Managing Aging Effects on Dry Cask Storage Systems for Extended Long-Term Storage and Transportation of Used Fuel

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Fuel Cycle Research & Development

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Used Fuel Disposition Campaign

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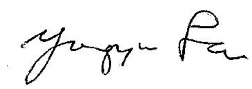
Submitted by:  2012.06.29 14:24:53 -05'00'

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EXECUTIVE SUMMARY

The cancellation of the Yucca Mountain repository program in the United States raises the prospect of extended long-term storage (i.e., >120 years) and deferred transportation of used fuel at operating and decommissioned nuclear power plant sites. Under U.S. federal regulations contained in Title 10 of the Code of Federal Regulations (CFR) 72.42, the initial license term for an Independent Spent Fuel Storage Installation (ISFSI) must not exceed 40 years from the date of issuance. Licenses may be renewed by the U.S. Nuclear Regulatory Commission (NRC) at the expiration of the license term upon application by the licensee for a period not to exceed 40 years. Application for ISFSI license renewals must include the following:

1. Time-limited aging analyses (TLAAs) that demonstrate that structures, systems, and components (SSCs) important to safety will continue to perform their intended function for the requested period of extended operation and

2. A description of the aging management program (AMP) for management of issues associated with aging that could adversely affect SSCs important to safety.

In addition, the application must also include design bases information as documented in the most recent updated final safety analysis report as required by 10 CFR 72.70. Information contained in previous applications, statements, or reports filed with the Commission under the license may be incorporated by reference provided that these references are clear and specific.

The NRC has recently issued the Standard Review Plan (SRP) for renewal of used-fuel dry cask storage system (DCSS) licenses and Certificates of Compliance (CoCs), NUREG-1927, under which NRC may renew a specific license or a CoC for a term not to exceed 40 years. Both the license and the CoC renewal applications must contain revised technical requirements and operating conditions (fuel storage, surveillance and maintenance, and other requirements) for the ISFSI and DCSS that address aging effects that could affect the safe storage of the used fuel. The information contained in the license and CoC renewal applications will require NRC review to verify that the aging effects on the SSCs in DCSSs/ISFSIs are adequately managed for the period of extended operation. To date, all of the ISFSIs across the United States with more than 1,500 dry casks loaded with used fuel have initial license terms of 20 years; three ISFSIs (Surry, H.B. Robinson, and Oconee) have received their renewed licenses for 20 years, and two other ISFSIs (Calvert Cliffs and Prairie Island) have applied for license renewal for 40 years.

This report examines issues related to managing aging effects on the SSCs in DCSSs/ISFSIs for extended long-term storage and transportation of used fuels, following an approach similar to that of the Generic Aging Lessons Learned (GALL) report, NUREG-1801, for the aging management and license renewal of nuclear power plants. The report contains five chapters and an appendix on quality assurance for aging management programs for used-fuel dry storage systems.

- Chapter I of the report provides an overview of the ISFSI license renewal process based on 10 CFR 72 and the guidance provided in NUREG-1927.
- Chapter II contains definitions and terms for structures and components in DCSSs, materials, environments, aging effects, and aging mechanisms.

- Chapter III and Chapter IV contain the TLAAs and AMPs, respectively, that have been developed for managing aging effects on the SSCs important to safety in the dry cask storage system designs described in Chapter V.
- Chapter V contains summary descriptions and tabulations of evaluations of AMPs and TLAAs for the SSCs that are important to safety in the DCSS designs (i.e., NUHOMS®, HI-STORM 100, Transnuclear [TN] metal cask, NAC International S/T storage cask, ventilated storage cask [VSC-24], and the Westinghouse MC-10 metal dry storage cask) that have been and continue to be used by utilities across the country for the dry storage of used fuel.

The goal of this report is to help establish the technical basis for extended long-term storage and transportation of used fuel. Future efforts will include other DCSS designs currently operated under 10 CFR 72.214 (development of additional AMPs and TLAAs that may be deemed necessary) and further evaluation of the adequacy of the generic AMPs and TLAAs that may need augmentation. Industry and site-specific operating experience from the various DCSSs/ISFSIs located across the country will be examined to (a) ascertain the potential aging effects on the SSCs in the DCSSs, thereby enabling a compilation of existing aging management activities, and (b) assess their adequacy for extended long-term storage and transportation of used fuel.

It should be noted that managing aging effects on dry cask storage systems for extended long-term storage and transportation of used fuel “begins” when the used fuel assemblies are loaded into a canister (or cask) under water in the spent fuel pool. The canister (or cask) containing the used fuel assemblies is then drained, vacuum dried, and back-filled with helium before the lid is closed, either by welding or bolted closure. The canister (or cask) is then placed inside a dry cask and transferred to an outdoor concrete pad of an ISFSI, where it would stay for 20 or 40 years of the initial license term (and up to another 40 years for a renewal license term), according to 10 CFR 72.42. More than 1,500 dry casks have begun long-term storage under the initial license terms; some of them have been in storage for over 20 years and are already in the renewed license term for up to 20 years.

Transferring from pool-to-pad or wet-to-dry storage is an abrupt change of environment for the used fuel assemblies, and the effects are most pronounced during vacuum drying, especially for high-burnup fuel, because of the likelihood of cladding radial hydride formation and embrittlement *[Daum et al. 2006, 2008, and Billone et al. 2012]*. The likelihood of this phenomenon will diminish only after the cladding temperature has dropped below 200^{o}C, because of the decrease of fission-product decay heat during prolonged cooling, which may occur 20–25 years after the high-burnup used fuel assemblies are placed under dry storage. Preventing and/or minimizing cladding embrittlement by radial hydrides during drying, transfer, and early stage of storage will, therefore, maintain the configuration of the used fuel in the dry canister (or cask) and ensure retrievability of the used fuel and its transportability after extended long-term storage.

Managing aging effects on dry cask storage systems for “extended” long-term storage of used fuel is no different from that required for long-term storage of used fuel. If aging effects on the SSCs important to safety in the DCSS/ISFSI are not adequately managed for the initial license term of storage, an application for a renewal of license for extended long-term storage is unlikely to be granted by the regulatory authority. Therefore, the same principle and guidance developed by NRC in the Standard Review Plan (SRP) for renewal of used fuel dry cask storage system licenses and Certificates of Compliance (CoC), NUREG-1927, should be applicable to extended long-term storage, as the period of operation, or term, reaches 20, 40, 60, 80, or >120 years. The term in the initial or renewal license is important and indicates a finite period of operation and, although not mentioned specifically in the current regulations, does not rule out license renewal of multiple terms, as long as aging effects are adequately managed.

Managing aging effects on dry cask storage systems for extended long-term storage and transportation of used fuel requires knowledge and understanding of the various aging degradation mechanisms for materials of the SSCs and their environmental exposure conditions for the intended period of operation. The operating experience involving the AMPs, including the past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a determination that the effects of aging will be adequately managed so that the intended functions of the SSCs will be maintained during the period of extended operation. Compared to nuclear power plants, the operating experience of the DCSSs and ISFSIs is not as extensive; however, evaluations have been performed of the NRC's Requests for Additional Information (RAIs) on applications for renewal of licenses for ISFSIs and DCSSs, as well as the applicant's responses to the RAIs, to assess their relevance to the AMPs and TLAAs described in Chapter IV and Chapter III of this report. Those found relevant have been incorporated into the AMPs and TLAAs.

Managing aging effects on dry cask storage systems for extended long-term storage and transportation of used fuel depends on AMPs to prevent, mitigate, and detect aging effects on the SSCs early – by condition and/or performance monitoring. Detection of aging effects should occur before there is a loss of any structure's or component's intended function and includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection, and timing of new/one-time inspection to ensure timely detection of aging effects. The challenges in the detection of aging effects will always be the inaccessible areas for inspection and monitoring and the frequency of inspection and monitoring (i.e., periodic versus continuous).

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ACRONYMS

ACI	American Concrete Institute
AISC	American Institute of Steel Construction
AMA	aging management activity
AMP	aging management plan or program
AMR	Aging Management Review
ANSI	American National Standards Institute
ASME	American Society of Mechanical Engineers
ASR	alkali-silica reaction
ASTM	American Society for Testing and Materials
BWR	boiling water reactor
CB	confinement boundary
CC	criticality control
CFR	Code of Federal Regulations
CoC	Certificate of Compliance
CUF	cumulative usage factor
DCSS	dry cask storage system
DOE	U.S. Department of Energy
DOR	Division of Operating Reactors
DSC	dry shielded canister or dry storage cask
ELTS	Extended Long Term Storage
EPRI	Electric Power Research Institute
FR	fuel retrievability
FSAR	Final Safety Analysis Report
GALL	Generic Aging Lessons Learned
GL	Generic Letter
GTCC	greater than class C
GWd	gigawatt-day
HSM	horizontal storage module
HT	heat transfer
ISFSI	Independent Spent Fuel Storage Installation
ISG	interim staff guidance
ITS	important to safety
ksi	kilopounds per square inch
MPa	megapascal
MPC	Multi-Purpose Canister/Cask

MRS	monitored retrievable storage
MSB	Multi-Assembly Sealed Basket
NAC	Nuclear Assurance Corporation International
NRC	U.S. Nuclear Regulatory Commission
NUHOMS	NUTECH horizontal modular storage
NUREG	U.S. Nuclear Regulatory Commission Regulation
ONS	Oconee Nuclear Station
PMS	pressure-monitoring system
ppm	part(s) per million
PT	liquid penetrant testing
PWR	pressurized water reactor
QA	quality assurance
RG	Regulatory Guide
RS	radiation shielding
SAR	Safety Analysis Report
SER	Safety Evaluation Report
SCC	stress corrosion cracking
SNF	spent nuclear fuel (used interchangeably with used nuclear fuel or used fuel)
SRP	Standard Review Plan
SS	structural support
SSC	structure, system, and component
S/T	storage/transfer
TLAA	time-limited aging analysis
TN	Transnuclear, Inc.
TSC	transportable storage canister
UMF	Universal Multi-Purpose Canister System
U.S.	United States (adjective)
UT	ultrasonic testing
VCC	vertical concrete cask
VSC	ventilated storage cask

I. INTRODUCTION

The cancellation of the Yucca Mountain repository program in the United States raises the prospect of extended long-term storage (i.e., >120 years) and deferred transportation of used fuel at operating and decommissioned nuclear power plant sites. Under U.S. federal regulations contained in Title 10 of the Code of Federal Regulations (CFR) 72.42, the initial license term for an Independent Spent Fuel Storage Installation (ISFSI) must not exceed 40 years from the date of issuance. Licenses may be renewed by the U.S. Nuclear Regulatory Commission (NRC) at the expiration of the license term upon application by the licensee for a period not to exceed 40 years. Application for ISFSI license renewals must include the following: (1) Time-limited aging analyses (TLAAs) that demonstrate that structures, systems, and components (SSCs) important to safety will continue to perform their intended function for the requested period of extended operation; and (2) a description of the aging management program (AMP) for management of issues associated with aging that could adversely affect SSCs important to safety. In addition, the application must also include design bases information as documented in the most recent updated final safety analysis report as required by 10 CFR 72.70. Information contained in previous applications, statements, or reports filed with the Commission under the license may be incorporated by reference provided that those references are clear and specific.

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The goal of this report is to help establish the technical basis for extended long-term storage and transportation of used fuel.

I.1 Overview of License Renewal Process

A licensee or a holder of a CoC must submit a license renewal application at least 2 years before the expiration of a specific license in accordance with the requirements of 10 CFR 72.42(b), or at least 30 days before the expiration of a general license or the associated CoC in accordance with the requirements of 10 CFR 72.240(b). A license or CoC is renewed on the bases that the existing licensing basis continues to remain valid and the intended functions of the SSCs important to safety are maintained during the period of extended operation. Therefore, the license renewal application includes the following: (1) general information related to the licensee/CoC holder and review of regulatory requirements, (2) scoping evaluation to identify the SSCs in the ISFSI or DCSS that are within the scope of license renewal, and (3) for all in-scope SSCs, an Aging Management Review (AMR) that includes (i) identification of their materials of construction and the operating environments, (ii) a list of potential aging effects and degradation mechanisms, and (iii) comprehensive AMPs that manage the effects of aging on SSCs that are important to safety and TLAAs that demonstrate that SSCs important to safety will continue to perform their intended function for the proposed period of extended operation. The application for the renewal of an ISFSI or DCSS license or CoC must contain revised technical requirements and operating conditions (e.g., fuel storage, surveillance and maintenance, and other requirements) for the ISFSI and DCSS that address aging effects that could affect the safe storage of the used fuel. Figure I.1, adapted from NREG-1927, presents a flowchart of the license renewal process.

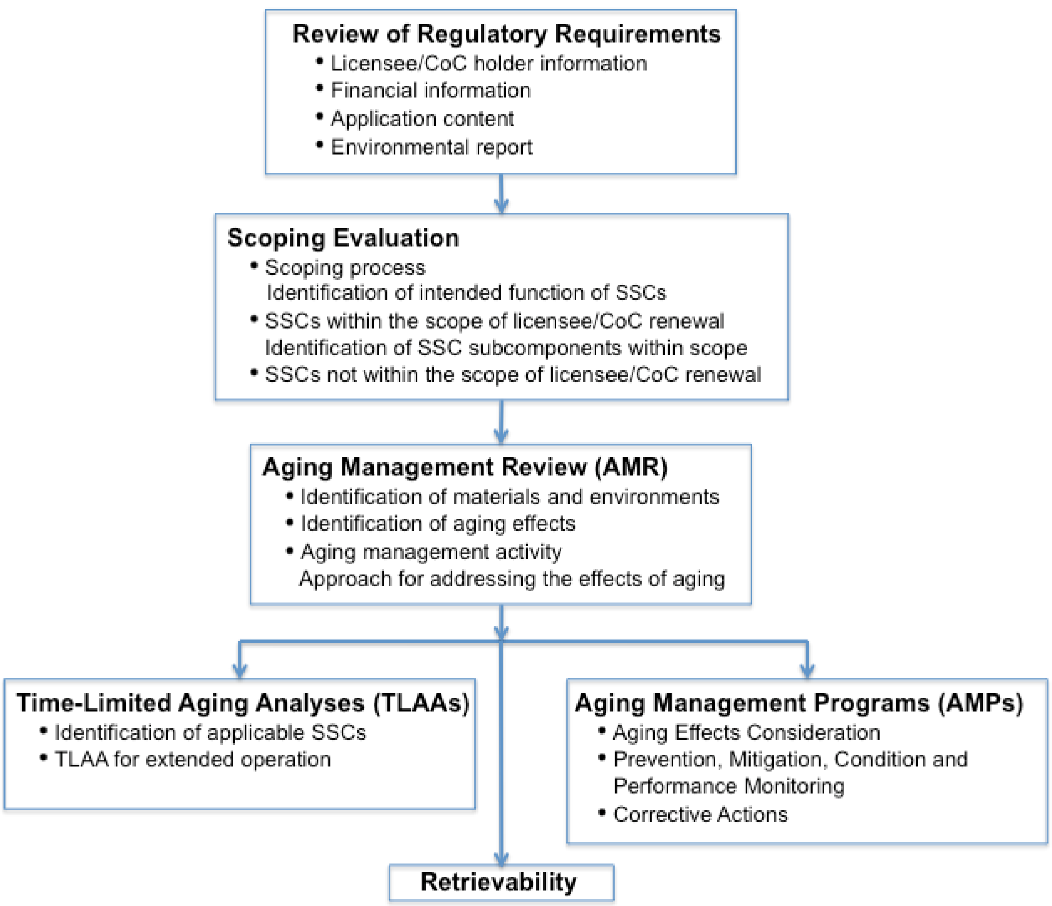

Figure I.1: Flowchart of the license renewal process (Adapted from NUREG-1927).

The items in Figure I.1 summarize the review of regulatory requirements in 10 CFR 72, or any other regulation, that may be applicable to the license renewal process, the scoping evaluation, aging management review (AMR), aging management programs (AMPs), and time-limited aging analyses (TLAAs). Pursuant to 10 CFR 72.42(b), the application must include design bases information as documented in the most recent updated final safety analysis report (SAR) as required by 10 CFR 72.70. Information contained in previous applications, statements, or reports filed with the Commission under the license may be incorporated by clear and specific reference. The contents of the application are in accordance with the applicable requirements in 10 CFR 72.48, "Changes, Tests, and Experiments," and 10 CFR 72.240, "Condition for Spent Fuel Storage Cask Re-approval," and licensee information is in accordance with 10 CFR 72.22, "Contents of Application: General and Financial Information." Also, as required by 10 CFR 51.60, "Environmental Reports: Materials Licenses," and 10 CFR 72.34, "Environmental Report," the renewal application contains an environmental report, or its supplement, that includes the information specified in 10 CFR 51.45, "Environmental Report: General Requirements."

If there have been any modifications in the design of the SSCs or if some components of the ISFSI or DCSS were replaced in accordance with 10 CFR 72.48, all additional information related to the updated final SAR, and changes or additions to the technical specifications, should be included in the application. All supporting information and documents incorporated by reference should be identified. Furthermore, these and other site-specific documents should be reviewed to identify whether any other NRC directives, such as interim staff guidance (ISG) or NUREG reports, are relevant for license renewal.

I.2 Scoping Evaluation

The scoping process identifies the SSCs of the ISFSI or DCSS that should be reviewed for aging effects. Figure I.2, adapted from NUREG-1927, presents a flowchart of the scoping evaluation process. Specifically, the application should include the following information related to the scoping evaluation:

- A description of the scoping process and methodology for inclusion of SSCs in the renewal scope
- A list of the SSCs (and appropriate subcomponents) that are identified as within the scope of renewal, their intended function, and safety classification or basis for inclusion
- A list of the sources of information used for scoping
- Any discussion needed to clarify the process, SSC designations, or sources of information used.

The guidance provided in Section 2.4, NUREG-1927 and the methodology of NUREG/CR-6407 *[McConnell et al. 1996]* are used in this report in determining the classification of DCSS components according to importance to safety. The components in a DCSS may be grouped into three safety categories similar to those defined in Section 3 of NUREG/CR-6407:

- *Category A – Critical to safe operation*: Includes SSCs whose failure could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.

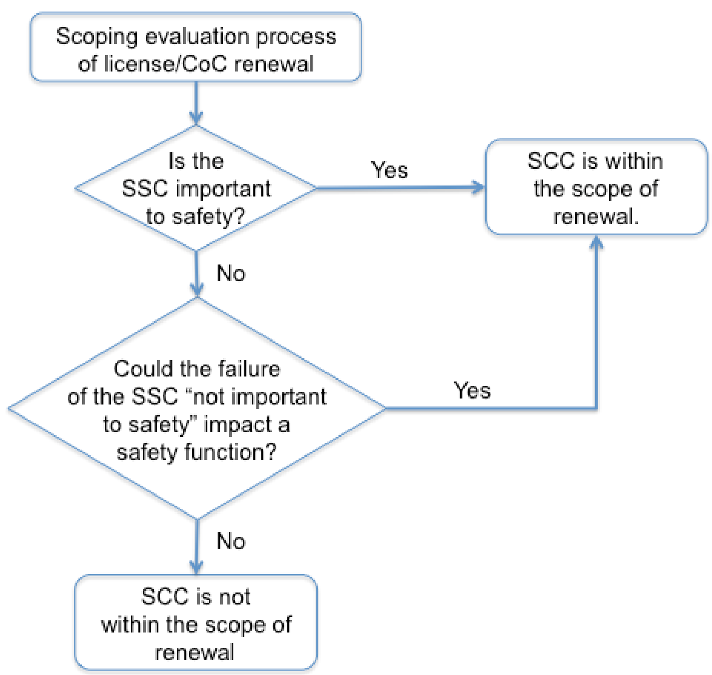

Figure I.2: Flowchart of scoping evaluation (Adapted from NUREG-1927).

- *Category B – Major impact on safety*: Includes SSCs whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with the failure of an additional item, could result in an unsafe condition.
- *Category C – Minor impact on safety*: Includes SSCs whose failure or malfunction would not significantly reduce the packaging (or storage) effectiveness and would not be likely to create a situation adversely affecting public health and safety.

NUREG-1927 defines the important safety functions of the SSCs in a DCSS as (1) criticality, (2) shielding, (3) confinement, (4) heat transfer, (5) structural integrity, and (6) retrievability. For indexing purposes for the SSCs identified in Chapter V of this report, these important safety functions are abbreviated and rearranged with the following definitions:

- *CB – Confinement Boundary*: The components and supporting materials that are incorporated into the storage system design for the purpose of retaining the radioactive material during normal and accident conditions.
- *CC – Criticality Control*: The components and supporting materials that are incorporated into the storage system design for the purpose of maintaining the contents in a subcritical configuration during normal and accident conditions.
- *RS – Radiation Shielding*: The components and supporting materials that are incorporated into the storage system design for the purpose of reducing radiation emitted by the contents during normal and accident conditions.
- *HT – Heat Transfer*: The components and supporting materials that are incorporated into the storage system design for the purpose of removing decay heat under normal conditions and protecting temperature-sensitive components (e.g., lead shielding and seals) under accident conditions.
- *SS – Structural Support*: The components and supporting materials that are incorporated into the storage system design for the purpose of maintaining the structure in a safe condition during normal and accident conditions.
- *FR – Fuel Retrievability*: The components and supporting materials that are incorporated into the storage system design for the purpose of operations support (e.g., for loading, unloading, maintenance, monitoring, or transporting) and the failure of which could impact fuel retrievability.

According to Section 2.4.2 of NUREG-1927, the SSCs within the scope of license renewal generally fall into the following two scoping categories:

1. Those that are classified as important to safety because they are relied upon to do one of the following:
   - Maintain the conditions required by the regulation, license, or CoC to store used fuel safely.
   - Prevent damage to the used fuel during handling and storage.
   - Provide reasonable assurance that used fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

2. Those that are classified as not important to safety but, according to the licensing basis, their failure could prevent fulfillment of a function that is important to safety, or their failure as support SSCs could prevent fulfillment of a function that is important to safety.

The in-scope SSCs are further reviewed to identify and describe the subcomponents that support the intended function or functions of the SSCs. All SSCs that are important to safety, or whose failure may prevent a function that is important to safety, should be identified in the renewal application in accordance with the applicable requirements of 10 CFR 72.3, 10 CFR 72.24, 10 CFR 72.120, 10 CFR 72.122, and 10 CFR 72.236.

Typically, all equipment connected with cask loading and unloading, such as vacuum-drying equipment, welding and sealing equipment, transfer casks and transporter devices, lifting rigs and slings, and other tools, fittings, and measuring devices are not important to safety and, therefore, not within the scope for license renewal. Also, unless the DCSS is anchored to the basemat (pad), the pad is not within the scope of license renewal because it does not perform a safety function or its failure is considered not to impact a safety function. However, for facilities where the DCSS is anchored to the pad, the pad is classified as important to safety and included in the AMR. Also in this report, the approach pad in the NUHOMS® dry storage system is considered within the scope of license renewal because differential settlement of the approach pad may prevent retrieval of the dry shielded canister from the NUHOMS® storage module.

