local authorizations.

9.3.2 Zoning

Most sites being considered for expanded at-reactor SNF storage will already have received appropriate zoning. Zoning requirements will still need to be verified. Two examples of disputes regarding local zoning permits for construction of ISFSIs at reactor sites are described below.

9.3.2.1 Oyster Creek Zoning Issues

In 1993, General Public Utility (GPU) Nuclear proposed to develop a dry cask facility near its Oyster Creek plant. (Oyster Creek is now owned by Exelon) GPU was required to obtain a zoning change from the local government in order to build the facility. The township granted the zoning change in April 1994, to allow construction of the dry cask facility. Local opponents filed suit in New Jersey state court to require the township to reconsider its action on the grounds that an environmental impact statement was not prepared. Ultimately, the lawsuit was not successful.

9.3.2.2 Connecticut Yankee Zoning Issues

Connecticut Yankee (CY) filed an application with the Town of Haddam, Planning and Zoning Commission to re-zone a portion of its land to build the Haddam Neck ISFSI. The Town of Haddam rejected Connecticut Yankee's application. Subsequently, Connecticut Yankee filed a complaint in Federal Court for denial of the re-zoning. On April 24, 2001, the court ruled in favor of the Town of Haddam's motion to dismiss the CY complaint. CY filed a second lawsuit in November 2001. In January 2002, the Haddam Board of Selectmen agreed to a settlement agreement between Connecticut Yankee and the Town of Haddam. The agreement allowed ISFSI construction to proceed at the location proposed by CY. The settlement also included deed restrictions that prohibit the storage of SNF from any other facility at the Haddam Neck IFSFI.

9.4 Elected Officials

A significant institutional consideration involves notifying elected officials at all levels of a nuclear operating company's plans for expanding at-reactor SNF storage. Particularly where the proposed expansion involves a new facility, such as a dry storage facility, it will generally be advantageous to brief elected officials prior to or at the same time as the nuclear operating company publicly announces the expansion plans. It is usually preferable that these officials be notified by the company before receiving media or constituent inquiries. Most, if not all, companies have existing practices in this area.

9.4.1 Congressional

Notification of the Congressional offices whose districts cover both the plant site and the home office should be considered. In some cases, notification of Senators' offices may also be appropriate. Although Congress has no direct responsibility for individual plant license applications, the political concerns will often reach these offices.

9.4.2 State Legislative and Executive Branches

State officials should be briefed on plans for expanded onsite storage. These may include the Governor's office, the state legislators whose districts cover the plant site and the company's home office, and the state legislators whose legislative committee responsibilities may be affected.

9.4.3 County and Municipal Officials

Often, the elected officials most directly interested in an at-reactor storage expansion project may be those from the county or municipal level. In some cases, these may also be the most supportive of such projects.

It is important that the local community be supportive of the project. This may be accomplished by having plant management brief local officials and explain the need for the storage expansion project.

9.5 Public Affairs and Communications

Nuclear operating companies will usually have a communications plan – or will develop such a plan – to deal with developments such as a planned at-reactor SNF expansion. It is important to define the need for the storage expansion project and to explain why it is necessary.

The communications plan can be used to outline the ways that a company will effectively communicate with its various stakeholders about the SNF storage expansion project. This might include:

- Public opinion polling and the use of focus groups to develop key messages.
- Early and frequent communications with elected Federal, state and local elected officials, near-by residents and opinion leaders, and employees are important factors in a communications strategy.
- Speaker training to key personnel to prepare them for meetings with key stakeholders.
- Development of brochures, presentations, and videos explaining the dry fuel project to use as an outreach tool.
- Conducting site tours and informational meetings for key public officials and media members.
- Conducting an open house for key stakeholders and members of the community with general information about the nuclear power plant and displays of the dry storage system.

9.5.1 Employees

In-house communications would typically be used to notify all company employees of a significant new project.

9.5.2 Media

A variety of techniques are available to inform the media, including press releases, press conferences, and briefings of editorial staffs. It may be useful to contact other companies that have recently implemented SNF expansion projects in order to understand the issues upon which the media focused. It is also recommended that the utility technical and professional spokespersons develop relationships with local media to ensure that they understand the purpose of the storage expansion project.

In some instances, the media will be opposed to a storage expansion project. It is beneficial to be prepared for this situation and be ready to answer questions and provide responses to "disinformation" that may be printed.

9.5.3 Citizens Groups

Companies may consider holding community meetings in the plant area to give members of the public a briefing on the proposed expansion as well as the opportunity to ask questions. Speakers bureau participants should also be prepared to address the project. Consideration may also be given to addressing groups who might generally be considered as opponents of the nuclear facility.

Early identification of possible local intervenor groups or national intervenor groups that might be active in a SNF storage expansion project is recommended. It would also be beneficial to communicate with other utilities that have experienced significant intervention to dry storage projects to gain an early understanding of the issues that may arise.

9.5.4 Business and Labor

Strong support for nuclear construction projects has often come from the labor (particularly construction trades) and business communities. Public affairs efforts aimed at these organizations may be appropriate.

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Electric Power Research Institute
3420 Hillview Avenue, Palo Alto, California 94304-1338 • PO Box 10412, Palo Alto, California 94303-0813 USA
800.313.3774 • 650.855.2121 • askepri@epri.com • www.epri.com