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I.3 Aging Management Review

The purpose of the aging management review (AMR) is to assess all SSCs, determined to be within the scope of renewal, that are subject to aging effects and the associated aging degradation processes, and define potential AMAs needed to manage all aging effects that could adversely affect the ability of these SSCs to perform their intended functions during the period of extended operation. The management of aging effects of SSCs in used-fuel dry casks for long-term storage is similar to that required for renewal of licenses for nuclear power plants under 10 CFR Part 54, “Requirements for Renewal of Operating License for Nuclear Power Plants.” Figure I.3, adapted from NUREG-1927, presents a flowchart of the AMR process for the SSCs in the license renewal of ISFSIs.

Pursuant to 10 CFR 72.24, the SAR for the ISFSI or DCSS contains an analysis and evaluation of the design and performance of SSCs important to safety, with the objective of assessing the impact on public health and safety resulting from the operation of the ISFSI or DCSS. The design-basis information includes determination of (a) the margins of safety during normal operations and expected operational occurrences during the life of the facility, and (b) the adequacy of the prevention and mitigation measures (i.e., the adequacy of the SSCs provided for the prevention of accidents and the mitigation of the consequences of accidents, including natural and manmade phenomena and events). Also, the requirements in 10 CFR 72.122(i) specify that systems that are required under accident conditions should be identified in the SAR.

I.3.1 Relevant Regulations for Aging Management Review

The design criteria contained in the SAR establish the design, fabrication, construction, testing, maintenance, and performance requirements for SSCs important to safety. The requirements for general design criteria for ISFSIs or DCSSs are contained in 10 CFR 72.120(a). The design requirements in 10 CFR 72.120(d) specify that ISFSIs must be designed, made of materials, and constructed to ensure that there will be no significant chemical, galvanic, or other reactions between or among the storage system components, spent fuel, and/or high-level waste including possible reaction with water during wet loading and unloading operations. Also, the behavior of materials under irradiation and thermal conditions must be taken into account.

The overall requirements for protection against environmental conditions and natural phenomena and protection against fire and explosions are contained in 10 CFR 72.122(b) and (c), respectively. Also, as part of the general design criteria, 10 CFR 72.122(f) requires that systems and components that are important to safety must be designed to permit inspection, maintenance, and testing. In addition, 10 CFR 72.122(h) establishes the requirements for confinement barriers and systems, which include capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry storage facilities, periodic monitoring is acceptable provided the monitoring instrumentation system and monitoring period are based on the dry storage cask design requirements. The requirements in 10 CFR 72.122 also specify that used fuel cladding must be protected during storage against degradation that leads to gross rupture or the fuel must be otherwise confined such that its degradation during storage will not pose operational safety problems with respect to its removal from storage. Such capabilities are generally included in the original design of the SSCs. Furthermore, 10 CFR 72.122(l) requires that storage systems must be designed to allow retrieval of the used fuel or high-level radioactive waste for further processing or disposal.

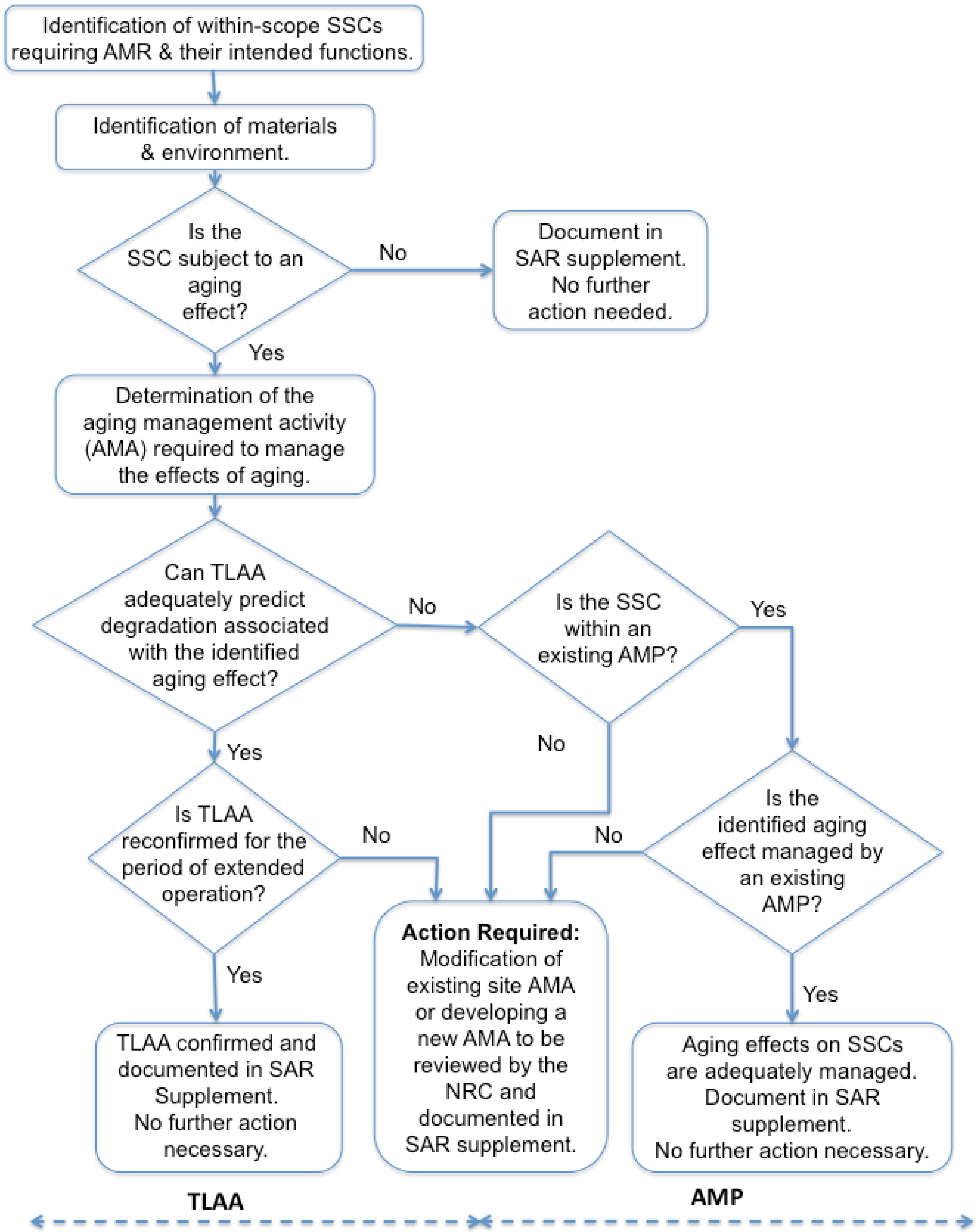

Figure I.3: Flowchart of the aging management review (AMR) process (Adapted from NUREG-1927).

The criteria for used fuel, high-level radioactive waste, and other radioactive waste storage and handling in 10 CFR 72.128(a) specify that the storage facilities for used fuel and such waste must be designed to ensure adequate safety under normal and accident conditions, and include the following:

1. A capability to test and monitor components important to safety,
2. Suitable shielding for radioactive protection under normal and accident conditions,
3. Confinement structures and systems,
4. A heat-removal capability having testability and reliability consistent with its importance to safety, and
5. Means to minimize the quantity of radioactive wastes generated.

The quality assurance and test control requirements must ensure that all testing performed to demonstrate that the SSCs will perform satisfactorily in service is in accordance with 10 CFR 72.162.

The criteria for radioactive materials in effluents and direct radiation from an ISFSI shall meet the requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b). The requirements for the design for nuclear criticality safety, methods for criticality control, and criticality monitoring are described in 10 CFR 72.124. The design requirements specify that the spent-fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the condition essential to nuclear criticality safety. The design for handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations. Furthermore, the methods for criticality control require that where solid neutron-absorbing materials are used, the design must provide for positive means of verifying their continued efficacy.

The quality assurance requirements of 10 CFR 72.170 state that measures must be taken to control materials, parts, or components that do not conform to their respective requirements in order to prevent their inadvertent use or installation. Nonconforming items must be reviewed and accepted, rejected, or repaired in accordance with documented procedures.

The specific requirements for approval and fabrication of spent-fuel storage casks are contained in 10 CFR 72.236. Some of the significant requirements are as follows:

(b) Design bases and design criteria must be provided for SSCs important to safety.

(c) The spent-fuel storage cask must be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

(d) Radiation shielding and confinement features must be provided sufficient to meet the requirements in 10 CFR 72.104 and 72.106.

(e) The spent-fuel storage cask must be designed to provide redundant sealing confinement systems.

(f) The spent-fuel storage cask must be designed to provide adequate heat removal capacity without active cooling systems.

(g) The spent-fuel storage casks must be designed to store the spent fuel safely for the term proposed in the application, and permit maintenance as required.

(l) The spent-fuel storage cask and its systems important to safety must be evaluated, by appropriate tests or by other means acceptable to the NRC, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

(m) To the extent practicable in the design of spent-fuel storage casks, consideration should be given to compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy.

For license renewal, the licensee or certificate holder should review its SAR and define all management activities to ensure that all aging effects are adequately managed and that the SSCs can perform their intended functions, consistent with the existing licensing basis, for the period of extended operations. The AMA of the SSCs that are subject to potential aging effects involves either an AMP or a TLAA, or both.

I.3.2 Time-Limited Aging Analysis

A TLAA is a process to assess SSCs that have a time-dependent operating life, as defined by a design basis such as the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB or Division 3, Subsection WC; American Concrete Institute (ACI) 349 or 318; and American Institute of Steel Construction (AISC) Codes. Time dependency may be fatigue life (cycles) or change in mechanical property such as fracture toughness or strength of materials due to irradiation, or time-limited operation of a component. Examples of possible TLAAs include (a) fatigue of metal and concrete structures and components, (b) corrosion analysis of metal components, (c) time-dependent degradation of neutron absorber materials, (d) time-dependent degradation of radiation shielding materials, (f) environmental qualification of electrical equipment, and (g) other site-specific time-limited aging analyses.

Also, the original design of ISFSIs or DCSSs may not have considered the conditions or aging-related degradation processes associated with extended long-term storage, but these need to be addressed to ensure that the existing licensing basis continues to remain valid during the period of extended operation. Examples of such issues include potential degradation of concrete structures due to long-term exposure to temperatures above 150°C (302°F) or gamma radiation. An AMR of these issues may involve a TLAA. Furthermore, as discussed in Section I.3.1, 10 CFR 72.124(b) requires that where solid neutron-absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. The continued efficacy may be confirmed by a TLAA showing that significant degradation of the neutron-absorbing material cannot occur for the term proposed in the license renewal application.

I.3.3 Aging Management Program

The purpose of the AMP is to ensure that the aging effects do not result in a loss of the intended safety functions of the SSCs that are within the scope of the original license agreements, or in the case of license renewal, for the term of the renewal. Managing aging effects on SSCs in used-fuel DCSSs during long-term storage includes identification of the materials of construction and the environments to which these materials are exposed. Service conditions, such as temperature, wind, humidity, rain/snow/water, marine salt, radiation field, and gaseous environment (e.g., external air environment, internal inert-gas environment such as helium), must be monitored in order to assess and manage the potential aging effects due to environmental degradation of materials. For example, the combination of aging effects and aging mechanisms for concrete structures may include scaling, cracking, and spalling due to freeze-thaw, leaching of calcium hydroxide, aggressive chemical attack, reaction with aggregates, shrinkage, or settlement; loss of material due to corrosion or abrasion and cavitation; and loss of strength and modulus due to elevated temperature or irradiation.

The aging effects/mechanisms for structural steel and reinforcing steel (rebar) may include loss of material due to corrosion; loss of strength and modulus due to elevated temperature; loss of fracture toughness due to radiation; and stress-corrosion cracking (SCC). The aging effects/mechanisms for the cask internals may include loss of material due to corrosion; change in dimensions due to creep; loss of preload due to stress relaxation; loss of fracture toughness due to thermal or neutron embrittlement; and crack initiation and growth due to SCC.

Also, as discussed in Section I.3.2, the original design of the ISFSI or DCSS may not have considered conditions or aging-related degradation processes unique to extended long-term storage, and these must be addressed to ensure that the existing licensing basis continues to remain valid during the period of extended operation. Furthermore, the original design of the dry storage facility may not permit the types of conditions and/or performance monitoring and inspections that are required for extended long-term storage. Therefore, an existing AMP may need to be augmented, or a new site-specific AMP may need to be developed, to ensure that the functional and structural integrity of the storage facility is maintained during the period of extended operation.

However, since ISFSIs or DCSSs consist of mostly passive SSCs, their degradation may not be readily apparent from a simple condition-monitoring program such as periodic inspection, and may require other AMPs that are generally of four types:

- *Prevention*: Programs that keep the aging effects from occurring, e.g., coating programs to prevent external corrosion of a carbon steel overpack component.
- *Mitigation*: Programs that slow the effects of aging, e.g., cathodic protection systems used to minimize corrosion of buried metallic components.

- *Condition Monitoring*: Programs that search for the presence and extent of aging effects, e.g., visual inspection of concrete structures for cracking; sensors that monitor temperatures, pressures, or fission gas such as Kr-85.
- *Performance Monitoring*: Programs that verify the ability of the SSCs to perform their intended safety functions, e.g., periodic radiation and temperature monitoring.

A typical example of a Condition Monitoring program for used-fuel dry casks is an analysis of historic radiation survey data. The operating experience of the used-fuel dry casks, including corrective actions and design modifications, is an important source of information for evaluating the ongoing conditions of the SSCs and for root-cause determinations. Such information is important to safety, and can be used to define mitigation programs that prevent similar recurrences in a timely manner.

While the types and the details of an AMP may vary depending on the specific SSC, the ten elements of an AMP (based on NUREG-1927 and presented in Table I.1) may be used to evaluate the adequacy and effectiveness of the AMP in managing the aging effects on SSCs in DCSSs for extended long-term storage. The evaluation process of an AMP is similar to that used in NUREG-1801, Rev. 2, by utilities and NRC for license renewal of operating nuclear power plants.

Table I.1 Definitions of Ten Elements in an AMP for Managing Aging Effects in SSCs of DCSSs or ISFSIs.

	AMP Element	Description
1.	Scope of the program	The scope of the program should include the specific structures and components subject to an AMR.
2.	Preventive actions	Preventive actions should mitigate or prevent the applicable aging effects.
3.	Parameters monitored or inspected	Parameters monitored or inspected should be linked to the effects of aging on the intended functions of the particular structure and component.
4.	Detection of aging effects	Detection of aging effects should occur before there is a loss of any structure's or component's intended function. This element includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspections to ensure timely detection of aging effects.
6.	Acceptance criteria	Acceptance criteria, against which the need for corrective action will be evaluated, should ensure that the particular structure's and component's intended functions are maintained under all current licensing basis design conditions during the period of extended operation.
7.	Corrective actions	Corrective actions, including root-cause determination and prevention of recurrence, should be timely.
8.	Confirmation process	The confirmation process should ensure that preventive actions are adequate and appropriate corrective actions have been completed and are effective.
9.	Administrative controls	Administrative controls should provide a formal review and approval process.
10.	Operating experience	Operating experience involving the AMP, including past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a determination that the effects of aging will be adequately managed so that the structures' and components' intended functions will be maintained during the period of extended operation.

I.4 Overview of Managing Aging Effects

As stated earlier in Section I, the goal of this report is to help establish the technical basis for extended long-term storage and transportation of used fuel. The report is being prepared in a format similar to that of NUREG-1801, but it is not an NRC document even though the report shares the same principles universally adopted in the aging management of SSCs for license renewal of nuclear power plants. The report follows closely the guidance provided in the Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance, NUREG-1927.

Managing aging effects on dry cask storage systems for extended long-term storage and transportation consists of three steps: (1) perform a scoping evaluation to identify the SSCs in the ISFSI or DCSS that are within the scope of license renewal, their materials of construction, and the operating environments; (2) for each in-scope SSC, list the potential aging effects and degradation mechanisms; and (3) provide an AMR to define comprehensive AMPs and TLAAs that manage the aging effects for each of these SSCs. An overview of the license renewal process, scoping evaluation and aging management review are given in Sections I.1, I.2 and I.3, respectively.

For each DCSS design described in Chapter V, tables have been constructed that identify SSCs and their subcomponents by Item, Structure and/or Component (with rankings of Safety Categories A, B, and C defined in Section I.2), Intended Safety Function (e.g., CB, CC, RS, HT, SS, FR defined in Section I.2), Material, Environment, Aging Effect/Mechanism, AMP or TLAA, and Program Type. Each line item in the table represents a unique component/material/environment/aging-effect combination and the AMP or TLAA for managing the aging effects such that the intended function of the component is maintained during the period of extended operation. Separate line items are included in these AMR tables not only for SSCs that are important to safety, but also for those SSCs that may not have such a function but whose failure could affect performance of the SSCs that are important to safety.

For a specific structure or component listed in the Chapter V tables, if the AMP is consistent with the applicable requirements of 10 CFR 72 and considered to be adequate to manage aging effects, the entry in the “Program Type” column in the table indicates a generic program described in Chapter IV of this report. For these AMPs, no further evaluation is recommended for license renewal. If there is no acceptable AMP to manage the aging effects for a specific combination of component/material/environment/aging effect, the entry in the “Program Type” column recommends further evaluation with details that may become part of a site-specific AMP.

Guidance on the evaluation of TLAAs is provided in Chapter III of this report. TLAAs are required for those SSCs that are subject to time-dependent degradation and meet the criteria of NUREG-1927, Section 3.5, “Identification of TLAAs.” These criteria are similar to those of 10 CFR 54.21(c) for the renewal of operating licenses for nuclear power plants. For example, the guidelines stated in NUREG-1927 Section 3.5.1(5)(i) are equivalent to those of 10 CFR 54.21(i) or (ii) in that the analyses have been projected to the end of the period of extended operation, and the guidelines stated in Section 3.5.1(5)(ii) are equivalent to those of 10 CFR 54.21(iii) in that the effects of aging on the intended functions of the SSC will be adequately managed for the period of extended operation (i.e., potential effects of time-dependent aging degradation evaluated in the TLAA will be managed by an AMP that includes future inspections or examinations).

In general, the nondestructive examination of ISFSI and DCSS components is to be performed in conformance with the ASME Code Section XI requirements. This is consistent with the recommendations of NUREG/CR-7116 *[Sindelar et al. 2011]*, which states that the inspection program recommended for the extended storage and transportation of spent nuclear fuel should be consistent with the requirements of ASME Section XI. NRC ISG-4, Rev. 1, additionally states that "welding processes, weld inspection criteria, and personnel qualifications should be verified as being in conformance with the ASME Code" and that dye-penetrant examinations should be performed in accordance with ASME Code Section V. Both this document and NUREG-1567 specify that the critical flaw size should be calculated in accordance with ASME Section XI methodology.

Quality assurance (QA) for AMPs is discussed in Appendix A. As stated in that appendix, those aspects of the AMR process that affects the quality of safety-related SSCs are subject to the QA requirements of 10 CFR Part 72, Subpart G, "Quality Assurance." For nonsafety-related SSCs subject to an AMR, the QA requirements of 10 CFR Part 50, Appendix B, may be used to address the elements of the corrective actions, confirmation process, and administrative controls for an AMP (see Table I.1) for extended long-term storage and transportation.

The goal of this report is to help establish the technical basis for extended long-term storage and transportation of used fuel. Future efforts will include other DCSS designs currently operated under 10 CFR 72.214 (development of additional AMPs and TLAAs that may be deemed necessary) and further evaluation of the adequacy of the generic AMPs and TLAAs that may need augmentation. Industry and site-specific operating experience from the various DCSSs/ISFSIs located across the country will be examined to (a) ascertain the potential aging effects on the SSCs in the DCSSs, thereby enabling a compilation of existing aging management activities, and (b) assess their adequacy for extended long-term storage and transportation of used fuel.

It should be noted that managing aging effects on dry cask storage systems for extended long-term storage and transportation of used fuel "begins" when the used fuel assemblies are loaded into a canister (or cask) under water in the spent fuel pool. The canister (or cask) containing the used fuel assemblies is then drained, vacuum dried and back-filled with helium before the lid is closed, either by welding or bolted closure. The canister (or cask) is then placed inside a dry cask and transferred to an outdoor concrete pad of an ISFSI, where it would stay for 20 or 40 years of the initial license term, and up to another 40 years for a renewal license term, according to 10 CFR 72.42. More than 1,500 dry casks have begun long-term storage under the initial license terms; some of them have been in storage for over 20 years and are already in the renewed license term for up to 20 years.

Transferring from pool to pad or wet to dry storage is an abrupt change of environment for the used fuel assemblies, and the effects are most pronounced during vacuum drying, especially for high-burnup fuel, because of the likelihood of cladding radial hydride formation and embrittlement *[Daum et al. 2006, 2008, and Billone et al. 2012].* This likelihood will diminish only after the cladding temperature has dropped below 200oC, owing to the decrease of fission-product decay heat during prolonged cooling, which may occur 20 to 25 years after the high-burnup used fuel assemblies are placed under dry storage. Preventing and/or minimizing cladding embrittlement by radial hydrides during drying, transfer and early stage of storage, therefore, will maintain the configuration of the used fuel in the dry canister (or cask), ensure retrievability of the used fuel and its transportability after extended long-term storage.

Managing aging effects on dry cask storage systems for "extended" long-term storage of used fuel is no different from that required for long-term storage of used fuel. If aging effects on the SSCs important to safety in the DCSS/ISFSI are not adequately managed for the initial license term of storage, an application for a renewal of license for extended long-term storage is unlikely to be granted by the regulatory authority. The same principle and guidance developed by NRC in the Standard Review Plan (SRP) for renewal of used fuel dry cask storage system licenses and Certificates of Compliance (CoC), NUREG-1927, therefore, should be applicable to extended long-term storage, as the period of operation, or term, reaches 20, 40, 60, 80, or >120 years. The term in the initial or renewal license is important and indicates a finite period of operation, and, although not mentioned specifically in the current regulations, does not rule out license renewal of multiple terms, as long as aging effects are adequately managed.

Managing aging effects on dry cask storage systems for extended long-term storage and transportation of used fuel requires knowledge and understanding of the various aging degradation mechanisms for materials of the SSCs and their environmental exposure conditions for the intended period of operation. The operating experience involving the AMPs, including the past corrective actions resulting in program enhancements or additional programs, should provide objective evidence to support a determination that the effects of aging will be adequately managed so that the intended functions of the SSCs will be maintained during the period of extended operation. Compared to nuclear power plants, the operating experience of the DCSSs and ISFSIs is not as extensive; however, evaluations have been performed of the NRC’s Requests for Additional Information (RAIs) on applications for renewal of licenses for ISFSIs and DCSSs, as well as the applicant’s responses to the RAIs, to assess their relevance to the AMPs and TLAAs described in Chapter IV and Chapter III of this report. Those found relevant have been incorporated into the AMPs and TLAAs.

Managing aging effects on dry cask storage systems for extended long-term storage and transportation of used fuel depends on AMPs to prevent, mitigate, and detect aging effects on the SSCs early – by condition and/or performance monitoring. Detection of aging effects should occur before there is a loss of any structure’s or component’s intended function, and includes aspects such as method or technique (i.e., visual, volumetric, surface inspection), frequency, sample size, data collection and timing of new/one-time inspection to ensure timely detection of aging effects. The challenges in the detection of aging effects will always be the inaccessible areas for inspection and monitoring and the frequency of inspection and monitoring (i.e., periodic versus continuous).

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II. DEFINITIONS AND TERMS FOR STRUCTURES, COMPONENTS, MATERIALS, ENVIRONMENTS, AGING EFFECTS, AND AGING MECHANISMS

The following tables define the terms used in Chapter V of this report, Application of Aging Management Programs.

II.1 Structures and Components

This report does not address scoping of structures and components (see Table II.1) for extending the duration or term of storage, i.e., license renewal. Scoping is storage-facility specific, and the results depend on the facility design and current licensing basis. The inclusion of a certain structure or component in this report does not mean that this particular structure or component is within the scope of extending storage terms for all facilities. Conversely, the omission of a certain structure or component from this report does not mean that this particular structure or component is not within the scope of extending storage terms for any facilities.

Table II.1 Selected Definitions and Use of Terms for Describing and Standardizing Structures and Components.

Term	Definition as used in this document
Anchor studs	Devices used to attach the DCSS to the ISFSI pad at a site where the postulated seismic event, defined by the three orthogonal zero-period accelerations, exceeds the maximum limit permitted for free-standing installation. The anchor studs are preloaded to a precise axial stress, which is kept below the material yield stress, such that during the seismic event the maximum axial stress remains below the limit prescribed for bolts in the ASME Code, Section III, Subsection NF (for Level D conditions). The preload produces a compressive load, F, at the DCSS/pad interface. This compressive force would generate friction force (μF) at the interface resisting the horizontal (sliding) force exerted on the cask under the postulated design basis earthquake seismic event. A version of HI-STORM, called HI-STORM 100A, is equipped with sector lugs to anchor it to the ISFSI pad. The design of the ISFSI pad and embedment is site-specific and it is the responsibility of the CoC holder and ISFSI licensee. For HI-STORM 100A, the anchor preload is less than 75% of the material yield stress; the coefficient of friction, μ, is less than or equal to 0.53; and the maximum friction force, μF, is many times greater than the sliding force exerted on the cask under the postulated seismic event. *(Holtec International 2010)*
Bolting	Structural bolting, closure bolting, ISFSI pad anchors and all other bolting. Within the scope of license renewal, ISFSI structures contain bolted closures that are necessary for joining the confinement/ containment boundaries or where a mechanical seal is required.
Canister	A metal cylinder that is sealed at both ends and is used to perform the function of confinement, while a separate overpack performs the functions of shielding and protecting the canister from the effects of impact loading. *(NUREG-1571, 1995)*
Cask	A stand-alone device that performs the functions of confinement, radiological shielding, and physical protection of used fuel during normal, off-normal, and accident conditions. *(NUREG-1571, 1995)*

Table II.1 (Cont.)

Term	Definition as used in this document
Confinement	The ability to prevent the release of radioactive substances into the environment. *(NUREG-1571, 1995)*
Confinement boundary	The outline formed by the sealed, cylindrical enclosure of the Multi-Purpose Canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.
Confinement systems	Those systems, including ventilation, that act as barriers between areas containing radioactive substances and the environment. *(10 CFR 72.3)*
Controlled area	The area immediately surrounding an ISFSI over which the licensee exercises authority and within which it performs ISFSI operations. *(10 CFR 72.3)*
Dry cask storage system (DCSS)	Any system that uses a cask or canister as a component in which to store used nuclear fuel without using water to remove decay heat. A DCSS provides confinement, radiological shielding, physical protection, and inherently passive cooling of its used nuclear fuel during normal, off-normal, and accident conditions. *(NUREG-1571, 1995)*
Fuel basket	A honeycombed structural weldment with square openings, which can accept a fuel assembly of the type for which it is designed.
Independent Spent Fuel Storage Installation (ISFSI)	A complex designed and constructed for the interim storage of used nuclear fuel, solid reactor-related greater-than-Class-C (GTCC) waste, and other radioactive materials associated with used fuel and reactor-related GTCC waste storage. *(10 CFR 72.3)*
Multi-Purpose Canister (MPC)	The canister that provides the confinement boundary for the used fuel. The MPC is a welded, all-stainless-steel cylindrical structure with a fixed outer diameter, consisting of baseplate, shell, lid, port covers, and closure ring.
Overpack	A device or structure into which a canister is placed. The overpack provides physical and radiological protection for canisters while allowing passive cooling by natural convection.
Pad	A reinforced concrete basemat on an engineered fill, serving as a foundation for supporting casks. A pad is typically partially embedded and is designed and constructed as foundation under applicable codes such as ACI 318 or ACI 349.
Radiation shielding	Barriers to radiation that are designed to meet the requirements of 10 CFR 72.104(a), 10 CFR 72.106(b), and 10 CFR 72.128(a)(2).
Spent nuclear fuel or spent fuel; Used nuclear fuel or used fuel (the terms “spent fuel” and “used fuel” are interchangeable)	Fuel that has been withdrawn from a nuclear reactor after irradiation, has undergone at least a 1-year decay process since being used as a source of energy in a power reactor, and has not been chemically separated into its constituent elements by reprocessing. Spent or used fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies. *(10 CFR 73.3)*

Table II.1 (Cont.)

Term	Definition as used in this document
Structures, systems, and components (SSCs) important to safety	Those features of the ISFSI and used-fuel storage cask with one of the following functions: (1) to maintain the conditions required to safely store used fuel, high-level radioactive waste, or reactor-related GTCC waste; (2) to prevent damage to the used fuel, high-level radioactive waste, or reactor-related GTCC waste container during handling and storage; or (3) to provide reasonable assurance that used fuel, high-level radioactive waste, or reactor-related GTCC waste can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. *(10 CFR 72.3)*

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II.2 Materials

Table II.2 defines generalized materials that are listed in Chapter V of this report, Application of Aging Management Programs.

Table II.2 Selected Definitions and Use of Terms for Describing and Standardizing Materials.

Term	Definition as used in this document
BISCO NS-3	A neutron shielding material encased in the access door in the early design of the NUHOMS® system. In the later design of the NUHOMS® system, concrete is used as shielding material in the access door.
BORAL	A hot rolled composite plate material consisting of a core of mixed aluminum and boron carbide particles with an 1100 series aluminum cladding on both external surfaces. BORAL absorbs neutrons and is often used as a control rod to help control the neutron flux. The boron carbide contained in BORAL is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. *[Holtec International 2010]*
Concrete	Normal concrete (or plain concrete) is a composite construction material composed of Portland cement or any other hydraulic cement, fine aggregate such as sand, coarse aggregate made of gravel or crushed rocks such as limestone or granite, and with or without chemical admixtures. Reactions between the hydroxyl ions in the Portland cement pore solution and reactive forms of silica in the aggregates (e.g., chert, quartzite, opal, strained quartz crystals), known as “alkali-silica reaction” (ASR), can cause serious expansion and cracking in concrete, resulting in major structural problems and sometimes necessitating demolition. Newly constructed and future ISFSI concrete facilities have increased potential for ASR to occur, because current-generation Portland cements have increased alkali contents that may result in reactivity of aggregates that were not reactive in the past, and the availability of good-quality aggregate materials is becoming limited in many areas of the U.S. Heavyweight concrete (also known as high-density concrete or shielding concrete), the concrete used for radiation shielding, is made by adding heavy natural aggregates such as barites or magnetite. Typically, the density with barites will be about 45% greater than that of normal concrete, while with magnetite the density will be about 60% greater than normal concrete. Reinforced concrete is concrete to which reinforcements (commonly rebars) are added to strengthen the concrete in tension. Concrete is strong in compression but weak in tension.
Elastomers	Flexible materials such as rubber, EPT, EPDM, PTFE, ETFE, viton, vitril, neoprene, and silicone elastomer. Hardening and loss of strength of elastomers can be induced by elevated temperature (above ~ 95°F or 35°C) and additional aging factors (e.g., exposure to ozone, oxidation, or radiation.). *[Gillen and Clough 1981]*
Galvanized steel	Steel coated with zinc, usually by immersion or electrode deposition. The zinc coating protects the underlying steel because the corrosion rate of the zinc coating in dry, clean air is very low. In the presence of moisture, galvanized steel is classified under the category “Steel.”

Table II.2 (Cont.)

Term	Definition as used in this document
Holtite™	The trade name for all present and future neutron-shielding materials formulated under Holtec International's R&D program dedicated to developing shielding materials for application in dry storage and transport systems. The Holtite development program is an ongoing experimentation effort to identify neutron-shielding materials with enhanced shielding and temperature tolerance characteristics. Holtite-A™ is the first and only shielding material qualified under the Holtite R&D Program. *[Holtec International 2010]*
METAMIC	The trade name for an aluminum/boron carbide composite neutron-absorber material qualified for use in the MPCs for the HI-STORM 100 ISFSIs. *[Holtec International 2010]*
Stainless steel	Products grouped under the term "stainless steel" include wrought or forged austenitic, ferritic, or martensitic, precipitation-hardened steel. These materials are susceptible to a variety of aging effects and mechanisms, including loss of material due to pitting and crevice corrosion and cracking due to stress corrosion. Examples of stainless steel designations that comprise this category include SA479-Gr. XM-19, SA564-Gr. 630, SA638-Gr. 660, and Types 304, 304LN, 308, 308L, 309, 309L, 316, and LN. *[Holtec International 2010]*
Steel	In some environments, carbon steel and high-strength low-alloy steel are vulnerable to general pitting and crevice corrosion, even though the rates of aging may vary. Consequently, these metal types are generally grouped under the broad term "steel." Note that this category does not include stainless steel, which has its own category. However, high-strength low-alloy steel with yield strength varying from 105 to 150 kilopounds per square inch is susceptible to SCC. Therefore, when these aging effects are being considered, these materials are specifically identified. Examples of designations for steels for bolts and studs include SA193-Gr. B7; SA194 2H; SA354-Gr. BC; SA540-Grs. B21, B23, and B24; and SA574-Grs. 4142 and 51B37M. Examples of designations for steels for other components include SA515-Gr. 70 and SA516-Gr. 70.

II.3 Environments

Table II.3 defines the standardized environments that are listed in Chapter V of this report, Application of Aging Management Programs.

Table II.3 Selected Definitions and Use of Terms for Describing and Standardizing Environments.

Term	Definition as used in this document
Adverse environment	An environment that is hostile to the component material, thereby leading to potential aging effects. The HI-STORM 100 overpack carbon steel shell can be subjected to an adverse localized marine environment. An adverse environment can be due to any of the following: high relative humidity, high temperature, salty air, or radiation.
Aggressive Environment (steel in concrete)	This environment affects steel embedded in concrete with a pH <5.5 or a chloride concentration >500 ppm or sulfate concentration >1500 ppm. *[NUREG-1557, 1996]*
Confinement environment	Normally, this environment includes inert (He) and non-aqueous (dry) atmospheres. During the initial storage period, the confinement environment may include some residual moisture, which is consumed gradually with time.
Air – outdoor	The outdoor environment consists of moist, possibly salt-laden atmospheric air, ambient temperatures and humidity, and exposure to weather, including precipitation and wind. The component is exposed to air and local weather conditions, including salt-water spray (if present). In marine environments, salt-laden air passing through air ducts may deposit salt on the stainless steel confinement components. Because these components are protected by overpack structure, precipitation does not wash away the deposited salt and the salt accumulates. The accumulated salt may cause SCC if high residual stress, relative humidity and temperature are present.
Air, moist	Air with enough moisture to facilitate the loss of material in steel caused by general, pitting and crevice corrosion. Moist air in the absence of condensation also is potentially aggressive (e.g., under conditions where hygroscopic surface contaminants are present).
Marine environment	An environment consisting of airborne salts (mainly sodium chloride [99.6%] and a small amount of magnesium chloride) in a humid environment. Its pH number and relative humidity may quantify the aggressiveness of the environment. The concentration of chlorides in the environment is dependent on the distance from a large body of salt water, altitude above sea level, and prevailing winds. The effect of sheltering on metal corrosion and SCC resistance also plays an important role. Chlorides that accumulate on an exposed surface can be washed away by precipitation. It has been demonstrated that, for some materials, sheltered exposures facilitate higher corrosion rates than unsheltered exposures in marine environments. With chloride deposits of 20 to 100 mg/m^2 on the surface of Type 304 stainless steel, cracking has been observed at temperatures as low as 86°F. Magnesium chloride (compared to sodium chloride) plays a major role in causing SCC because its deliquescence point is lower (it forms a chloride solution at lower relative humidity). *[Gustafsson and Franzén 1996; Meira et al. 2006; Caseres and Mintz 2010]*

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II.4 Aging Effects

Table II.4 defines the standardized aging effects due to associated aging mechanisms that are listed in Chapter V of this report, Application of Aging Management Programs.

Table II.4 Selected Use of Terms for Describing and Standardizing Aging Effects

Term	Usage in this document
Concrete cracking and spalling	Cracking and exfoliation of concrete as the result of freeze-thaw, aggressive chemical attack, and reaction with aggregates.
Cracking	Synonymous with the phrase “crack initiation and growth” in metallic substrates. Cracking in concrete is caused by restraint shrinkage, creep, settlement, and aggressive environments.
Cracking, loss of bond, and loss of material (spalling, scaling)	Phenomena caused by corrosion of steel embedded in concrete.
Cracks; distortion; increase in component stress level	Phenomena in concrete structures caused by settlement. Although settlement can occur in a soil environment, the symptoms can also be manifested in either an air-indoor uncontrolled or air-outdoor environment.
Cumulative fatigue damage	Damage due to fatigue, as defined by ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB. *[ASME 2004]*
Expansion and cracking	Phenomena within concrete structures caused by reaction with aggregates.
Increase in porosity and permeability, cracking, loss of material (spalling, scaling), loss of strength	Phenomena within concrete structures caused by aggressive chemical attack. In concrete, the loss of material (spalling, scaling) and cracking can also result from freeze-thaw processes, and loss of strength can result from leaching of calcium hydroxide from the concrete.
Loss of confinement	Behavior of concrete caused by pitting, crevice corrosion, or SCC of an austenitic stainless steel confinement boundary.
Loss of material	A phenomenon due to general corrosion, pitting corrosion, galvanic corrosion, crevice corrosion, erosion, and aggressive chemical attack. In concrete structures, loss of material can also be caused by abrasion or cavitation or corrosion of embedded steel.
Loss of material, loss of form	In earthen water-control structures, phenomena resulting from erosion, settlement, sedimentation, frost action, waves, currents, surface runoff, and seepage.
Loss of preload	A phenomenon due to stress relaxation and self-loosening (which includes vibration, joint flexing, and thermal cycles). *[EPRI NP-5067 1987/1990; EPRI TR-104213 1995]*
Reduction in concrete anchor capacity due to local concrete degradation	A phenomenon resulting from service-induced cracking or other concrete aging mechanisms.
Reduction in foundation strength, cracking, differential settlement	Phenomena that can result from erosion of a porous concrete subfoundation.
Reduction of heat transfer	A phenomenon that can result from the blockage of air duct screens by blowing debris, animals, etc.
Reduction of neutron-absorbing capacity (or shielding capacity)	Capacity reduction phenomenon that can result from degradation of neutron-absorbing materials such as BISCO NS-3.

Table II.4 (Cont.)

Term	Usage in this document
Reduction of strength and modulus	In concrete, a phenomenon that can be attributed to elevated temperatures (>150°F general; >200°F local).
Wall thinning	A specific type of loss of material attributed in the AMR line items to general corrosion.

II.5 Significant Aging Mechanisms

An aging mechanism is considered significant when it may result in aging effects that produce a loss of integrity and/or functionality of a component or structure during the current or extended license period, if allowed to continue without mitigation. Table II.5 defines the standardized aging mechanisms that are listed in Chapter V of this report, Application of Aging Management Programs.

Table II.5 Selected Definitions and Use of Terms for Describing and Standardizing Aging Mechanisms.

Term	Definition as used in this document
Abrasion	As water migrates over a concrete surface, it may transport material that can abrade the concrete. The passage of water also may create a negative pressure at the water/air-to-concrete interface that can result in abrasion and cavitation degradation of the concrete. This damage may result in pitting or aggregate exposure due to loss of cement paste. *[NUMARC 1991a]*
Aggressive chemical attack	Concrete, being highly alkaline (pH >12.5), is degraded by strong acids. Chlorides and sulfates of potassium, sodium, and magnesium may attack concrete, depending on their concentrations in soil/ground-water that comes into contact with the concrete. Exposed surfaces of Class 1 structures may be subject to sulfur-based acid-rain degradation. The minimum thresholds causing concrete degradation are 500 ppm chlorides and 1500 ppm sulfates. *[NUMARC 1991a]*
Corrosion	Chemical or electrochemical reaction between a material, usually a metal, and the environment or between two dissimilar metals that produces a deterioration of the material and its properties.
Corrosion of carbon steel storage overpack components	Corrosion can occur on carbon steel components including overpack shell, lid studs and nuts, baseplate, sector lugs, covers for concrete shielding blocks, etc.
Corrosion of embedded steel	If the pH of concrete in which steel is embedded is reduced below 11.5 by intrusion of aggressive ions (e.g., chlorides at >500 ppm) in the presence of oxygen, the embedded steel may corrode. The leaching of alkaline products through cracks, entry of acidic materials, or carbonation may also cause a reduction in pH. Chlorides may be present in the constituents of the original concrete mix. The properties and types of cement, aggregates, and moisture content affect the severity of the corrosion. *[NUMARC 1991b]*
Freeze-thaw, frost action	Repeated freezing and thawing can cause severe degradation of concrete, characterized by scaling, cracking, and spalling. The cause is water freezing within the pores of the concrete, creating hydraulic pressure. If unrelieved, this pressure will lead to freeze-thaw degradation. If the temperature cannot be controlled, other factors that enhance the resistance of concrete to freeze-thaw degradation are (a) adequate air content (i.e., within ranges specified in ACI 301-84), (b) low permeability, (c) protection until adequate strength has developed, and (d) surface coatings applied to frequently wet-dry surfaces. *[NUMARC 1991b]*

Table II.5 (Cont.)

Term	Definition as used in this document
General corrosion	General corrosion, also known as uniform corrosion, proceeds at approximately the same rate over all parts of a metal surface. Loss of material due to general corrosion is an aging effect requiring management for low-alloy steel, carbon steel, and cast iron in outdoor environments. Some potential for pitting and crevice corrosion may exist even when pitting and crevice corrosion is not explicitly listed in the aging effects/aging mechanism column in NUREG-1801, Rev. 2, AMR line-items and when the descriptor may only be loss of material due to general corrosion. This is so because the visual inspection required for detecting the effects of general corrosion acts as a de facto screening for pitting and crevice corrosion, since the symptoms of general corrosion will be noticed first.
Settlement	Settlement of a containment structure may be due to changes in the site conditions (e.g., water table). The amount of settlement depends on the foundation material. *[Gavrilas et al. 2000]*
Stress corrosion cracking (SCC)	The cracking of a metal produced by the combined action of corrosion and tensile stress (applied or residual), especially at elevated temperature. SCC is highly chemically specific in that certain alloys are likely to undergo SCC only when exposed to certain types of chemical environments. SCC includes intergranular SCC, transgranular SCC, and low-temperature crack propagation as aging mechanisms.
Thermal effects, gasket creep, and self-loosening	Loss of preload due to gasket creep, thermal effects (including differential expansion and creep or stress relaxation), and self-loosening (which may be due to vibration, joint flexing, cyclic shear loads, or thermal cycles). *[Bickford 1995]*
Thermal fatigue	The progressive and localized structural damage that occurs when a material is subjected to cyclic loading. The maximum stress values are less than the ultimate tensile stress limit, and may be below the yield stress limit of the material. Higher temperatures generally decrease fatigue strength. Thermal fatigue can result from variations in ambient temperature, increase in temperature due to reduction in heat transfer capability, and differential thermal expansion of the adjacent components.
Weathering	The mechanical or chemical degradation of external surfaces of materials when exposed to an outside environment.
Wind-induced abrasion	Abrasion that occurs when the fluid carrier of abrading particles is wind rather than water/liquids. (See “abrasion.”)

II.6 References

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NUREG-1571, Information Handbook on Independent Spent Fuel Storage Installations, U.S. Nuclear Regulatory Commission, Washington, DC, 1995.

III. TIME-LIMITED AGING ANALYSES

III.1 Identification of Time-Limited Aging Analyses

Section 3.5, "Time-Limited Aging Analysis Evaluation," in NUREG-1927, "Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance," defines a TLAA as a process to assess SSCs that have a time-dependent operating life. Thus, TLAAs are certain site-specific safety analyses of SSCs that have a time-dependent operating life, as defined in the design basis, and are based on an explicitly assumed design life (for example, aspects of the used-fuel canister design or time-dependent degradation of the neutron-absorber material). Time dependency may relate to fatigue life (number of cycles to predicted failure) or time-limited design life (number of operating hours until replacement), or time-dependent degradation of mechanical properties of the material (aging effects). Furthermore, TLAAs may have developed since the issuance of ISFSI licenses or DCSS CoCs. Managing aging effects for license renewal, therefore, would require an evaluation of those TLAAs that (i) include time-limited assumptions, (ii) were utilized in determining the acceptability of SSCs within the scope of license renewal, and (iii) are based upon the existing license term plus the period of extended operation requested in the renewal application. Examples of possible TLAAs are (1) fluence level that causes embrittlement of metallic components, (2) time-dependent degradation of neutron-absorber material or radiation shielding material, and (3) thermal fatigue of canister shells or concrete structures.

Analogous to the requirements of 10 CFR 54.21(c) for the renewal of operating licenses for nuclear power plants, NUREG-1927 states that the applicant for an ISFSI license renewal or DCSS CoC is required to evaluate TLAAs. Section 3.5.1 of NUREG-1927 provides a brief guidance on the identification of TLAAs.

III.1.1 Description of the Time-Limited Aging Analyses

Pursuant to 10 CFR 72.24, a license renewal application to store used fuel in an ISFSI must include the design criteria for the proposed storage installation, which establish the design, fabrication, construction, testing, maintenance and performance requirements of SSCs important to safety as defined in 10 CFR 72.3. In addition, "Subpart F, General Design Criteria" of 10 CFR 72 includes the following Subsections that may be important in the identification of TLAAs:

10 CFR 72.120(d) *General consideration*: The behavior of materials (used in construction of ISFSIs) under irradiation and thermal conditions must be taken into account.

10 CFR 72.122(b) *Protection against environmental conditions and natural phenomena*: SSCs important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornados, lightning, hurricanes, floods, tsunamis, and seiches, without impairing their capability to perform safety functions.

10 CFR 72.122(f) *Testing and maintenance of systems and components*: Systems and components that are important to safety must be designed to permit inspection, maintenance, and testing.

10 CFR 72.122(h) *Confinement barriers and systems*: The monitoring period must be based upon the used-fuel storage cask design requirements.

10 CFR 72.124(b) *Method of criticality control*: When practicable, the design of an ISFSI must be based on favorable geometry, permanently fixed neutron-absorbing materials (poisons), or both. Where solid neutron-absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry used-fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron-absorbing materials cannot occur over the life of the facility.

10 CFR 72.128(a) *Criteria for used fuel and high-level radioactive waste storage and handling system*: Used-fuel storage must be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with (1) a capability to test and monitor components important to safety, (2) suitable shielding for radioactive protection under normal and accident conditions, (3) confinement structures and systems, (4) a heat-removal capability having testability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive wastes generated.

Managing aging effects for license renewal must demonstrate that the existing licensing basis remains valid and the intended functions of the SSCs important to safety are maintained during the period of extended operation. Therefore, the design basis documents should be reviewed to identify TLAAs, including time-dependent degradation of mechanical properties of materials due to aging effects. The impact of such degradation on the design basis or on the design margins should also be evaluated. The applicant should ensure that the license renewal application does not omit any TLAAs. The license renewal application should also include a list of site-specific exemptions granted in accordance with 10 CFR 72.7 that are based on TLAAs.

III.1.1.1 Acceptance Criteria

By definition, TLAAs are aging analyses of safety-significant SSCs within the scope of the license renewal. The acceptance criteria for the TLAAs delineate acceptable methods for meeting the requirements in 10 CFR 72. For existing or newly identified SSCs with a time-dependent operating life, the identification of TLAAs should verify that the TLAAs meet the following five criteria listed in Section 3.5.1 of NUREG-1927:

(1) The TLAA should involve time-limited assumptions defined by the current operating term (e.g., 20 years). The defined operating term should be explicit in the analyses. Simply asserting that the SSC is designed for a service life or ISFSI life is not sufficient. Calculations, analyses, or testing that explicitly includes a time limit should support the assertions.

(2) The TLAA should already be contained or incorporated by reference in the design documents. Such documentation includes the (1) SAR, (2) safety evaluation report (SER), (3) technical specifications, (4) correspondence to and from NRC, (5) QA plan, and (6) topical reports included as references in the SAR.

(3) The TLAA must address SSCs that are within the scope of license renewal and have a predetermined lifespan.

(4) The TLAA must consider the extended operational lifetime of any SSC materials that have a defined lifetime limit (e.g., thermal fatigue condition).

(5) The TLAA should provide conclusions or a basis for conclusions regarding the capability of the SSC to perform its intended function through the license period of extended operation. The TLAA must show either one of the following:

(i) The analyses have been projected to the end of the period of extended operation.

(ii) The effects of aging on the intended function(s) of the SSC will be adequately managed for the period of extended operation. Component replacement is an acceptable option for managing the TLAA.

III.1.2 Dispositioning the Time-Limited Aging Analyses

The licensee should provide a justification and basis for addressing each SSC that has a predetermined lifespan or is subject to time-dependent aging degradation and is determined to be within the scope of renewal. For any analyses that are not identified as TLAAs, the applicant should verify that they do not meet at least one of the five acceptance criteria described in Subsection III.1.2. Information regarding the methodology used for identifying TLAAs may be helpful in evaluating analyses that did not meet the five criteria discussed below.

(1) *Have a time-limited assumption defined by the current operating term (e.g., 20 years)*. The defined operating term should be explicit in the analyses. The assertion that the SSC is designed for a specific service life should be supported by a calculation, analysis, or testing that explicitly includes a time limit. The TLAA must consider the extended operational lifetime of any SSC materials that have a defined lifetime limit (e.g., thermal fatigue condition).

(2) *Are contained or incorporated by reference in the design documents*. The design documents include the technical specifications in accordance with the requirements of 10 CFR 72.44 and a summary statement of the justification for these technical specifications (as defined in 10 CFR 72.26), or the licensee commitments documented in the site-specific documents contained or incorporated by reference in the design basis analyses, including but not limited to the material license, Final Safety Analysis Report (FSAR), docketed licensing correspondence, NRC SERs, hazard analyses, the QA plans, vendor topical reports incorporated by references in the FSAR, and other NRC communications.

(3) *Address SSCs that are within the scope of license renewal and have a predetermined lifespan*. Chapter 2 of the NUREG-1927 for renewal of ISFSI licenses and DSCC CoCs provides the regulatory requirements and guidance on the scoping and screening methodology for the inclusion of SSCs in the renewal process.

(4) *Consider the extended operational lifetime of any SSC materials that have a defined lifetime limit*. The analyses involve conclusions or provide the basis for conclusions related to the license renewal process. The effects of aging degradation should be incorporated in these analyses. These effects include but are not limited to loss of material, change in dimension, change in material properties, loss of toughness, loss of pre-stress, settlement, cracking, and loss of dielectric properties. Analyses that do not affect the intended function of the SSCs are not considered TLAAs.

(5) *Provide conclusions or a basis for conclusions regarding the capability of the SSC to perform its intended function through the license period of extended operation*. An analysis is considered relevant if it can be shown to have a direct bearing on the action

taken as a result of the analysis. Such analyses would have provided the basis for the applicant's initial safety determination, and without these analyses the applicant may have reached a different safety conclusion.

III.1.3 Final Safety Analysis Report Supplement

The specific criterion for meeting the guidance of NUREG-1927, Section 1.4.4, is that the renewal applications for IFSFI licenses and DSCC CoCs should include a supplement to the updated FSAR that provides a summary description of the evaluation of TLAAs for the period of extended operation such that later changes can be controlled by 10 CFR 72.48. If the TLAAs have been dispositioned in accordance with option (ii), the FSAR supplement should include an adequate description of the proposed AMP to manage the aging effects of fatigue damage on the intended function of the SSCs during the period of extended operation. It should also state that the results of this activity are evaluated relative to the applicable codes, standards, and guidelines. The description should contain sufficient information associated with the TLAAs regarding the basis for determining that the applicant has followed the guidance of NUREG-1927, Sections 1.4.4 and 3.5.1.

The license renewal process requires the applicant to update its FSAR to include this FSAR supplement at the next update required pursuant to 10 CFR 72.70. As part of the license condition, until the FSAR update is complete, the applicant may make changes to the programs described in its FSAR supplement without prior NRC approval, provided that the applicant evaluates each such change pursuant to the criteria set forth in 10 CFR 72.48. If the applicant updates the FSAR to include the final FSAR supplement before the license is renewed, no condition will be necessary. The applicant need not incorporate the implementation schedule into its FSAR. However, the applicant should identify and commit in the license renewal application to any future AMAs, including enhancements and commitments to be completed before the period of extended operation. The license renewal applicant should commit to complete these activities no later than this specified date, i.e., prior to entering the period of extended operation.

TLAAs that need to be included in the renewal application are not limited to those that have been previously reviewed and approved by the NRC. The following examples illustrate TLAAs that may need to be addressed but which were not previously reviewed and approved by the NRC:

- The FSAR states that the design complies with a certain national code and standard. A review of the code and standard reveals that it calls for an analysis or calculation. Some of these calculations or analyses will be TLAAs. The applicant performed the actual calculation to meet the code and standard, but the specific calculation was not referenced in the FSAR and the NRC had not reviewed the calculation. Also, some of these TLAAs may not have been relevant for the original licensing period but may be significant for the period of extended operation.
- In response to a NRC generic letter, a licensee submitted a letter to the NRC committing to perform a TLAA that would address the concern in the generic letter. The NRC had not documented a review of the applicant's response and had not reviewed the actual analysis.

The following examples illustrate potential TLAAs that are not TLAAs and need not be addressed:

- Analyses with a time-limited assumption less than the current license period of the ISFSI: For example, an analysis for a component based on a service life that would not reach the end of the current license period.
- Analyses that do not involve aging effects. For example, wind speed of 100 mph is expected to occur once every 50 years.

The number and type of TLAAs vary depending on the site-specific design basis for the ISFSI or DCSS. All five criteria described in Subsection III.1.2 must be satisfied to conclude that a calculation or analysis is a TLAA. Table III.1 provides examples of how the five criteria may be applied. Table III.2 provides a list of generic TLAAs that may be included in a license application. Table III.3 provides a list of other potential site-specific TLAAs. It is not expected that all license renewal applications would identify all TLAAs in these tables for their facilities. Also, an applicant may perform specific TLAAs for its facility that are not shown in these tables.

Sections III.2 to III.6 describe typical TLAAs for managing aging effects in used-fuel dry storage facilities. Section III.7 describes other site-specific TLAAs.

Table III.1 Sample Process for Identifying Potential TLAAs and Basis for Disposition.

Example	Disposition
Maximum wind speed of 100 mph is expected to occur once per 50 years.	Not a TLAA because it does not involve an aging effect.
The applicant states that the spacer plate welded to the gamma shielding cross plate in the air inlet of the HI-STORM storage system is certified by the vendor to last for 40 years.	This component was not credited in any safety evaluation, and therefore the analysis is not considered a TLAA. It does not meet criterion (4) of the TLAA definition in Subsection III.1.2.
Fatigue analyses for the used-fuel canister shell, performed in accordance with the criteria in ASME Section III, NB-3222.4, showed that no consideration of fatigue is required for the 50-year service life.	This is a TLAA because it meets all five criteria defined in Subsection III.1.2. The applicant's fatigue design basis relies on assumptions defined by the 50-year service life for this component.
The integrated fluence is estimated for the 60-year service life of the shielding material in horizontal storage module doors.	This is a TLAA because it meets all five criteria defined in Subsection III.1.2. The design basis for the use of the shielding material is currently limited to 60 years, and needs to be reanalyzed for the period of extended operation beyond 60 years.

Table III.2 Generic TLAAs

Fatigue of Metal and Concrete Structures and Components (Subsection III.2)
Corrosion Analysis of Metal Components (Subsection III.3)

Table III.3 Examples of Potential Site-Specific TLAAs

Flaw growth analyses that demonstrate structure stability for extended service.
Strength reduction of concrete structures and components.
Time-dependent degradation of neutron absorbing materials (Subsection III.4)
Time-dependent degradation of radiation shielding materials (Subsection III.5)
Environmental Qualification of Electrical Equipment (Subsection III.6)

III.1.4 References

NUREG-1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance—Final Report, U.S. Nuclear Regulatory Commission, Washington, DC, March 2011.

III.2 Fatigue of Metal and Concrete Structures and Components

III.2.1 Description of the Time-Limited Aging Analysis

Metal and concrete structures and components in ISFSIs and DCSSs are subject to degradation and failure due to fatigue under cyclic loading conditions, such as may occur under temperature and/or pressure cycling or vibrational loading. Such failures can occur at stress amplitudes significantly below the design static loads. Fatigue in metals typically occurs through a process of crack initiation and subsequent growth through the thickness of the affected component. Plain concrete, when subject to repeated loads, may exhibit excessive cracking and may eventually fail after a sufficient number of cycles at load levels less than the static strength of the material. The fatigue analysis of ISFSI and DCSS casks and canisters is covered in ASME Boiler and Pressure Vessel Code Section III, Division 1, Subsection NB; guidance on the design of concrete structures and components is given in the ACI Committee Report ACI 215R-74; and the fatigue of other steel support structures is covered under the AISC Standards ANSI/AISC N690-06 and ANSI/AISC N360-10, Appendix 3. The ASME is currently preparing a Section III, Division 3 Code for dry cask transportation and storage systems that includes Subsections WA-general, WB-transportation casks, WC-storage casks, and WD-cask internals.

The ASME Code Section III, Division 1, Subsection NB-3200 requires a fatigue analysis for all Class 1 components unless exempted by the Code under applicable Section III provisions. This analysis considers all cyclic loads on the basis of the anticipated number of loading cycles, and includes calculation of the parameter “cumulative usage factor” (CUF), which is used for estimating the extent of fatigue damage in the component. The ASME Code limits the CUF to a value of less than or equal to one for acceptable fatigue design. A CUF below a value of one provides reasonable assurance that no crack has been formed. A CUF greater than one allows for the possibility that a crack may form, and that if left unmitigated, the crack could propagate under fatigue loading and eventually result in component leakage or structural failure. In cases where fatigue of metallic components has been evaluated on the basis of an assumed number of load cycles, the validity of this analysis must be reviewed for the period of extended operation.

The fatigue strength of concrete is defined as a fraction of the static strength that it can support repeatedly for a given number of cycles. The fatigue stress vs. cycles (S-N) curves for concrete represent average behavior (i.e., 50% probability of failure), and are approximately linear between 10^2 and 10^7 cycles. However, the design curve may be based on a lower probability of failure. These curves indicate that concrete does not exhibit an endurance limit up to 10 million cycles. For a life of 10 million cycles of compressive, tensile, or flexure loading, the fatigue strength of concrete is about 55% of its static strength. The ACI Committee Report ACI 215R-74 provides background information and general guidance on the design of concrete structures and components for fatigue. The information is presented in the form of diagrams and algebraic relationships that can be used for design. Typically, for a zero minimum stress level (i.e., a load ratio R = 0), the maximum stress level the concrete can support for one million cycles without failure is taken conservatively as 50% of the static load. The maximum allowable stress increases with increasing load ratio. The effects of different values of maximum stress are estimated from constant-stress fatigue tests using Miner’s Rule (i.e., $\sum(n_r/N_r) = 1$, where n_r is the number of applied stress cycles and N_r is the number of cycles that will cause failure at that same stress). Also, the effects of loading rate and hold periods have little effect on fatigue strength.

The AISC Standard ANSI/AISC N690-06 addresses the design, fabrication, and erection of safety-related steel structures for nuclear facilities. This standard is an extension of ANSI/AISC N360-10, which addresses the same topics for structural steel buildings in general. In particular, the guidance for fatigue design in ANSI/AISC N690-06 refers directly to Appendix 3, “Design for Fatigue,” of ANSI/AISC N360-10. This appendix specifically applies to structural steel members and connections subject to high-cycle fatigue stresses within the elastic range but of sufficient magnitude to initiate potential cracking and progressive fatigue failure. Guidance is provided on calculating the maximum allowable stress range under cyclic loading conditions for steel structural elements away from and adjacent to welds, mechanically fastened joints, and welded joints of various geometries. Fabrication guidelines for reducing the susceptibility of fabricated steel structures to fatigue are given in the accompanying Commentary on the Specification for Structural Steel Buildings, Appendix 3 of N360-10.

To ensure that fatigue or flaw growth/tolerance evaluations are valid for the period of extended long-term storage, the fatigue analyses should include the following:

1. CUF calculations for ASME Code Class 1 components designed to ASME Section III requirements or to other codes that are based on a CUF calculation.
2. Maximum stress range values and associated numbers of loading cycles, as well as fabrication procedures and techniques employed to reduce susceptibility to fatigue failure for concrete components designed in accordance with the general guidance given in ACI 215R-74.
3. Maximum stress range values and associated numbers of loading cycles, as well as fabrication procedures and techniques employed to reduce susceptibility to fatigue failure for other steel support structures designed in conformance with Appendix 3 of ANSI/AISC N360-10.

III.2.2 Dispositioning the Time-Limited Aging Analysis

The acceptance criteria for the TLAAs associated with fatigue of metal and concrete structures and components should delineate acceptable methods by following the NRC’s guidelines stated in NUREG-1927, Section 3.5.1(5) and listed in Subsection III.1.2.

III.2.2.1 ASME Section III Design

For components designed or analyzed to ASME Code Section III requirements for Class I components or other codes that require a CUF calculation, the acceptance criteria depend upon the choice of the criteria in NUREG-1927 Section 3.5.1(5)(i) or (ii). For option (i), the analyses are projected to the end of the period of extended long-term storage. This is achieved by either (a) demonstrating that the existing CUF calculations would remain valid for the period of extended long-term storage because the actual number of accumulated cycles would not exceed the design basis cycles used in the original analysis, or (b) projecting the CUF calculations to the end of the period of extended long-term storage on the basis of projecting the cumulative number of design basis cycles through to the expiration of the period of extended long-term storage, and ensuring the resultant CUF values do not exceed the design limit of 1.0.

For option (ii), a site-specific AMP should be developed to ensure that the effects of cumulative fatigue damage on intended safety functions of the SSCs will be adequately managed during the period of extended long-term storage. For example, monitoring and tracking the number of fatigue cycles for the fatigue-sensitive locations and components is an acceptable AMP. Such a program has been developed for the license renewal of nuclear power plants and is described in NUREG-1801, Rev. 2, Section X.M1 ("Fatigue Monitoring"). If future inspections or examinations are to be used to manage potential effects of cumulative fatigue damage evaluated in the TLAAs, then the adequacy of the program must be demonstrated, particularly by identifying the fatigue-sensitive locations and their accessibility for inspection, and demonstrating the adequacy of the inspection interval. Furthermore, such proposed future actions may need to be included as conditions for license renewal. Alternatively, as stated in NUREG-1927, Section 3.5, the component can be replaced and the allowable stresses for the replacement will be as specified by the code during the period of extended long-term storage.

III.2.2.2 Design of Concrete Structures—ACI-215

For concrete structures and components designed in accordance with the guidance provided in ACI 215R-74, one may either choose option (i) and project the analyses to the end of the period of extended long-term storage, or choose option (ii) and develop a site-specific AMP to ensure that the effects of cumulative fatigue damage on the intended safety functions of the concrete structures will be adequately managed during the period of extended long-term storage. Under option (i), one needs to demonstrate that (a) the current fatigue TLAA remains valid because the severity (i.e., maximum stress) and the number of design cycles would not be exceeded during the extended period of long-term storage, or (b) if either one of these is exceeded, the new set of maximum stress values and the corresponding allowable number of cycles are reevaluated to ensure that the design basis value remains acceptable during the period of extended long-term storage.

Under option (ii), one needs to (a) develop an AMP that monitors and tracks the severity and number of load cycles to ensure that they remain below the design limit during the period of extended long-term storage, or (b) if the AMP includes future inspections or examinations to manage potential effects of fatigue damage, the adequacy of the program must be demonstrated. For the latter option, one should identify the fatigue-sensitive locations and their accessibility for inspection, and demonstrate the adequacy of the inspection interval.

III.2.2.3 ANSI/AISC 690 and ANSI/AISC 360 Design

For other steel support structures designed to ANSI/AISC N690-06 and ANSI/AISC N360-10 Appendix 3 standards, since the analyses in these standards apply only to high-cycle fatigue loading stresses within the elastic range, under option (i) one needs to demonstrate that the structures in question (a) have not been and will not be subjected to loading in excess of the elastic limit during the period of extended long-term storage, and (b) have not been subject to aging degradation processes, such as loss of material due to corrosion and wear or SCC, that could degrade their structural integrity. If either of these recommendations cannot be met, a site-specific AMP for managing this fatigue TLAA should be developed.

III.2.3 Final Safety Analysis Report Supplement

Information should be included in the FSAR supplement that provides a summary description of the evaluation of the TLAA on fatigue of metal and concrete structures and components. Additional information is given in Subsection III.1.3.

III.2.4 References

ACI 215R-74, Considerations for Design of Concrete Structures Subjected to Fatigue Loading, American Concrete Institute, Farmington Hills, MI, December 1992.

ANSI/AISC N360-10, Specification for Structural Steel Buildings, American Institute of Steel Construction, Chicago, IL, June 22, 2010.

ANSI/AISC N690-06, Specification for Safety-Related Steel Structures for Nuclear Facilities, American Institute of Steel Construction, Chicago, IL, 2007.

ASME Boiler and Pressure Vessel Code, Section III, Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NB, Class 1 Components, American Society of Mechanical Engineers, New York, 2004.

NUREG-1800, Standard Review Plan for License Renewal Applications for Nuclear Power Plants, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, December 2010.

NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, December 2010.

NUREG-1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance—Final Report, U.S. Nuclear Regulatory Commission, Washington, DC, March 2011.

III.3 Corrosion Analysis of Metal Components

III.3.1 Description of the Time-Limited Aging Analysis

Table III.2 of this document lists corrosion analysis of metal components (or metal corrosion allowance) as a generic TLAA. The SRP for the design of used-fuel dry storage facilities (NUREG-1567) also refers to the provision of appropriate corrosion allowances for materials susceptible to corrosion. Accordingly, the loss of material due to general corrosion is treated here as a time-dependent aging effect requiring management for low-alloy steel, carbon steel, and cast iron components in outdoor environments. Examples of such components include anchoring dowels between concrete storage modules and concrete pads, anchors in concrete walls to support canisters inside storage modules, steel liners, and carbon steel heat-shielding plates. For such components and structures that are subject to loss of material due to general corrosion in outdoor or uncontrolled indoor environments, the applicant for a license renewal must ensure that these corrosion analyses are valid for the period of extended operation.

III.3.2 Dispositioning the Time-Limited Aging Analysis

The acceptance criteria for the TLAAs associated with corrosion analysis of metal components should delineate acceptable methods by following the NRC's guidelines stated in NUREG-1927, Section 3.5.1(5) and listed in Subsection III.1.2.

III.3.2.1 Corrosion Allowances

NUREG-1567 states in Section 5.4.1.3 that, in the design criteria for confinement structures in used-fuel dry storage facilities, "appropriate corrosion allowances should be established and used in the structural analyses." In addition, Section 5.5.1.3 of NUREG-1567 states that for confinement SSCs, the applicant should evaluate the potential for corrosion to ensure that adequate corrosion allowances for materials susceptible to corrosion have been provided in these analyses. These same considerations carry over to the evaluation of applications for license extension for such facilities. In order to satisfy Section 3.5.1(5) criterion (i) of NUREG-1927, the applicant must demonstrate that the corrosion allowance for the SSC being evaluated is sufficient to accommodate the anticipated loss of material due to general corrosion projected through the end of the period of extended operation.

III.3.2.2 Corrosion Effects Management

The management of corrosion effects, as described in Section 3.5.1(5) criterion (ii) of NUREG-1927, may be used for managing this TLAA for SSCs where insufficient corrosion allowance is available to satisfy criterion (i). The most direct approach to such corrosion management is replacement of the SSC in question, but the applicant may propose alternative site-specific approaches such as inspection or surveillance of corrosion management to satisfy criterion (ii). The applicant may propose periodic surface and volumetric inspections of those SSCs subject to loss of material due to corrosion during the period of extended operation using techniques and procedures similar to those described in the Welded Canister Seal and Leakage Monitoring AMP IV.M3 in Chapter IV of this report. Such inspections are subject to the general requirements of the ASME Boiler and Pressure

Vessel Code, Section XI, Subsections IWB-1100, IWC-1100, and IWD-1100 for Class 1, 2, and 3 components, respectively.

NUREG-1927 Appendix E Component-Specific Aging Management describes “Lead Canister” external remote visual inspection and Horizontal Storage Module (HSM) canister support steel. A lead canister is selected on the basis of longest time in service, or hottest thermal load, and/or other parameters that contribute to degradation. A similar methodology is acceptable for casks that perform a confinement function. The interior of the associated concrete overpack or HSM should also be examined as part of the lead canister inspection. This inspection is performed before submittal of the license renewal application and the inspection results become part of the justification for license renewal. Typically, a repeat inspection is conducted at 20-year intervals as a license condition for renewal. The licensee/certificate holder may propose alternative inspection intervals for NRC approval.

NUREG-1927 Appendix E also states that the inspections are especially pertinent for ISFSIs located at coastal marine sites where atmospheric corrosion is known to be more severe. Support structure inspection may be done on a sampling basis. Selection of one or more support structures to be inspected should be based on longest service time, material, and/or environmental conditions. Normally, carbon steel is specified for this support structure. Some locations may have employed protective coatings on the support structure. Other ISFSI locations may have employed 0.2% copper-bearing steel. Differences in materials and environmental conditions at various sites could make comparisons between different ISFSI sites invalid. The licensee should specify the re-inspection interval for the support structure based upon the findings of the initial license renewal inspection.

Some components, such as anchoring dowels between concrete storage modules and concrete pads, are inaccessible for inspection. In this situation, a site-specific AMP will be required if the TLAA cannot demonstrate that failure is not expected.

III.3.3 Final Safety Analysis Report Supplement

Information should to be included in the FSAR supplement that provides a summary description of the evaluation of the TLAA on corrosion analysis of metal components. Additional information is given in Subsection III.1.3.

III.3.4 References

ASME Section XI, Rules for Inservice Inspection of Nuclear Power Plant Components, ASME Boiler and Pressure Vessel Code, 2004 edition as approved in 10 CFR 50.55a, American Society of Mechanical Engineers, New York, 2004.

NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities, U.S. Nuclear Regulatory Commission, Washington, DC, March 2000.

NUREG-1800, Standard Review Plan for Review of License Renewal Applications for Nuclear Power Plants, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, December 2010.

NUREG-1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance—Final Report, U.S. Nuclear Regulatory Commission, Washington, DC, March 2011.

III.4 Time-Dependent Degradation of Neutron-Absorbing Materials

III.4.1 Description of the Time-Limited Aging Analysis

Dry cask storage systems commonly incorporate neutron-absorbing materials into their design to provide neutron shielding and, during the short period of time when the cask is flooded with water during fuel loading and unloading, to provide criticality control. Once the cask has been drained, dried, and inerted, the absence of the moderating effect of the hydrogen atoms in the water renders the fuel subcritical by a substantial margin. The specific neutron absorber most commonly used in neutron-absorbing materials is boron in a chemically stable form such as B_4C, or less commonly AlB_2, TiB_2, or CrB_2. The isotope B-10, which comprises approximately 20% of naturally occurring B, has a very large thermal neutron absorption cross section. The B_4C is incorporated into a suitable matrix, which may be metallic, polymeric, or cementitious, to provide the mechanical, physical, and fabrication characteristics required during use.

A distinction is made between neutron-absorbing materials, which are covered under this TLAA, and gamma and neutron radiation-shielding materials, which are covered under TLAA III.5 (described in the next section). For the purposes of this distinction, the “neutron-absorbing materials” discussed here are those materials that are positioned in and immediately around the fuel basket inside the canister for the primary purpose of criticality control. These materials incorporate boron (or, less commonly, cadmium or gadolinium) in some chemical form as a neutron absorber or neutron “poison.” Some of these same materials, as well as polymer resins, polyethylene, and other low-Z materials that do not contain neutron poisons, are also positioned outside the canister to attenuate or absorb neutrons, primarily for biological shielding. Even though some of the materials are the same, the operating environment and functional requirements are different, and the biological shielding application is treated in the subsequent TLAA III.5 on radiation-shielding material.

Specific examples of commercial neutron-absorbing materials, some of which are used in DCSSs, are listed below. All use B_4C as the neutron absorber, except as indicated:

<u>Aluminum alloy/boron carbide metal matrix composites</u>
Metamic®, BORTEC®

<u>Aluminum alloy/boron carbide cermets</u>
BORAL® (Al clad)

<u>Borated aluminum alloys</u>
BorAluminum® (AlB_2/TiB_2 absorber)

<u>Borated Stainless Steels</u>
NeutroSorb® Plus, Neutronit®

<u>Silicone rubber/boron carbide composites</u>
BISCO NS-1®, Boro-Silicon®, Bocarsil®

<u>Borated phenolic resin compounds</u>
BISCO NS-4-FR®, Holtite-A®, Carborundum B_4C

<u>Borated concrete</u>
BISCO NS-3®

III.4.1.1 Degradation of Neutron-Absorbing Materials

Neutron-absorbing materials in spent-fuel storage pools have been found to be subject to several forms of time-dependent degradation. Similar degradation of these materials in DCSSs has not been reported to date, but service times in DCSS applications have not been as long as in spent-fuel storage pools. In addition, unlike spent-fuel storage pools, the neutron-absorbing materials in DCSSs are not accessible for periodic examination. Examples of the degradation of specific neutron-absorbing materials in spent-fuel storage pools, as well as some potential degradation modes that have not been observed in service, are summarized below.

III.4.1.1.1 BORAL®

BORAL has been found to be subject to degradation due to localized corrosion of and blister formation in the Al alloy cladding. In a BWR pool environment, localized corrosion can occur at weak spots in the passivation oxide film. In a borated PWR pool environment, localized corrosion can occur at sites of surface imperfections and/or residual surface contaminants left from the manufacturing process. This localized corrosion can take the form of pitting, crevice corrosion, galvanic corrosion, intergranular corrosion, or exfoliation. Some of the most extensive localized corrosion has been observed in PWR surveillance coupons clad in stainless steel capsules (EPRI 1019110). It should be noted that, to date, no instances of decrease in B-10 areal density in BORAL neutron absorbers in spent-fuel storage pool service have been observed as a result of localized corrosion of the cladding. In addition, localized corrosion processes appear to be much less likely in the dry, inert conditions that are present in DCSS service except during loading and unloading. However, localized corrosion cannot be ruled out without additional site-specific information concerning DCSS design parameters and operating conditions.

Blistering of the BORAL Al cladding was first observed at Yankee Rowe in 1964. Blisters have been observed in both surveillance coupons and spent-fuel storage racks containing BORAL, including more recent occurrences at Beaver Valley and Susquehanna (NRC IN 2009-26). The blisters are characterized by a local area where the Al cladding separates from the underlying B_4C-Al composite, and the cladding is physically deformed outward. While this blistering has not been observed to alter the neutron-absorption properties of the material, it does lead to concerns about fuel removal in storage pools and potential increases in the reactivity state of the fuel/rack configuration due to geometry changes (EPRI 1019110).

In 1998, Vogtle installed additional spent-fuel racks in the Unit 1 pool that used BORAL as the neutron absorber. Aluminum concentrations in the spent-fuel pool water have increased since the introduction of these racks, but the resulting concentrations have not resulted in any significant problems. Nonetheless, this situation bears continued monitoring.

For dry fuel storage, the NRC has noted that water can penetrate into the porous BORAL neutron-absorber inner layer during fuel loading and may lead to blistering of the Al cladding, physical degradation (“crumbling”) of the underlying composite material, or deformation (“relocation”) of the BORAL during the elevated-temperature storage period. In addition, physical damage to the BORAL is considered possible when the cask is quenched by reflooding during subsequent fuel unloading. This leads to concerns of inadvertent criticality during unloading because of the re-introduction of water, which serves as a neutron moderator. Maintaining the integrity of the BORAL neutron-absorber material throughout the life of the cask is therefore important. As possible solutions, the NRC has proposed testing of BORAL under simulated conditions to determine if crumbling or relocation occurs, and, if it does, to repair the susceptible casks under dry conditions or introduce water with a soluble neutron absorber during unloading (*Sindelar 2011*).

III.4.1.1.2 Carborundum Borated Phenolic Resin Plates

In July 2008, Palisades discovered that the spent-fuel pool storage racks contained less neutron-absorbing material than assumed in their criticality analysis and that they were in noncompliance with the applicable technical specification. An inspection of selected fuel racks revealed that the B-10 areal density was, at a minimum, approximately one-third of its original design value. The exact degradation mechanism or mechanisms are not clearly understood but likely involve changes in the physical properties of the Carborundum B_4C plates that occur during prolonged exposure to the spent-fuel pool environment. The degradation manifests itself in the form of absorber plate swelling and deformation and loss of B-10 areal density (NRC IN 2009-26). Other neutron-absorbing materials utilizing a phenolic resin base should be carefully evaluated for similar aging-related degradation, on the basis of the experience described here.

III.4.1.1.3 Aluminum Matrix Absorber Materials

Metallic materials are generally considered to be subject to creep under conditions of extended exposure to temperature in excess of a homologous temperature of $0.4T_m$. For aluminum alloys, this translates to a temperature of approximately 100°C. Extended times at temperatures in excess of this value are anticipated for many cask designs, and the potential for creep deformation of Al-alloy and Al-matrix neutron-absorber materials must be considered. Such deformation and loss of dimensional stability could adversely impact both criticality control and fuel retrievability. No operating experience on the exposure of these materials to temperatures in excess of 100°C for tens of years is available, and the variability in basket and cask designs requires that any such creep analysis be performed on a design-specific basis.

The potential long-term oxidation of Al-alloy and Al-matrix neutron-absorber materials must also be considered. Limited corrosion associated with residual moisture within the canister as a result of incomplete drying or waterlogged fuel rods may occur immediately after fuel loading. This residual water would be consumed very early in the storage period through corrosion processes, and the corrosion would cease. However, air ingress into a breached canister could lead to continued oxidation of the basket and neutron-absorber materials. This could result in a loss of structural strength and dimensional stability. Again, the variability in basket and cask designs requires that the analysis of any long-term oxidation effects be performed on a design-specific basis.

III.4.2 Dispositioning the Time-Limited Aging Analysis

The aging-related degradation of neutron-absorbing materials in the spent-fuel racks of nuclear power plants is treated by two AMPs in NUREG-1801, Rev. 2, namely, XI.M22 (“Boraflex Monitoring”) and XI.M40 (“Monitoring of Neutron-Absorbing Materials Other than Boraflex”). Both of these are condition-monitoring programs that include no preventive actions. However, the neutron-absorbing material in a dry cask storage module is not accessible for direct condition monitoring. Thus, an AMP based on condition monitoring is not appropriate for managing possible aging effects. For this reason, NUREG 1927 cites depletion of neutron-absorber material as an example of a situation requiring a TLAA. Because of the variety of DCSSs in use and the wide range of associated operating parameters and conditions, no universal guidelines can be provided for meeting the NRC’s TLAA review guidance stated in NUREG-1927, Section 3.5.1(5). Therefore, this

TLAA should be performed on a site-specific basis using a suitable methodology that conforms to the general guidance summarized below.

III.4.2.1 10 CFR 72.124(b) Requirements

All applicable requirements in 10 CFR Part 72.124(b) for methods of criticality control for the licensing of an ISFSI also apply to license renewal for that facility, as well as for monitored retrievable storage (MRS). Specifically, 10 CFR 72.124(b) states the following:

> Methods of criticality control. When practicable, the design of an ISFSI or MRS must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design must provide for positive means of verifying their continued efficacy. For dry spent fuel storage systems, the continued efficacy may be confirmed by a demonstration or analysis before use, showing that significant degradation of the neutron absorbing materials cannot occur over the life of the facility.

As stated above, the demonstration or analysis that significant degradation of the neutron-absorbing materials cannot occur over the extended period of operation of the facility must be performed on a site-specific basis.

III.4.2.2 NRC ISG-23

ASTM Standard Practice C1671-07 addresses issues related to this TLAA. NRC ISG-23 provides guidance on the application of this standard practice when performing technical reviews of spent-fuel storage and transportation packaging licensing actions. This guidance is also applicable to the license renewal process for DCSSs.

ISG-23 basically states that ASTM Standard Practice C1671-07 provides acceptable guidance for performing technical reviews of spent-fuel storage and transportation packaging licensing actions with the additions, clarifications and exceptions noted. With respect to the time-dependent corrosion and elevated-temperature degradation of neutron-absorbing materials, ISG-23 states the following:

> <u>Clarification regarding use of Section 5.2.1.3 of ASTM C1671-07.</u> If the supplier has shown that process changes do not cause changes in the density, open porosity, composition, surface finish, or cladding (if applicable) of the neutron absorber material, the supplier should not need to re-qualify the material with regard to thermal properties or resistance to degradation by corrosion and elevated temperatures.

III.4.2.3 NRC ISG-15

NRC ISG-15, Section X.5.2.7 provides guidance and review procedures for neutron-absorbing/poison materials for control of criticality. This guidance states the following:

> For all boron-containing materials, the reviewer should verify that the SAR and its supporting documentation describe the chemical composition, physical and mechanical properties, fabrication process, and minimum poison content of the material. This description should be detailed enough to verify the adequacy and reproducibility of

properties important to performance as required in the SAR. For plates, the minimum poison content should be specified as an areal density (e.g., milligrams of ^{10}B per cm^2). For rods, the mass per unit length should be specified.

In heterogeneous absorber materials, the neutron poisons may take the form of particles dispersed or precipitated in a matrix material. Materials with large poison particles (e.g., 80-micrometer particles of unenriched boron carbide) have been shown to absorb significantly fewer neutrons than homogeneous materials with the same poison loading. The reduced neutron absorption in heterogeneous materials results from particle self-shielding effects, streaming and channeling of neutrons between poison particles. Therefore, the reviewer should verify that the absorber material's heterogeneity parameters (e.g., particle composition, size, dispersion) are adequately characterized and controlled, and that the criticality calculations employ appropriate corrections (e.g., reduced poison content) when modeling the heterogeneous material as an idealized homogeneous mixture.

III.4.3 Final Safety Analysis Report Supplement

Information should be included in the FSAR supplement that provides a summary description of the evaluation of the TLAA associated with the degradation of neutron-absorbing materials. Additional information is given in Section III.1.3.

III.4.4 References

10 CFR 72.124(b), Criteria for Nuclear Criticality Safety, Methods of Criticality Control, Office of the Federal Register, National Archives and Records Administration, 1999.

ASTM C1671-07, Standard Practice for Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage Systems and Transportation Packaging, American Society for Testing and Materials, West Conshohocken, PA, 2007.

EPRI 1019110, Handbook of Neutron Absorber Materials for Spent Nuclear Fuel Transportation and Storage Applications—2009 Edition, Electric Power Research Institute, Palo Alto, CA, November 2009.

EPRI 1021048, Industry Spent Fuel Storage Handbook, Electric Power Research Institute, Palo Alto, CA, July 2010.

NRC Interim Staff Guidance 15, Materials Evaluation, U.S. Nuclear Regulatory Commission, Washington, DC, January 2001.

NRC Interim Staff Guidance 23, Application of ASTM Standard Practice C1671-07 When Performing Technical Reviews of Spent Fuel Storage and Transportation Packaging Licensing Actions, U.S. Nuclear Regulatory Commission, Washington, DC, January 2011.

NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, December 2010.

Sindelar, R. L., et al., Materials Aging Issues and Aging Management for Extended Storage and Transportation of Spent Nuclear Fuel, NUREG/CR-7116, Savannah River National Laboratory, Aiken, SC, November 2011.

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III.5 Time-Dependent Degradation of Radiation-Shielding Materials

III.5.1 Description of the Time-Limited Aging Analysis

DCSSs commonly incorporate radiation (from neutron and gamma)-shielding materials into their design to provide radiation protection. Concrete, steel, depleted uranium, and lead typically serve as gamma shields, while hydrogenous materials such as polymer resins and polyethylene, as well as other low-Z materials, are often used for neutron shielding. These shielding materials may be subjected to time-dependent degradation due to various factors.

As stated in TLAA III.4, materials containing boron or some other neutron poison, which are generally installed inside the spent-fuel canister as neutron absorbers, are also sometimes used for neutron shielding outside the canister. Because the operating environment and functional requirements are different for the two applications, the outside-the-canister installation of these materials is considered here rather than in the previous TLAA.

A TLAA may consist of calculation of a bounding dose based on a bounding dose rate of the source term or a bounding acceptable dose of the SSCs under consideration over the extended period of operation, supplemented in some cases (e.g., in reasonably accessible locations/environments) with periodic inspection/monitoring. NRC ISG-15 provides guidance on gamma- and neutron-shielding materials and their possible degradation processes.

III.5.1.1 Degradation of Radiation-Shielding Materials

Radiation-shielding materials in DCSSs and transportation packages may be subject to several forms of time-dependent degradation. Examples of the degradation of specific radiation-shielding materials in DSCC environments are provided in ISG-15 and summarized below.

The properties and performance of these shielding materials are temperature-sensitive and one must ensure that these shielding materials will not be subject to temperatures at or above their design limits during either normal or accident conditions. The potential for shielding materials to experience changes in material densities at temperature extremes at some point in time during their usage needs to be taken into account. Higher temperatures may reduce hydrogen content through loss of water in concrete or other hydrogenous shielding materials.

For externally deployed polymer-based neutron-shielding materials, the thermal stability of the resin over the design life at the higher end of the design operating temperature regime needs to be verified. Reasonable assurance may be provided through testing programs. Polymers generally have a relatively large coefficient of thermal expansion when compared to metals. Therefore, the neutron shield design needs to include elements to ensure that excessive neutron streaming will not occur as a result of shrinkage under conditions of extreme cold. Any changes or substitutions made to the shield material formulation would require new testing on the differently formulated materials' neutron absorption, thermal stability, and handling properties during mixing and pouring or casting. It also should be verified that any filled channels used on production casks did not have significant voids or defects that could lead to greater than calculated dose rates.

III.5.2 Dispositioning the Time-Limited Aging Analysis

NUREG-1927, Section 3.5.1, provides guidance and criteria for the review of TLAAs contained in applications for ISFSI license renewal. However, because of the variety of DCSSs in use and the wide range of associated operating parameters and conditions, no universal guidelines can be provided for meeting the guidance stated in NUREG-1927, Section 3.5.1(5) or the acceptance criteria in Subsection III.1.1.1. Therefore, this TLAA should be performed on a site-specific basis using a suitable methodology that conforms to the general guidance summarized below.

III.5.2.1 10 CFR 72.42(a)(1), 72.126(a), and 72.128(a) Requirements

All applicable requirements in 10 CFR Part 72.126, 128 for criteria for radiological protection and for storage and handling of an ISFSI also apply to license renewal for that facility. Specifically, 10 CFR 72.126(a) (6) states the following:

> *Exposure Control*. Radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested so as to control external and internal radiation exposures to personnel. The design must include means to… (6) Shield personnel from radiation exposure.

10 CFR 72.128(a)(2) states the following:

> *Spent fuel, high-level radioactive waste, reactor related greater than Class C waste, and other radioactive waste storage and handling systems.* Spent fuel storage, high-level radioactive waste storage, reactor-related GTCC waste storage and other systems that might contain or handle radioactive materials associated with spent fuel, high-level radioactive waste, or reactor-related GTCC waste, must be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with ….. (2) Suitable shielding for radioactive protection under normal and accident conditions,

The demonstration or analysis that significant degradation of the shielding capabilities of radiation-shielding materials cannot occur over the extended period of operation of the facility may be satisfied by a TLAA. This is in accordance with the requirements of 10 CFR 72.42 (Duration of License Renewal), which states that "Application for ISFSI license renewals must include the following: (1) TLAAs that demonstrate that structures, systems, and components important to safety will continue to perform their intended function for the requested period of extended operation; …. ."

III.5.2.2 NRC ISG-15

NRC ISG-15, Section X.5.2.6 provides guidance and review procedure for gamma and neutron shielding materials for radiation protection. The guidance states the following:

> Concrete, steel, depleted uranium, and lead typically serve as gamma shields, while filled polymers are often used for neutron shielding materials. The reviewer should confirm that temperature-sensitive shielding materials will not be subject to temperatures at or above

their design limits during both normal and accident conditions. The reviewer should determine whether the applicant properly examined the potential for shielding material to experience changes in material densities at temperature extremes. (For example, elevated temperatures may reduce hydrogen content through loss of water in concrete or other hydrogenous shielding materials.)

With respect to external, polymer neutron shields, the reviewer should verify that the application

- Describes the test(s) demonstrating the neutron absorbing ability of the shield material.
- Describes the testing program and provides data and evaluations that demonstrate the thermal stability of the resin over its design life while at the upper end of the design temperature range. It should also describe the nature of any temperature-induced degradation and its effect(s) on neutron shield performance.
- Describes what provisions exist in the neutron shield design to assure that excessive neutron streaming will not occur as a result of shrinkage under conditions of extreme cold. This description is required because polymers generally have a relatively large coefficient of thermal expansion when compared to metals.
- Describes any changes or substitutions made to the shield material formulation. For such changes, describes how they were tested and how that data correlated with the original test data regarding neutron absorption, thermal stability, and handling properties during mixing and pouring or casting.
- Describes the acceptance tests that were conducted to verify that any filled channels used on production casks did not have significant voids or defects that could lead to greater than calculated dose rates.

III.5.2.3 Oconee Site--specific ISFSI License Renewal

The Oconee's site-specific ISFSI license renewal was granted by NRC in May 2009 for a total period of 40 years, 20 years for the renewal license term and an additional 20 years under an exemption request per 10 CFR 72.42 before 2011. The NRC Safety Evaluation Report for the license renewal application states the following:

> The ONS Site-Specific ISFSI uses the NUHOMS®-24P horizontal storage model design. Each HSM contains one dry storage canister (DSC), and each DSC contains 24 irradiated fuel assemblies (IFAs). This design employs a stainless steel, all-welded, DSC that is placed horizontally into a concrete shielding structure called the HSM. The HSM functions as the primary radiation shield, ...

Two SSCs are relevant for consideration of time-dependent degradation of radiation-shielding materials. They are the HSMs and the transfer cask. The neutron-shielding materials in the HSM doors are BISCO NS-3® and concrete. The gamma energy flux deposited in the HSM concrete is 6.8 x 10^{10} MeV/cm^2-sec. The accumulated fluence for the HSM is 1.44 x 10^{14} neutron/cm^2 for 60 years.

The licensee did not identify any aging effects on BISCO NS-3® shielding material in the HSM doors that require management during the renewal license term, since the material is fully encapsulated and exposure temperature would not cause degradation. However, degradation due to radiation exposure was considered to be manageable by a TLAA program through the renewal license term. The licensee did not provide any information on changes to the shielding capability of concrete material density at temperature extremes. However, the licensee’s monthly dose rate measurement of the exterior of HSMs provides a trending of any adverse condition that can be addressed by Duke’s Problem Investigation Process. The assumptions in the analyses are conservative, given that the licensee assumed a constant dose rate. The actual dose rate will decrease significantly during the 60-year service life.

The transfer cask provides radiological shielding during the DSC drying operation and during the transfer to the HSM. Since the fuel canister does not contain sufficient shielding materials by itself, a separate, heavily shielded transfer cask is used to transport the loaded fuel canisters from the loading building to the HSM.

The SER describes the BISCO NS-3® and concrete radiation exposure TLAA as follows:

> The materials that provide radiation shielding during the service life of ONS Site-Specific ISFSI are BISCO NS-3® and the HSM concrete. Concrete typically serves as gamma and neutron shielding, while the polymers are usually only used for neutron shielding. BISCO NS-3® neutron shielding material is used at phase 1 HSM doors and between the cask outer shell and the neutron shield jacket of the transfer cask.
>
> HSM Doors**:** The licensee calculated the gamma dose rate to be 330 mrem/hr to the door cavity of the HSM. This results in an integrated gamma dose of approximately 1.8 x 10^5 Rads for a service life of 60 years. This is well below the service limit of 1.5 x 10^{10} Rads for the BISCO NS-3® material.
>
> Transfer Cask**:** The licensee estimated gamma and neutron dose rates at the inner surface of BISCO NS-3® in the cask are 250 mrem/hr and 959 mrem/hr, respectively. The licensee conservatively assumed that the transfer cask neutron shielding was exposed to the same neutron fluence as the HSM interior concrete surface. The integrated neutron fluence is 1.44 x 10^{14} neutrons/cm^2, and is less than the service limit for the BISCO NS-3® material for both fast and thermal neutron exposure.
>
> HSM Concrete**:** The integrated neutron fluence in the HSM concrete for 60 years is 1.44 x 10^{14} neutron/cm^2. This is below the service limits for the material for fast and thermal neutron exposure, 1.6 x 10^{17} neutron/cm^2 and 1.5 x 10^{19} neutron/cm^2, respectively. The staff finds that the assumptions in the analyses above are conservative because the licensee assumed a constant dose rate while the actual dose rate decreases significantly during the 60-year service life. Furthermore, the licensee is committed to perform monthly dose rate measurement of the exterior of the HSMs. The staff finds that the licensee's TLAAs provide reasonable assurance that the BISCO NS-3® and HSM concrete materials will perform their intended function for the term of the license renewal period, require no further action and meet the requirements for license renewal.

III.5.3 Final Safety Analysis Report Supplement

Information should be included in the FSAR supplement that provides a summary description of the evaluation of the TLAA associated with the radiation shielding material. Additional information is given in Subsection III.1.3.

III.5.4 References

10 CFR 72.42(1), Duration of License; Renewal, Office of the Federal Register, National Archives and Records Administration, 2001.

10 CFR 72.126(a), Criteria for Radiological Protection, Office of the Federal Register, National Archives and Records Administration, 2001.

10 CFR 72.128(a), Criteria for Spent Fuel, High-Level Radioactive Waste, and Other Radioactive Waste Storage and Handling, Office of the Federal Register, National Archives and Records Administration, 2001.

NRC Interim Staff Guidance 15, Materials Evaluation, U.S. Nuclear Regulatory Commission, Washington, DC, January 2001.

NUREG-1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance—Final Report, U.S. Nuclear Regulatory Commission, Washington, DC, March 2011

Safety Evaluation Report, Docket No. 72-04, Duke Power Company, LLC Oconee Nuclear Station Independent Spent Fuel Storage Installation, License No. SNM-2503 License Renewal (ML 091520159), December 2010.

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III.6 Environmental Qualification of Electrical Equipment

III.6.1 Description of the Time-Limited Aging Analysis

Electrical equipment and components in an ISFSI or MRS installation are subject to degradation and failure by a variety of mechanisms as a result of extended service in the operating environment. These mechanisms include the following:

- Increased resistance of electrical connections due to thermal cycling, ohmic heating, electrical transients, vibration, chemical contamination, corrosion, and oxidation.
- Loss of conductor strength due to corrosion.
- Reduced insulation resistance due to moisture, salt deposits, and surface contamination.
- Reduced insulation resistance due to thermal/thermoxidative degradation of organics, radiolysis, and photolysis (UV sensitive materials only) of organics; radiation-induced oxidation; and/or moisture intrusion.
- Insulation surface cracking, crazing, scuffing, dimensional change, shrinkage, discoloration, hardening and loss of strength due to elastomer degradation.

The NRC has established environmental qualification (EQ) requirements for nuclear plant SSCs important to safety in 10 CFR Part 50, Appendix A, Criterion 4. In addition, 10 CFR 50.49 specifically requires environmental qualification for nuclear plant electrical equipment that is (1) safety related, (2) non-safety-related, but whose failure under postulated environmental conditions could prevent satisfactory accomplishment of safety functions, and (3) certain post-accident monitoring equipment. The NRC has also established, in 10 CFR part 72, monitoring requirements for an ISFSI or an MRS installation in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. All applicable EQ requirements in 10 CFR Part 50, Appendix A, Criterion 4, and 10 CFR 50.49 for nuclear stations (power plants) apply to monitoring requirements for an ISFSI/MRS as established in 10 CFR part 72.

10 CFR 50.49 specifically requires each nuclear power plant licensee to establish a program to qualify certain electric equipment (not including equipment located in mild environments) so that such equipment, in its end-of-life condition, will meet its performance specifications during and following design basis accidents under the most severe environmental conditions postulated at the equipment's location after such an accident. For ISFSI/MRSs, the most severe environmental conditions include, among others, loss-of-ventilation accidents and post-loss-of-ventilation-accident radiation and heat. The guidance in NUREG-1927, Section 2.4, and NUREG-1567, and the methodology of NUREG/CR-6407 can be followed in determining the classification of electrical equipment according to importance to safety. NUREG-1567 specifies the category of safety-related electrical equipment for spent-fuel dry storage facilities. Equipment qualified by test must be preconditioned by aging to its end-of-life condition (i.e., the condition at the end of the current operating term). Those components with a qualified life equal to or greater than the duration of the current operating term are covered by TLAAs.

III.6.2 Dispositioning the Time-Limited Aging Analysis

The acceptance criteria for the TLAA on environmental qualification of electrical equipment should delineate acceptable methods by following the NRC's guidelines stated in NUREG-1927, Section 3.5.1(5) and listed in Subsection III.1.2.

As stated above, guidance on the development of a program for the EQ of electrical equipment important to safety is provided in 10 CFR 50.49. Supplemental EQ regulatory guidance for compliance with these different qualification criteria is provided in NRC Regulatory Guide (RG) 1.89, Rev. 1, the Division of Operating Reactors (DOR) Guidelines, and NUREG-0588. The principal nuclear industry qualification standards for electric equipment are IEEE STD 323-1971 and IEEE STD 323-1974. These standards contain explicit EQ considerations based on TLAAs.

III.6.2.1 10 CFR 50.49 Requirements

All applicable requirements in 10 CFR Part 50.49 for important-to-safety electrical components in operating nuclear plants also applies to important-to-safety electrical components for an ISFSI/MRS/DCSS. 10 CFR 50.49 defines the scope of components to be included, requires the preparation and maintenance of a list of in-scope components, and requires the preparation and maintenance of a qualification file that includes component performance specifications, electrical characteristics, and environmental conditions. 10 CFR 50.49(e)(5) contains provisions for aging that require, in part, consideration of all significant types of aging degradation that can affect component functional capability. 10 CFR 50.49(e) also requires component replacement or refurbishment prior to the end of designated life, unless additional life is established through ongoing qualification. 10 CFR 50.49(f) establishes four methods of demonstrating qualification for aging and accident conditions. 10 CFR 50.49(k) and (l) permit different qualification criteria to apply based on plant and component vintage.

III.6.2.2 NRC RG 1.89, Rev. 1

NRC RG 1.89, Rev. 1, describes a method acceptable to the NRC staff for complying with the EQ requirements of 10 CFR 50.49 for electric equipment important to safety. It basically states that the procedures described by IEEE Std. 323-1974 "are acceptable to the NRC staff for satisfying the Commission's regulations on environmental qualifications, subject to certain additional requirements related to the types of equipment requiring qualification, the performance and environmental conditions included in the equipment specifications, test conditions, the time margins for equipment operability after a design basis accident, equipment aging, and qualification documentation."

III.6.2.3 DOR Guidelines

The qualification of electric equipment that is subject to significant known degradation due to aging where a qualified life was previously required to be established in accordance with Section 5.2.4 of the DOR Guidelines, should be reviewed for the period of extended operation according to those requirements. If a qualified life was not previously established, the qualification should be reviewed in accordance with Section 7 of the DOR Guidelines.

III.6.2.4 NUREG-0588

The qualification of certain electric equipment important to safety for which qualification was required in accordance with NUREG-0588, Category II, should be reviewed for conformance to those requirements for the period of extended operation to assess the validity of the extended qualification. These requirements include IEEE STD 382-1972 for valve operators and IEEE STD 334-1971.

The qualification of certain electric equipment important to safety for which qualification was required in accordance with NUREG-0588, Category I, should be reviewed for conformance to those requirements for the period of extended operation to assess the validity of the extended qualification.

III.6.3 Final Safety Analysis Report Supplement

Information should be included in the FSAR supplement that provides a summary description of the evaluation of the TLAA associated with EQ of electrical equipment. Additional information is given in Subsection III.1.3.

III.6.4 References

Division of Operating Reactors Guidelines, Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors, November 1979.

Generic Safety Issue-168, Environmental Qualification of Low-Voltage Instrumentation and Control Cables, U.S. Nuclear Regulatory Commission, Washington, DC, February 2001.

IEEE STD 323-1971, IEEE Trial Use Standard; General Guide for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

IEEE STD 323-1974, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations.

IEEE STD 382-1972, Standard for Qualification of Actuators for Power Operated Valve Assemblies with Safety Related Functions for Nuclear Power Plants.

IEEE STD 334-1971, IEEE Standard for Type Tests of Continuous Duty Class 1E Motors for Nuclear Power Generating Stations.

McConnell, J. W., Jr., Ayers, A. L., Jr., and Tyacke, M. J., Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, NUREG/CR-6407, Idaho National Engineering Laboratory, Idaho Falls, ID, February 1996.

NRC Regulatory Issue Summary 2003-09, Environmental Qualification of Low-Voltage Instrumentation and Control Cables, U.S. Nuclear Regulatory Commission, Washington, DC, May 2, 2003.

NRC RG 1.89, Environmental Qualification of Certain Electric Equipment Important to Safety for Nuclear Power Plants, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, June 1984.

NUREG-0588, Interim Staff Position on Environmental Qualification of Safety-Related Equipment, U.S. Nuclear Regulatory Commission, Washington, DC, July 1981.

NUREG-1567, Standard Review Plan For Spent Fuel Dry Storage Facilities, U.S. Nuclear Regulatory Commission, Washington, DC, March 2000.

NUREG-1801, Generic Aging Lessons Learned (GALL) Report, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, December 2010.

NUREG-1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance—Final Report, U.S. Nuclear Regulatory Commission, Washington, DC, March 2011.

III.7 Other Site-Specific Time-Limited Aging Analyses

III.7.1 Description of the Time-Limited Aging Analyses

Certain site-specific safety analyses may have been performed that were based on an explicitly assumed 40- to 60-year SSC life (for example, aspects of the cask or canister design) and may, therefore, be TLAAs. The concern for license renewal is that these analyses may not have properly considered the length of the extended period of operation, which may change conclusions with regard to safety and the capability of SSCs within the scope of the license renewal requirement to perform one or more safety functions. The review of these TLAAs provides the assurance that the aging effect is properly addressed through the period of extended operation. Analogously to the requirements of 10 CFR 54.21(c) for the renewal of operating licenses for nuclear power plants, NUREG-1927 states that the applicant for an ISFSI license renewal or DCSS CoC is required to evaluate TLAAs. Also, site-specific TLAAs may have evolved since issuance of the initial operating license or CoC. Section 3.5.1 of NUREG-1927 provides the guidance on identification of TLAAs.

As stated in NUREG-1927, an applicant must provide a listing of applicable TLAAs in the renewal application, and these TLAAs are identified following the guidance in Section III.1 of this document. On the basis of lessons learned in the review of the initial license renewal applications, evaluations of several commonly encountered TLAAs have been developed and are described in Sections III.2 to III.6. Other site-specific TLAAs that are identified by the applicant are evaluated following the generic guidance in this section.

III.7.2 Dispositioning the Time-Limited Aging Analyses

The acceptance criteria for the TLAAs identified in Subsection III.7.1 should delineate acceptable methods by following the NRC’s guidelines stated in NUREG-1927, Section 3.5.1(5) and listed in Subsection III.1.2.

III.7.2.1 NUREG-1927 Section 3.5.1(5)(i)

The applicant should demonstrate that the analyses have been projected to the end of the period of extended operation. Either the analyses are shown to be bounding even during the period of extended operation, or the analyses are revised for the extended period to show that the TLAA acceptance criteria continue to be satisfied for the period of extended operation.

The applicant should describe the TLAA with respect to the objectives of the analysis, assumptions used in the analysis, conditions, acceptance criteria, relevant aging effects, and intended function(s). The applicant should demonstrate that (a) conditions and assumptions used in the analysis already address the relevant aging effects for the period of extended operation, and (b) acceptance criteria are maintained to provide reasonable assurance that the intended function(s) is maintained for renewal. Thus, no reanalysis would be necessary for license renewal.

In some instances, the applicant may identify activities to be performed to verify the assumption basis of the calculation, such as cycle counting. The applicant should provide an evaluation of that activity. It should be verified that the applicant’s activity is sufficient to confirm the calculation assumptions for the initial licensing period plus the additional renewal period. If necessary, the TLAA may require modification or recalculation to extend the period of evaluation to include the period of extended operation.

III.7.2.2 NUREG-1927 Section 3.5.1(5)(ii)

The applicant may propose an aging management program to manage the aging effects associated with the TLAA using inspections or examinations. The AMP should ensure that the effects of aging on the intended function(s) of the SSCs important to safety are adequately managed in a manner consistent with the original licensing basis for the period of extended operation.

Under this option, the applicant identifies the SSCs associated with the TLAA, and demonstrates the adequacy of the inspection interval of the AMP. If a mitigation or inspection program is proposed, the applicant should use the guidance provided in Section 3.5 of NUREG-1927 to ensure that the effects of aging on the intended function(s) of the structures and components are adequately managed for the period of extended operation.

III.7.3 Final Safety Analysis Report Supplement

Information should be included in the FSAR supplement that provides a summary description of the evaluation of the site-specific TLAAs. Additional information is given in Subsection III.1.3.

III.7.4 References

10 CFR 50.55, Domestic Licensing of Production and Utilization Facilities: Conditions of Construction Permits, Early Site Permits, Combined Licenses, and Manufacturing Licenses, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 54.3, Requirements for Renewal of Operating Licenses for Nuclear Power Plants: Definitions, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR 54.21, Requirements for Renewal of Operating Licenses For Nuclear Power Plant: Contents of Application—Technical Information, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR 54.30, Requirements for Renewal of Operating Licenses For Nuclear Power Plant: Matters Not Subject to a Renewal Review, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR 72.48, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste: Changes, Tests, and Experiments, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR 72.62, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste: Backfitting, Office of the Federal Register, National Archives and Records Administration, 2011.

NUREG-1927, Standard Review Plan for Renewal of Spent Fuel Dry Cask Storage System Licenses and Certificates of Compliance—Final Report, U.S. Nuclear Regulatory Commission, Washington, DC, March 2011.

IV. DESCRIPTION AND EVALUATION OF AGING MANAGEMENT PROGRAMS

IV.S1 Structures Monitoring Program

IV.S1.1 Program Description

The objective of the program is to manage, for plain and reinforced concrete structures, the aging effects of cracking due to freeze-thaw, aggressive chemical attack, loss of bond, and expansion from reaction with aggregates; loss of material and loss of bond due to aggressive chemical attack and corrosion of embedded steel; and increase in porosity and permeability and loss of strength due to aggressive chemical attack and leaching of calcium hydroxide and carbonation. It also manages the aging effects of loss of sealing capacity due to loss of material, cracking, or hardening from weathering of elastomer, rubber, and other similar materials; loss of material due to general, pitting, and crevice corrosion of steel components (screens, frames, bolts, and anchorage); loss of preload due to weathering of bolting; and degradation of lightning protection system components due to wear, corrosion, or weathering.

The structures monitoring program consists of periodic visual inspections of the interior and exterior surfaces of ISFSI structures and structural components at a frequency of at least once a year for evidence of degradation to ensure that aging effects are adequately managed during the period of extended operation. Identified aging effects are evaluated by qualified personnel using criteria derived from industry codes and standards contained in the facility current licensing bases, including ACI 349.3R, ACI 318, SEI/ASCE 11, and the AISC specifications, as applicable.

Paragraph (a)(1) of 10 CFR 72.128 requires that used-fuel storage systems be designed with a capability to test and monitor components important to safety. Implementation of structures monitoring under 10 CFR 50.65 (the Maintenance Rule) is addressed in NRC RG 1.160, Rev. 2, and NUMARC 93-01, Rev. 2. These two documents provide guidance for development of structures monitoring programs to monitor the condition of structures and structural components such that there is no loss of their intended function. Structures monitoring ensures that design assumptions and margins in the original design basis of the ISFSI are maintained and are not unacceptably degraded.

For bolted canister and overpack designs, the program includes preventive actions delineated in NUREG-1339 and in Electric Power Research Institute (EPRI) NP-5769, NP-5067, and TR-104213 to ensure structural bolting integrity. The concrete around the anchorage is inspected for potential loss of concrete anchor capacity due to local concrete degradation such as cracking of concrete due to freeze-thaw at anchor bolt blockouts. Visual inspections should be supplemented with volumetric or surface examinations to detect SCC in high-strength (actual measured yield strength greater than or equal to 150 ksi or greater than or equal to 1,034 MPa) structural bolts greater than 1 inch (25 mm) in diameter.

The program includes the inspection of interior surfaces of concrete overpack or concrete modules for evidence of aging degradation. This inspection may be performed using a video camera and/or fiber optic technology through the openings of the storage system, such as air inlets, air outlets, and access doors. One or more concrete overpack or concrete module housing lead canisters are selected for inspection to demonstrate that concrete surfaces have not undergone unanticipated degradation. A lead canister is one that has the longest time in service, greatest thermal load, and/or other parameters that contribute to degradation.

The program also includes (1) periodic sampling and testing of groundwater chemistry at intervals not to exceed six (6) months to monitor potential seasonal variations due to effects such as winter salting, (2) assessment of the impact of any changes in groundwater chemistry on below-grade concrete structures, and (3) if applicable, monitoring the effectiveness of cathodic protection systems embedded in concrete structures.

IV.S1.2 Evaluation and Technical Basis

1. ***Scope of Program***: The scope of the program includes visual inspection of the surfaces of all concrete structures, anchor bolts, embedments, anchorage systems, and structural commodities such as seals, moisture barriers, caulking, flashing, and other sealants within the scope of aging management. Examples of concrete structures, components, and commodities within the scope of the program are concrete walls, roofs, slabs and pads, structural bolting, anchor bolts and embedment welds. Other structures or components may include lightning protection system components that are within the scope of its structures monitoring program. The inspection is conducted at least once every year. The program also includes periodic sampling and testing of groundwater at intervals not to exceed six months and, if applicable, periodic monitoring of the effectiveness of cathodic protection systems.

   For ISFSIs with roof systems such as horizontal storage concrete modules, this program monitors the structural integrity of air-outlet shielding blocks, deteriorated penetrations (i.e., drains, vents, etc.), and signs of water infiltration, cracks, ponding, and flashing degradation. The air-outlet shielding blocks are generally welded to the embedded base plate in the roof, and the structural integrity of the welds, base plate, and surrounding concrete is also monitored. Significant concrete cracks have been observed in HSMs due to freeze-thaw of concrete at the locations of anchor bolt blockouts, according to a 2011 NRC inspection report of an ISFSI facility (NRC Inspection Report ML11097A028).

2. ***Preventive Action***: The program is primarily a condition-monitoring program. The program also includes, if applicable, preventive actions delineated in NUREG-1339 and in EPRI NP-5769, NP-5067, and TR-104213 to ensure structural bolting integrity. These actions emphasize proper selection of bolting material, lubricants, and installation torque or tension to prevent or minimize loss of bolting preload and cracking of high-strength bolting. If the structural bolting consists of ASTM A325, ASTM F1852, and/or ASTM A490 bolts, the preventive actions for storage, lubricants, and SCC potential discussed in Section 2 of the RCSC (Research Council for Structural Connections) publication “Specification for Structural Joints Using ASTM A325 or A490 Bolts” should be used.

3. ***Parameters Monitored or Inspected***: For each structure/aging effect combination, the specific parameters monitored or inspected depend on the particular structure, structural component, or commodity. Parameters monitored or inspected are commensurate with industry codes, standards, and guidelines and also consider industry and site-specific operating experience. ACI 349.3R and ANSI/ASCE 11 provide an acceptable basis for selection of parameters to be monitored or inspected for

concrete and steel structural elements and for joints, coatings, and waterproofing membranes (if applicable).
For concrete structures, parameters monitored include (1) cracking, loss of bond, and loss of material (spalling and scaling) due to corrosion of embedded steel, freeze-thaw, or aggressive chemical attack; (2) cracking due to expansion from reaction with aggregates or increased stress levels from soil settlement; (3) increase in porosity and permeability due to leaching of calcium hydroxide and carbonation or aggressive chemical attack; (4) reduction of concrete strength and modulus due to elevated temperature (>150°F general; >200°F local) or concrete interaction with aluminum; (5) reduction of foundation strength from cracking due to differential settlement and erosion of the porous concrete subfoundation; and (6) reduction of concrete anchor capacity due to local concrete degradation.

Steel structures and components are monitored for loss of material due to corrosion and cracking due to cyclic load and other conditions such as deflections, twisted beams, loss or missing anchors, and missing or degraded grout under base plates. Welds at access doors and other locations are monitored for cracked welds due to cyclic loads and other conditions.

Structural bolting is monitored for loss of preload due to self-loosening, missing or loose nuts, and conditions indicative of loss of preload. High-strength (actual measured yield strength ≥150 ksi or 1,034 MPa) structural bolts greater than 1 inch (25 mm) in diameter are monitored for SCC. Other structural bolting (ASTM A-325, ASTM F1852, and ASTM A490 bolts) and anchor bolts are monitored for loss of material and loose or missing nuts. The concrete around the anchorage is inspected to detect any reduction of concrete anchor capacity due to local concrete degradation such as cracks due to freeze-thaw at anchor bolt blockouts and other effects.

Structural sealants and moisture barriers are monitored for loss of sealing due to cracking, loss of material, and hardening, if applicable. These parameters and other monitored parameters are selected to ensure that aging degradation leading to loss of intended functions will be detected and the extent of degradation can be determined. In the HSM, steel dowels are placed between the precast concrete storage module and pad to serve as seismic restraints for preventing sliding of the module. Special attention should be given to the sealants and moisture barriers around the joints to ensure that corrosion of the dowels does not occur because of degradation of the sealants.

Groundwater chemistry (pH, chlorides, and sulfates) is monitored periodically to assess its potential to promote aggressive chemical attack on below-grade concrete structures. If a site de-watering system is necessary for managing settlement and erosion of porous concrete sub-foundations, its continued functionality is monitored. For cathodic protection systems, the effectiveness of cathodic protection current is monitored. The site-specific structures monitoring program should contain sufficient detail on parameters monitored or inspected to verify that this program attribute is satisfied.

4. ***Detection of Aging Effects***: The exterior surfaces of concrete and structural components are monitored under this program using periodic visual inspection of each structure/aging effect combination by a qualified inspector to ensure that aging

degradation will be detected and quantified before there is loss of intended functions. Inspection of the interior surfaces of concrete structures and structural components may be performed using a video camera and/or fiber optic technology through the openings of the storage system, such as air inlets, air outlets, and access doors. Visual inspection of high-strength (actual measured yield strength ≥150 ksi or 1,034 MPa) structural bolting greater than 1 inch (25 mm) in diameter is supplemented with volumetric or surface examinations to detect cracking. Other structural bolting (ASTM A-325, ASTM F1852, and ASTM A490 bolts) and anchor bolts are monitored for loss of material, loose or missing nuts, and cracking of concrete around the anchor bolts. Surfaces of concrete pads are inspected for indication of sliding for the vertical free-standing canister and overpack. All structures are inspected every year and groundwater quality is monitored at least once every six months. Inspector qualifications should be consistent with industry guidelines and standards and guidelines for implementing the requirements of 10 CFR 50.65. Qualifications of inspection and evaluation personnel specified in ACI 349.3R are acceptable for license renewal, as prescribed in 10 CFR 72.158.

The structures monitoring program addresses detection of aging effects for inaccessible, below-grade concrete structural elements. For facilities with non-aggressive groundwater/soil (pH >5.5, chlorides <500 ppm, and sulfates <1500 ppm), the program recommends (a) evaluating the acceptability of inaccessible areas when conditions exist in accessible areas that could indicate the presence of, or result in, degradation in such inaccessible areas and (b) examining representative samples of the exposed portions of the below-grade concrete, when excavated for some other reason.

Facilities with aggressive groundwater/soil (pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm) and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.

5. ***Monitoring and Trending***: 10 CFR 72.122(h)(4) states that “Storage confinement systems must have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. For dry used fuel storage, periodic monitoring is sufficient.” Regulatory Position 1.5, “Monitoring of Structures,” in NRC RG 1.160, Rev. 2, provides an acceptable basis for satisfying this program element. A structure may be monitored in accordance with 10 CFR 50.65(a)(2) provided there is no significant degradation of the structure. A structure is monitored in accordance with 10 CFR 50.65(a)(1) if the extent of degradation is such that the structure may not meet its design basis or, if allowed to continue uncorrected until the next normally scheduled assessment, may not meet its design basis.

   A baseline inspection result for the condition of the SSCs should be established. The conditions of the SSCs observed in subsequent inspections should be compared with the baseline conditions of the SSCs for trending purposes.

6. ***Acceptance Criteria***: The structures monitoring program calls for inspection results to be evaluated by qualified engineering personnel based on acceptance criteria selected for

each structure/aging effect to ensure that the need for corrective actions is identified before loss of intended function occurs. The criteria are derived from design basis codes and standards that include ACI 349, ACI 318, SEI/ASCE 11, ASME Code, or the relevant AISC specifications, as applicable, and consider industry and facility operating experience. The criteria are directed at the identification and evaluation of degradation that may affect the ability of the structure or component to perform its intended function. Licensees who are not committed to ACI 349.3R and elect to use site-specific criteria for concrete structures should describe the criteria and provide a technical basis for deviations from those in ACI 349.3R. Loose bolts and nuts and cracked high-strength bolts are not acceptable unless approved by engineering evaluation.

Structural sealants are acceptable if the observed loss of material, cracking, and hardening will not result in loss of sealing. The structures monitoring program is to contain sufficient detail on acceptance criteria to conclude that this program attribute is satisfied.

7. ***Corrective Actions*****:** Evaluations are performed for any inspection results that do not satisfy established criteria. Corrective actions are initiated in accordance with the corrective-action process if the evaluation results indicate that there is a need for a repair or replacement. As discussed in the appendix to this report, the requirements of 10 CFR Part 72, Appendix G, are acceptable to address the corrective actions.

   Cracks in concrete have many causes. The root cause of the cracking must be evaluated to ensure that the condition of the concrete will not accelerate structural degradation during the period of extended operation. Good concrete crack repair techniques also depend upon understanding the causes of cracking and selecting appropriate repair procedures. Guidance on the causes of concrete cracking, crack evaluation, and repair of concrete structures is given in ACI 224.1R. Guidance on controlling the corrosion of embedded reinforcing steel and the repair and rehabilitation of concrete structures with corroded reinforcing steel is given in ACI 222R.

8. ***Confirmation Process*****:** Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 72, Appendix G. As discussed in the appendix to this report, the requirements of 10 CFR Part 72 Appendix G are acceptable to address the corrective actions, confirmation process, and administrative controls.

9. ***Administrative Controls***: See element 8, above.

10. ***Operating Experience*****:** Structures monitoring programs have been implemented for managing aging effects during the extended period of license renewal of the operating reactor plants. NUREG-1522 documents the results of a survey in 1992 to obtain information on the types of distress in the concrete and steel structures and components, the type of repairs performed, and the durability of the repairs. Licensees who responded to the survey reported cracking, scaling, and leaching of concrete structures. The degradation was attributed to drying shrinkage, freeze-thaw, and abrasion. The degradation also includes corrosion of component support members and anchor bolts, cracks and other deterioration of masonry walls, and groundwater leakage

and seepage into underground structures. The degradations at coastal plants were more severe than those observed in inland plants as a result of contact with brackish and seawater. The license renewal applicants reported similar degradation and corrective actions taken through their structures monitoring program. There is reasonable assurance that implementation of the structures monitoring program described here will be effective in managing the aging of the in-scope structures and components of ISFSI facilities through the period of extended operation.

IV.S1.3 References

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10 CFR Part 72, Appendix G, Quality Assurance Criteria for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste, Office of the Federal Register, National Archives and Records Administration, as amended June 9, 2008.

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NRC Regulatory Guide 1.160, Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Revision 2, U.S. Nuclear Regulatory Commission, Washington, DC, March 1997.

NUMARC, Industry Guideline for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants (Line-In/Line-Out Version), NUMARC Report 93-01, Revision 2, Nuclear Utility Management and Resources Council, Washington, DC, April 1996.

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IV.S2 Monitoring of Protective Coating on Carbon Steel Structures

IV.S2.1 Program Description

The objective of the program is to manage the aging effects on the protective coating on carbon steels structures used in the dry cask storage systems. Proper maintenance of protective coatings on the external surfaces of carbon steel overpack structures exposed to outdoor air is essential to provide protection to the exposed metal surfaces of the ISFSI overpack. Degraded coatings on steel structures inside the overpack could clog the penetrations and reduce the airflow through the system. Such clogging could cause an unacceptable increase in temperature of the confinement and adversely affect the function of the overpack. For this reason, the coatings for the ISFSI overpack should be treated as Service Level III coatings as defined by American Society for Testing and Materials (ASTM) D 5144-08.

Maintenance of coatings applied to carbon steel surfaces of the overpack (e.g., steel overpack structure, lid plates, lid studs and nuts, base plate, plates for inlet vents, and concrete shield block and pedestal shield) also serves to prevent or minimize loss of material due to corrosion of carbon steel components. Regulatory Position C4 in NRC RG 1.54, Rev. 2, refers to ASTM D 7167-05 as an acceptable technical basis for monitoring the performance of Service Level III coatings that can be credited for establishing procedures to monitor the performance of Service Level III coatings on the exposed surfaces of the ISFSI overpack. In addition, EPRI Report 1003102, Guidelines for Inspection and Maintenance of Safety-related Protective Coatings, provides additional information on the ASTM Standard guidelines.

A comparable program for monitoring and maintaining Service Level III protective coatings for nuclear power plants developed in accordance with NRC RG 1.54, Rev. 2, is acceptable as an AMP for ISFSI overpack coatings.

IV.S2.2 Evaluation and Technical Basis

1. ***Scope of Program:*** The minimum scope of this program includes coatings applied to steel surfaces of the overpack that are exposed to the outside environment. The scope of the program also includes any coatings that are credited by the licensee for preventing loss of material due to corrosion.

2. ***Preventive Action:*** The program is a condition-monitoring program and does not recommend any preventive actions. However, for applicants that credit coatings for minimizing loss of material, this program is a preventive action.

3. ***Parameters Monitored or Inspected:*** For components with coatings, coating deterioration is an indicator of possible underlying degradation. ASTM D 7167-05 provides guidelines for establishing procedures to monitor the performance of Service Level III coatings. It also refers to other ASTM standards, such as test methods for estimating dry film thickness and adhesion strength of coatings that are determined to be deficient or degraded, which may be followed, as appropriate, for monitoring the performance of ISFSI overpack coatings.

4. ***Detection of Aging Effects:*** For coated surfaces, confirmation of the integrity of the paint or coating is an effective method for managing the effects of corrosion on the metallic surface. For metallic components under a protective cover, confirmation of absence of any leakage of rainwater is an effective method for managing the effects of corrosion on the metallic components under the coating. ASTM D 7167-05, Paragraph 6, provides guidelines for determining the inspection frequency of the coatings on the ISFSI overpack, and Paragraph 10 provides guidelines for developing an inspection plan and selecting the test methods to be used. Subparagraph 10.2 states, "Condition assessment shall include a visual inspection of the designated lined surfaces to identify defects, such as blistering, cracking, flaking/peeling/delamination, rusting, and physical damage." Field documentation of inspection results is addressed in Subparagraph 10.3 and Paragraph 11.

5. ***Monitoring and Trending:*** Subparagraph 7.2 of ASTM D 7167-05 identifies monitoring and trending activities, and specifies a pre-inspection review of the previous two or more monitoring reports; Paragraph 12 specifies that the inspection report should prioritize repair areas as either needing repair during the same outage or postponed to future outages, but under surveillance in the interim period.

6. ***Acceptance Criteria:*** ASTM D 7167-05, Subparagraphs 10.2.1 through 10.2.6, 10.3, and 10.4, contain one acceptable method for characterization, documentation, and testing of defective or deficient coating surfaces that exhibit blistering, cracking, flaking, peeling, delamination, and rusting. Additional ASTM standards and other recognized test methods are available for use in characterizing the severity of observed defects and deficiencies. Paragraph 12 addresses evaluation. It specifies that the inspection report is to be evaluated by the responsible evaluation personnel, who prepare a summary of findings and recommendations for future surveillance or repair, including an analysis of reasons or suspected reasons for failure. Areas requiring repair work are prioritized as major or minor defective areas.

7. ***Corrective Actions:*** A recommended corrective action plan is required for timely repair of major defective areas. As discussed in the appendix to this report, the requirements of 10 CFR Part 72, Appendix G, are acceptable to address the corrective actions.

8. ***Confirmation Process:*** Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 72, Appendix G. As discussed in the appendix to this report, the requirements of 10 CFR Part 72, Subpart G, are acceptable to address the confirmation process and administrative controls.

9. ***Administrative Controls:*** See element 8 above.

10. ***Operating Experience:*** Operating experience with Service Level III coatings on ISFSI overpack is limited. However, the experience with Service Level I coatings at nuclear power plants can provide some useful insight. NRC Information Notice 88-82, NRC Information Notice 97-13, NRC Bulletin 96-03, NRC GL 04-02, and NRC GL 98-04 describe industry experience pertaining to coatings degradation inside nuclear power plant containments and the consequential potential clogging of sump strainers. NRC RG 1.54, Rev. 2, was issued in July 2010. Monitoring and maintenance of Service Level III coatings

conducted in accordance with Regulatory Position C4 is considered to be an effective program for managing degradation of Service Level III coatings on ISFSI overpacks.

IV.S2.3 References

10 CFR Part 72, Appendix G, Quality Assurance Criteria for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste, Office of the Federal Register, National Archives and Records Administration, as amended June 9, 2008.

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2009.

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IV.M1 External Surfaces Monitoring of Mechanical Components

IV.M1.1 Program Description

The objective of the program is to manage the aging effects on the mechanical components of the dry cask storage systems. The program consists of periodic visual inspections of surfaces of metallic (and polymeric) components such as used-fuel canisters, support structures, access doors, vents, and other components within the scope of the used-fuel storage facility license, and subject to aging effects that need to be managed for the period of the extended operation. The program manages aging effects through inspection of external surfaces for evidence of loss of material due to corrosion and wear, cracking due to SCC or fatigue, leakage of rainwater under the protective cover causing corrosion of carbon steel components, and change in material properties due to temperature and radiation. Inspection of surfaces in narrow spaces or annuli and in areas with limited access may be performed using a camera and/or fiber optic technology through the openings of the storage system, such as air inlet/outlet vents or access doors. The inspections are conducted on a sampling basis at a frequency of at least once every year. One or more lead canisters are selected for inspection to demonstrate that canisters have not undergone unanticipated degradation. A lead canister is one that has the longest time in service, greatest thermal load, and/or other parameters that contribute to degradation. The aging degradation of protective coatings on the external surfaces of carbon steel structures (e.g., cracking, flaking, and blistering) is managed by AMP IV.S2 “Monitoring of Protective Coating on Carbon Steel Structures.”

IV.M1.2 Evaluation and Technical Basis

1. ***Scope of Program:*** This program visually inspects and monitors the external surfaces of mechanical components in used-fuel storage systems that are subject to loss of materials, change in mechanical properties, or leakage of rainwater causing corrosion of steel components. The program scope includes surfaces of metallic (and polymeric) components such as used-fuel canisters, support structures, access doors, vents, and other components within the scope of the ISFSI license. Cracking of stainless steel components exposed to an air environment containing halides may also be managed. In addition, this program visually inspects and monitors the external surfaces of polymeric components in mechanical systems within the scope of license renewal and subject to AMR for changes in material properties (such as hardening and loss of strength), cracking, and loss of material due to wear.

2. ***Preventive Actions:*** The program is a condition-monitoring program that does not include preventive actions.

3. ***Parameters Monitored or Inspected:*** The External Surfaces Monitoring of Mechanical Components program provides visual inspections to monitor for material degradation of the canister and other mechanical components. Inspection can reveal cracking; loss of material due to corrosion; and indications of degradation due to wear, such as verification of clearances, settings, loose or missing parts, debris, loss of integrity at bolted or welded connections, or indication of rainwater leakage.

   Examples of inspection parameters for metallic components include the following:

   - Corrosion and material wastage (loss of material)

- Worn, flaking, or oxide-coated surfaces (loss of material)
- Corrosion stains on adjacent components and structures (loss of material)
- Surface cracks (cracking)
- Stains caused by leaking rainwater

Examples of inspection parameters for polymers include:

- Surface cracking, crazing, scuffing, and dimensional change (ballooning and necking)
- Discoloration
- Hardening as evidenced by a loss of suppleness during manipulation at the location where the component and material are amenable to manipulation
- Exposure of internal reinforcement for reinforced elastomers

4. ***Detection of Aging Effects:*** This program manages aging effects of loss of material due to corrosion, cracking due to SCC or cyclic load, and changes in mechanical properties due to temperature and radiation, using visual inspection. When required by the ASME Code, inspections are conducted in accordance with the applicable code requirements. In the absence of applicable code requirements, site-specific visual inspections of metallic and polymeric component surfaces are performed using site-specific personnel qualification procedures. The inspections are capable of detecting age-related degradation and are performed at a frequency of at least once in 10 years. Visual inspection is capable of detecting age-related degradation such as loss of material due to corrosion, and cracking of metallic components, welds, and concrete. Remote inspection using a camera and/or fiber optic technology through openings, such as air inlets and outlets, is acceptable. Also, access doors or covers can be removed for inspection of the canister and support structure for the canister for sign of aging degradation.

   Visual inspection will identify indirect indicators of flexible-polymer hardening and loss of strength and will include the presence of surface cracking, crazing, discoloration, and for elastomers with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Visual inspection should cover 100 percent of accessible components. Visual inspection will identify direct indicators of loss of material due to wear, to include dimensional change, scuffing, and for flexible polymeric materials with internal reinforcement, the exposure of reinforcing fibers, mesh, or underlying metal. Manual or physical manipulation can be used to augment visual inspection to confirm the absence of hardening and loss of strength for flexible polymeric materials where appropriate. Hardening and loss of strength and loss of material due to wear for flexible polymeric materials are expected to be detectable prior to any loss of intended function.

5. ***Monitoring and Trending:*** Visual inspections are performed at intervals not to exceed 10 years except for indications of rainwater leakage, which are the subject of inspection every year. The associated personnel are qualified in accordance with site-controlled procedures and processes as prescribed in 10 CFR 72.158. Standardized monitoring and

trending activities are used to track degradation, such as performing a baseline inspection for subsequent trending. Deficiencies are documented using approved processes and procedures, such that results can be trended. This monitoring should be conducted in accordance with the guidance provided in 10 CFR 122(h)(4).

6. ***Acceptance Criteria:*** For each component/aging effect combination, the acceptance criteria are defined to ensure that the need for corrective actions will be identified before loss of intended functions. Any indications of relevant degradation detected should be evaluated for continued service in the corrective-action program. For stainless steel surfaces, a clean shiny surface is expected. The appearance of discoloration may indicate the loss of material on the stainless steel surface. For aluminum and copper alloys exposed to marine or industrial environments, any indications of relevant degradation that could impact their intended function are evaluated. For flexible polymers, a uniform surface texture and uniform color with no unanticipated dimensional change is expected. Any abnormal surface condition may be an indication of an aging effect for metals and for polymers. For flexible materials, changes in physical properties (e.g., the hardness, flexibility, physical dimensions, and color of the material relative to when the material was new) should be evaluated for continued service in the corrective-action program. Cracks should be absent within the material. For rigid polymers, surface changes affecting performance, such as erosion, cracking, crazing, checking, and chalking, are subject to further investigation. Acceptance criteria include design standards, facility procedural requirements, current licensing basis, industry codes or standards, and engineering evaluation.

7. ***Corrective Actions****:* Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 72, Appendix G. As discussed in the appendix to this report, the requirements of 10 CFR Part 72, Appendix G, are acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process:*** See element 7, above.

9. ***Administrative Controls:*** See element 7, above.

10. ***Operating Experience:*** External surface inspections as a part of system inspections have been in effect at many nuclear utilities since the mid-1990s in support of the Maintenance Rule (10 CFR 50.65) and have proven effective in maintaining the material condition of plant systems. The elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice.

IV.M1.3 References

10 CFR Part 72, Appendix G, Quality Assurance Criteria for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste, Office of the Federal Register, National Archives and Records Administration, as amended June 9, 2008.

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2009.

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Virginia Electric and Power Company, Surry Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application, Docket No. 72-2, April 29, 2002.

IV.M2 Ventilation Surveillance Program

IV.M2.1 Program Description

The objective of the program is to manage the aging effects of loss of material due to corrosion and wear, and cracking of the external surfaces of the components in the ventilation systems of ISFSIs. The Ventilation Surveillance program is based on system inspections and walkdowns. This program consists of daily visual inspections of the components of ventilation systems such as air inlets, air outlets, and other components. The program manages aging effects through visual inspection of external surfaces of the ventilation system to ensure that the air inlets and outlets and other components are intact and free from blockage, loss of material due to corrosion and wear, and cracking. The aging degradation of protective coatings on the external surfaces of the ventilation system components (e.g., cracking, flaking, and blistering) is managed by AMP IV.S2 "Monitoring of Protective Coating on Carbon Steel Structures."

IV.M2.2 Evaluation and Technical Basis

1. ***Scope of Program:*** This program visually inspects and monitors the external surfaces of the components in the ventilation system such as air inlets and outlets and other components to ensure they are free from blockage, loss of material due to corrosion and wear, and cracking. The inspection covers all the storage units at a site.

2. ***Preventive Actions:*** The daily inspection maintains the inlets and outlets free from obstruction and other aging effects to ensure that temperatures are not elevated for prolonged periods. This measure prevents thermally induced damage to concrete components and overheating of the canister.

3. ***Parameters Monitored or Inspected:*** The Ventilation Surveillance program utilizes daily system inspections and walkdowns to monitor for material degradation and blockage of air inlets and outlets.

   Examples of inspection parameters for the components include the following:

   - Blockage of air inlet and outlet opening (reduction of heat transfer capability)
   - Corrosion and wear (loss of material)
   - Cracks in stainless steel components exposed to outdoor environments

4. ***Detection of Aging Effects:*** The program manages aging effects including reduction of heat transfer capability due to blockage of air inlet and outlet openings, loss of material due to corrosion and wear, and cracking due to SCC or cyclic load, using visual inspection.

   Visual inspections should be performed daily and should cover 100% of the accessible components. Visual inspection is capable of detecting age-related degradation such as blockage, corrosion, wear, and cracking. The inspection should be conducted daily to ensure that the components' intended function is maintained. Inspection frequencies other than daily, such as every 2 to 3 days, should be justified to ensure that elevated temperatures are not occurring within the inspection period, thereby causing damage to concrete components and overheating of the canister.

5. ***Monitoring and Trending:*** Visual inspections are performed daily and associated personnel are qualified in accordance with site-controlled procedures and processes as prescribed in 10 CFR 72.158. Standardized monitoring and trending activities are used to track degradation. Deficiencies are documented using approved processes and procedures, such that results can be trended.

6. ***Acceptance Criteria:*** For each component/aging-effect combination, the acceptance criteria are defined to ensure that the need for corrective actions will be identified before loss of intended functions. Any indications of relevant degradation detected should be evaluated for continued service in the corrective-action program. Cracks should be absent within the material. Acceptance criteria include design standards, facility procedural requirements, current licensing basis, industry codes or standards, and engineering evaluation.

7. ***Corrective Actions:*** Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 72, Appendix G. As discussed in the appendix to this report, the requirements of 10 CFR Part 72, Appendix G, are acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process*:** See element 7, above.

9. ***Administrative Controls*:** See element 7, above.

10. ***Operating Experience:*** External surface inspections by means of system inspections and walkdowns have been in effect at many nuclear utilities since the mid-1990s in support of the Maintenance Rule (10 CFR 50.65) and have proven effective in maintaining the material condition of plant systems. The elements that comprise these inspections (e.g., the scope of the inspections and inspection techniques) are consistent with industry practice.

IV.M2.3 References

10 CFR Part 72, Appendix G, Quality Assurance Criteria for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste, Office of the Federal Register, National Archives and Records Administration, as amended June 9, 2008.

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 50.65, Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2009.

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Safety Evaluation Report, Docket No. 72-09, U.S. Department of Energy Fort St. Vrain Independent Spent Fuel Storage Installation License No. SNM-2504 License Renewal, U.S. Nuclear Regulatory Commission, Washington, DC, 2011.

Safety Evaluation Report of Vectra Technologies, Inc., a.k.a. Pacific Nuclear Fuel Services, Inc., Safety Analysis Report for the Standarized NUHOMS Horizontal Modular Storage for Irradiated Nuclear Fuels (ML053410448), U.S. Nuclear Regulatory Commission, Washington, DC, Dec. 1994

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IV.M3 Welded Canister Seal and Leakage Monitoring Program

IV.M3.1 Program Description

The objective of the program is to manage the aging effects of cracking and leakage of the welded used-fuel storage canisters due to SCC when exposed to moisture and aggressive chemicals in the environment (e.g., marine environment). 10 CFR 72.122(h)(4) specifies that storage confinement systems must have the capability for continuous monitoring, such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions, and 10 CFR 72.128(a)(1) specifies that used-fuel storage systems must be designed with a capability to test and monitor components important to safety. In addition, NRC ISG-25, "Pressure and Helium Leakage Testing of the Confinement Boundary of Spent Fuel Dry Storage Systems," provides supplemental guidance for evaluating the helium leakage testing and ASME Code-required hydrostatic/pneumatic pressure testing that is specified for the DCSS confinement boundary.

The DCSS confinement boundary for welded canisters consists of the canister shell, bottom plate, top lid, vent and drain port cover plates, and the interconnecting welds. Typically, the shell-to-bottom-plate and shell seam welds are prepared in the fabrication shop, while the lid-to-shell and vent port cover plate welds are prepared in the field after fuel loading. The "Discussion" section of ISG-25 states, "If the entire confinement boundary is tested to be "leak tight" in accordance with ANSI-N14.5 'Leakage Tests on Packages for Shipment of Radioactive Materials,' (i.e. 1.0×10^{-7} ref. cm^3/sec) and the canister lid-to-shell weld conforms to the criteria of ISG-18, then leakage is not considered credible and effluents are not required to be considered in confinement dose analyses."

ISG-5 Rev.1, "Confinement Evaluation," provides guidance for evaluating the design and analysis of the proposed cask confinement system for normal, off-normal, and accident conditions. Acceptance criteria IV(4) of ISG-5, Rev. 1, state that

> The applicant should describe the proposed monitoring capability and/or surveillance plans for mechanical closure seals. In instances involving welded closures, the staff has previously accepted that no closure monitoring system is required. This practice is consistent with the fact that other welded joints in the confinement system are not monitored. However, the lack of a closure monitoring system has typically been coupled with a surveillance program that would enable the licensee to take timely and appropriate corrective actions to maintain safe storage conditions if closure degradation occurred.
>
> To show compliance with 10 CFR 72.122(h)(4), cask vendors have proposed, and the staff has accepted, routine surveillance programs and active instrumentation to meet the continuous monitoring requirement.

The Welded Canister Seal and Leakage Monitoring Program is a site-specific management program that consists of the following:

(a) Assessment of the storage canisters/casks to verify that they were designed, fabricated, erected, and tested in accordance with the guidance of NRC ISG-15 for evaluating material-related issues for used-fuel storage canisters under normal, off-normal, and accident conditions, and with the recommendations of NRC ISG-18 for the design and testing of the various closure welds, or "lid welds," associated with the redundant closure of all-welded austenitic stainless steel canisters.

(b) Examination of the actual environmental conditions of the used-fuel storage canister welds to establish the surface temperature, humidity at the canister surface, and deposits (e.g., chlorides) on canister.

(c) Evaluation of the susceptibility of the storage canister welds to SCC under the actual environmental conditions at the used-fuel storage site. In addition, based on the assessment in item (a) and the site-specific inspection results (e.g., presence of surface corrosion pits), establish the stress conditions of the canister welds.

(d) Based on the information obtained from items (a) through (c), develop a site-specific program to manage the aging effects of cracking and leakage of the welded used-fuel storage canisters due to SCC in an aggressive environment (e.g., marine environment). The licensee may participate in industry programs for investigating and managing effects of SCC of canister welds in marine environments, and evaluate and implement the results as applicable to the specific site.

An acceptable program should include (i) a remote inspection technique to examine the canister weld surface because access to the canister surface is very limited, (ii) monitoring of weld surface temperatures and humidity of the environment near the canister confinement welds, (iii) mechanical means (e.g., tapes, vacuum brush, or coupons) for identifying surface deposits, and (iv) methods to remove the deposits of aggressive chemicals or the use of inhibitors to counteract their effect without removal.

IV.M3.2 Evaluation and Technical Basis

1. ***Scope of Program:*** The program consists of monitoring or examination of the welds of the canister confinement boundary that includes welds of closure lid and penetrations such as vent and drain port cover plates and other interconnecting welds to ensure that timely and appropriate corrective actions can be taken to maintain safe storage conditions of the canister.

2. ***Preventive Actions:*** The preventive actions for this program include monitoring of the confinement weld surface conditions such as temperature, humidity, and surface deposits (e.g., chlorides) to identify and avoid conditions that may cause deliquescence of dry salt deposits and result in SCC of austenitic stainless steels. For ISFSI site conditions where surface deposits of sea salt or other aggressive chemicals are observed, preventive actions also include methods to remove the deposits of aggressive chemicals or the use of inhibitors to counteract their effect.

   In addition, preventive actions include an assessment of compliance with the recommendations of NRC ISG-15, "Materials Evaluation." It provides specific guidance for evaluating material-related issues for used-fuel storage canisters under normal, off-normal, and accident conditions. For continued confinement effectiveness during storage, the welded closure canisters rely on weld integrity. Preparation and examination of a weld in accordance with ISG-15 provides reasonable assurance that no flaw of significant size exists such that it could impair the structural strength or confinement capability of the weld. Therefore, helium leakage testing of such welds is unnecessary provided the weld is also in compliance with the guidance of ISG-18.

ISG-18, "The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage," addresses the design and testing of the various closure-welds, or "lid welds," associated with the redundant closure of all-welded austenitic stainless steel canisters. 10 CFR 72.236(e) states that, "the spent-fuel storage cask must be designed to provide redundant sealing of confinement systems." For a welded canister design, the NRC staff has accepted closure designs employing redundant lids or covers, each with independent field welds. Thus, a potential leak path would have to sequentially breach two independent welds before the confinement system would be compromised.

As discussed above, the entire confinement boundary of storage canisters is examined and helium leak tested to meet the requirements of 10 CFR 72. For continued confinement effectiveness during storage, the bolted-closure canisters incorporate a helium monitoring system, whereas the welded-closure canisters rely on weld integrity. Consequently, at least one of the redundant welded closures must be helium leakage tested per the method of ANSI N14.5, except when the large, multi-pass weld joining an austenitic stainless steel canister shell to the canister lid is prepared and examined in accordance with guidance provided in ISG-15 and meets the Helium Leakage Test—Large Welded Exception Criteria of ISG-18R1. A "multi-pass weld" represents a weld with three or more individual layers of weld metal, in which each layer may be composed of a single weld-bead or several adjacent weld beads of common thickness. A minimum of three layers minimizes the probability of a weld flaw propagating through the weld layers, resulting in a leakage path. Thus, a multi-pass weld provides reasonable assurance against the existence of a flaw of sufficient size to impair the structural strength or confinement capability of the weld. Therefore, helium leakage testing of such multi-pass welds is unnecessary, provided the weld is prepared and examined in accordance with ISG-15 and follows the technical review guidance of ISG-18.

3. ***Parameters Monitored or Inspected:*** The program monitors and inspects imperfections such as cracking due to SCC or fatigue/cyclic loading; loss of material due to general corrosion or pitting; or other aging degradation in the canister confinement welds that could significantly reduce its structural integrity and confinement effectiveness. The program manages cracking by monitoring for evidence of surface breaking linear discontinuities or pinholes if a remote visual inspection technique is used. The program also manages loss of material by monitoring for gross or abnormal surface condition such as corrosion products or pitting on the surface of the welds and heat-affected-zone (HAZ) adjacent to the weld. Furthermore, the environmental conditions near the confinement welds such as temperature, humidity, and surface deposits (e.g., chlorides) are monitored to identify and avoid conditions that cause deliquescence of dry salt deposits, which may lead to SCC of the welds or HAZ adjacent to the welds.

4. ***Detection of Aging Effects:*** This program manages aging effects of cracking due to SCC or cyclic load, and loss of material due to corrosion, using remote visual inspection. When required by the ASME Code, inspections are conducted in accordance with the applicable code requirements. In the absence of applicable code requirements, site-

specific visual inspections of metallic component surfaces are performed using approved site-specific procedures. Remote inspection using a camera and/or fiber optic technology through openings, such as air inlets and outlets, is acceptable. These methods include various visual examinations for detecting aging-related degradation such as general surface condition to detection and sizing of surface breaking discontinuities. Surface examinations may also be used as an alternative to visual examinations for detection and sizing of surface breaking discontinuities.

The sample size of storage canisters to be inspected is to be based on an assessment of compliance with the guidance of ISG-15 and ISG-18, environment, estimated stress state of the weld, and operating experience. The interval between examinations should not exceed 10 years. For ISFSIs in marine environments (i.e., salty air), a shorter inspection interval should be considered to ensure that the corrective action is taken in a timely manner if closure degradation has occurred.

The inspection of the weld should be performed by qualified personnel who meet the requirements of ASME B&PV Code Section XI, IWA-2300, "Qualification of Nondestructive Examination Personnel," as prescribed in 10 CFR 72.158.

5. ***Monitoring and Trending:*** The methods for monitoring, recording, evaluating, and trending the results from the inspection program are in accordance with the applicable ASME Code Section XI requirements or approved site-specific procedures.

6. ***Acceptance Criteria:*** The program provides specific examination acceptance criteria for the canister confinement welds remote inspections. For examinations performed in accordance with ASME Code Section XI, the acceptance criteria of Subsection IWB-3500 apply.

7. ***Corrective Actions:*** Any detected conditions that do not satisfy the examination acceptance criteria are required to be dispositioned through the site corrective action program, which may require repair, repackaging, or analytical evaluation for continued service until the next inspection. If inspection indicates that a flaw is unacceptable, repair is in accordance with ASME Code, Section XI, IWA-4000, or perform flaw evaluation in accordance with ASME Code, Section XI IWA-4422.1, to determine whether the flaw is acceptable for continued service.

8. ***Confirmation Process*:** Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 72, Appendix G. As discussed in the appendix to this report, the requirements of 10 CFR Part 72, Appendix G are acceptable to address the confirmation process, and administrative controls.

9. ***Administrative Controls*:** See element 8, above.

10. ***Operating Experience:*** Studies on the susceptibility to SCC of Type 304, 304L, and 316L austenitic stainless steels and their welds in marine environments indicate that chloride-induced SCC is strongly dependent on the concentration of salt deposits, residual stress, cask temperature, and the relative humidity of the surrounding environment *(NUREG-7030).* The results of salt fog tests, although considered conservative because

of the high absolute humidity used in these tests, demonstrate that the deliquescence of dry salt deposits can lead to SCC of austenitic stainless steels at temperatures that are only slightly greater than ambient temperatures [e.g., 43°C (109°F)]. Isolated corrosion pits and general corrosion is also observed at these temperatures, particularly in the heat-affected zone because of chromium depletion from the matrix. Cracking is primarily transgranular with sections of intergranular branching, and occurs in regions where tensile stresses are the greatest or near the pits in the HAZ of the welds. None of the specimens exposed to the salt fog at 85 and 120°C (185 and 248°F) exhibited cracking because of the inability of salt deposits to deliquesce at high temperatures.

In May 1996, a hydrogen explosion occurred during welding operations while sealing the shield lid of a VSC-24 canister at the Point Beach plant. The VSC-24 is a welded canister. The root cause of hydrogen release was the interaction of boric acid with the zinc coating on the fuel basket. The boric acid was from the used-fuel pool and the zinc coating had been applied to the fuel basket to prevent corrosion. This event resulted in an extensive investigation by the NRC.

In April 1997, an 18-inch crack was found in the weld of the lid of a VSC canister in the ANO plant. As with the hydrogen explosion event, this event brought an immediate halt to all fuel-loading operations and resulted in an extensive investigation by the NRC.

IV.M3.3 References

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10 CFR Part 72, Appendix G, Quality Assurance Criteria for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste, Office of the Federal Register, National Archives and Records Administration, as amended June 9, 2008.

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NRC Confirmatory Action Letter 97-7-001, Technical Evaluation, U.S. Nuclear Regulatory Commission, Office of Nuclear Material Safety and Safeguards, Spent Fuel Project Office (ML060940604), 1998.

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NRC Interim Staff Guidance 5, Confinement Evaluation, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, May 1999.

NRC Interim Staff Guidance 15, Materials Evaluation, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, January 2001.

NRC Interim Staff Guidance 18, The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, Oct. 2008.

NUREG-7030, Atmospheric Stress Corrosion Cracking Susceptibility of Welded and Unwelded 304, 304L, and 316L Austenitic Stainless Steels Commonly Used for Dry Cask Storage Containers Exposed to Marine Environments, U.S. Nuclear Regulatory Commission, Washington, DC, October 2010.

Request for Additional Information for Renewal Application for Special Nuclear Materials License for the Calvert Cliffs Site Specific Independent Spent Fuel Storage Installation (ML103540592), U.S. Nuclear Regulatory Commission, Washington, DC, December 16, 2010.

Request for Additional Information for Renewal Application for Special Nuclear Materials License for the Calvert Cliffs Site Specific Independent Spent Fuel Storage Installation (ML111180260), U.S. Nuclear Regulatory Commission, Washington, DC, April 28, 2011.

Safety Evaluation Report, Docket No. 72-2, Surry Independent Spent Fuel Storage Installation License No. SNM-2501 License Renewal, U.S. Nuclear Regulatory Commission, Washington, DC, 2005.

Safety Evaluation Report, Docket No. 72-09, U.S. Department of Energy Fort St. Vrain Independent Spent Fuel Storage Installation License No. SNM-2504 License Renewal, U.S. Nuclear Regulatory Commission, Washington, DC, 2011.

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Safety Evaluation Report, Transnuclear, Inc., NUHOMS HD Horizontal Modular Systems for Irradiated Nuclear Fuel, Docket No. 72-1030, Amendment No. 1, U.S. Nuclear Regulatory Commission, Washington, DC, 2004.

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IV.M4 Bolted Canister Seal and Leakage Monitoring Program

IV.M4.1 Program Description

The objective of the program is to manage the aging effects on the confinement boundary of a bolted canister [e.g., Transnuclear (TN) Metal casks and MC-10 casks]. These aging effects include loss of material due to corrosion, loss of sealing forces due to stress relaxation and creep of the metallic O-rings, corrosion and loss of preload of the closure bolts, and SCC of the penetration welds. The program includes an overpressure leakage monitoring system for continuous monitoring of the pressure between the metallic seal assemblies in the TN casks and inside the cask cavity in the MC-10 casks. A low-pressure alarm is triggered when the pressure reaches a predetermined threshold. The continuous pressure monitoring ensures timely detection of aging effects in the confinement boundary so appropriate corrective actions can be taken to maintain safe storage conditions of the DCCS.

The program also includes periodic surveillance and maintenance of the overpressure leakage monitoring system and the associated instrumentation per facility specification to meet requirements of 10 CFR 72.122(h)(4), 10 CFR 72.122(i) and 10 CFR 72.128(a)(1).

Code of Federal Regulations 10 CFR 72.122(h)(4) specifies that storage confinement systems must have the capability for continuous monitoring, such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions. In addition, 10 CFR 72.122(i) specifies that instrumentation systems for dry storage casks must be provided in accordance with cask design requirements to monitor conditions that are important to safety over anticipated ranges for normal and off-normal conditions.

ISG- 5 Rev.1, “Confinement Evaluation,” provides guidance for evaluating the design and analysis of the proposed cask confinement system for normal, off-normal, and accident conditions. Acceptance criteria IV(1, 3, and 4) of ISG-5, Rev. 1, state the following:

> The cask design must provide redundant sealing of the confinement boundary. Typically this means that field closures of the confinement boundary must either have two seal welds or two metallic O-ring seals.
>
> The applicant should describe the proposed monitoring capability and/or surveillance plans for mechanical closure seals. To show compliance with 10 CFR 72.122 (h) (4), cask vendors have proposed, and the staff has accepted, routine surveillance programs and active instrumentation to meet the continuous monitoring requirement.
>
> The applicant must specify the maximum allowed leakage rates for the total primary confinement boundary and redundant seals. ……However, this is unnecessary for storage casks having closure lids that are designed and tested to be “leak tight” as defined in “American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials,” ANSI N14.5-1997.

In the Transnuclear Metal casks, an overpressure leakage monitoring system provides continuous monitoring of pressure in the region between the redundant metallic seal assemblies, which is pressurized with a non-reactive gas to a pressure greater than the helium pressure in the cask cavity. A decrease in pressure indicates that the non-reactive gas is leaking either into the cask cavity or into the atmosphere because of degradation of the metallic seal assemblies or the lid penetration welds. Therefore, for TN-32 casks, the applicant does not have to specify the maximum allowed leakage rate because leakage of radioactive contents through the seals is not a credible event. An overpressure leakage monitoring system in MC-10 casks provides continuous monitoring of pressure inside the cask cavity and a decrease in pressure indicates a leakage from the cask cavity through the seals or lid welds, and the applicant needs to specify the maximum allowed leakage rate for MC-10 casks as required by ISG-5, Rev. 1.

IV.M4.2 Evaluation and Technical Basis

1. ***Scope of Program:*** This AMP is used for managing the aging effects of the bolted canisters. These aging effects include loss of material due to corrosion, loss of sealing forces due to stress relaxation and creep of the metallic O-rings, SCC of the penetration welds, and loss of preload of the closure bolts. The specific components and systems that are typically managed by this AMP include shield lid, primary lid, closure lid, penetration steel covers, O-ring assemblies, and associated bolts and penetration welds. The program consists of continuous monitoring of the pressure and leakage rate between the metallic seal assemblies to manage aging effects. The program also includes periodic surveillance and maintenance of the overpressure leakage monitoring system and the associated instrumentation.

2. ***Preventive Actions:*** The overpressure leakage monitoring system is periodically checked per facility surveillance requirements to meet requirements of 10 CFR 72.122 (h)(4) and 10 CFR 72.122 (i). This periodic surveillance ensures proper functioning of the overpressure leakage monitoring system. Proper functioning of this system prevents the loss of intended function of the sealing components in the confinement boundary of the bolted canister.

3. ***Parameters Monitored or Inspected:*** The program monitors the pressure and leak rate of the non-reactive cover gas between the metallic seal assemblies to verify the integrity of the seal assemblies in the bolted canister.

4. ***Detection of Aging Effects:*** The overpressure leakage monitoring system continuously monitors the pressure and leak rate between the seal assemblies with a low-pressure alarm. A decrease in pressure between these seals indicates that the non-reactive gas is leaking either to the cask or to the atmosphere. Once the maximum allowable leak rate is reached, the leakage monitoring system alarm will be triggered. Continuous monitoring of the pressure and leak rate between the seal assemblies with a low-pressure alarm provides a means for early detection of the aging effects in the seal assemblies and penetration welds.

   The leakage monitoring system and the associated instrumentation are periodically checked per facility surveillance requirements to meet requirements of 10 CFR 72.122(h)(4) and 10 CFR 72.122(i). The condition monitoring thresholds should be periodically verified for the correct set point. A properly maintained overpressure leakage monitoring system ensures timely detection of aging effects.

5. ***Monitoring and Trending:*** The pressure level and leak rate of the non-reactive cover gas between the metallic seal assemblies are monitored continuously. The pressure and leakage rate data are trended to provide early detection of aging effects and to indicate when corrective action needs to be taken to maintain safe storage conditions, as required in 10 CFR 72.122(h)(4).

6. ***Acceptance Criteria:*** The maximum allowable leakage rates for the total confinement boundary and redundant seals, including the leakage rate of each seal, are specified in the facility Technical Specifications. The acceptance criterion for the pressure monitoring is the absence of an alarmed condition. The facility's Technical Specifications contain pressure monitoring alarm response procedures that include criteria and specifications for corrective actions and response.

   For defective penetration welds, the cracks are evaluated in accordance with ASME Code Section XI, IWB-3100, by comparing inspection results with the acceptance standards of IWB-3400 and IWB-3500. If the crack is unacceptable, further flaw evaluation is performed in accordance with ASME Code Section XI IWA-4422.1 to determine whether the flaw is acceptable for continued operation until the next inspection, or the weld is repaired in accordance with ASME Code Section XI IWA-4000.

7. ***Corrective Actions:*** Once the low-pressure alarm is triggered, a root-cause analysis of the pressure leakage should be performed and an engineering evaluation conducted to determine whether the degradation of the seal assemblies should be promptly corrected or evaluated as acceptable.

   Corrective actions include repair and replacement of the defective metallic seals. Corrective actions may also include periodic replacement of the metallic O-rings. The replacement frequency is estimated by correlating loss of sealing forces over time with leakage tightness, based on test data for the O-ring materials and the compartment configuration of the cover gas, as recommended in NUREG/CR-7116 *[Sindelar 2011]*. The defective welds are repaired in accordance with the ASME code and cask design requirements.

8. ***Confirmation Process*:** Site QA procedures, review and approval processes, and administrative controls are implemented according to the requirements of 10 CFR Part 72, Appendix G. As discussed in the appendix to this report, the requirements of 10 CFR Part 72, Appendix G, are acceptable to address the confirmation process, and administrative controls.

9. ***Administrative Controls*:** See element 8, above.

10. ***Operating Experience:*** Helium leakage in the TN-68 bolted canister at Peach Bottom was detected in October 2010. There was no loss of confinement capability. The root cause was a manufacturing defect in the weld that provides sealing of the drilled interseal passageway associated with the drain port penetration. The defective welds were repaired in accordance with the ASME Code and cask design requirements. Corrosion of the TN-32 lid bolts and outer metallic lid seals have been observed in the

Surry ISFSI owing to external water intrusion in the vicinity of the lid bolts and outer metallic seals.

IV.M4.3 References

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR Part 72, Appendix G, Quality Assurance Criteria for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste, Office of the Federal Register, National Archives and Records Administration, as amended June 9, 2008.

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Sindelar, R. L., et al., Materials Aging Issues and Aging Management for Extended Storage and Transportation of Spent Nuclear Fuel, NUREG/CR-7116, Savannah River National Laboratory, Aiken, SC, November 2011.

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IV.M5 Canister Structural and Functional Integrity Monitoring Program

IV.M5.1 Program Description

The objective of the program is to manage the aging effects of degradation of heat transfer, radiation shield, criticality control, confinement boundary, or structural support functions of the internals of a dry storage canister/cask (DSC), multi-purpose canister/cask (MPC), multi-assembly sealed basket (MSB), or fuel basket assembly due to corrosion, creep, distortion, cracking, peeling of laminates, etc. caused by extended exposure to high temperature and radiation. The regulations for the used-fuel dry cask storage designs , as delineated in 10 CFR 72, have the following common safety objectives: (1) ensure that the doses are below the limits prescribed in the regulations, (2) maintain subcriticality under all credible conditions of storage and transportation, (3) ensure that there is adequate confinement and containment of the used fuel under all credible conditions of storage and transportation, and (4) ensure that the used fuel is readily retrievable from the storage systems. In addition, 10 CFR 72.44 requires that the technical specifications of the used-fuel storage cask (also called canister) include functional and operating limits, monitoring instruments, and limiting control settings to protect the integrity of the stored fuel and to guard against the uncontrolled release of radioactive materials. The limiting conditions are the lowest functional capability or performance levels of equipment required for safe operation. 10 CFR 72.44(c) also requires that the storage canister technical specification include surveillance requirements for inspection, monitoring, testing, and calibration activities to confirm that operation of the canister is within the required functional and operating limits and that the limiting conditions for safe storage are met.

The Canister Structural and Functional Integrity Monitoring Program consists of:

(a) An assessment of the storage canisters/casks to verify that they were designed, fabricated, erected, and tested in accordance with the guidance and recommendations of the applicable NRC ISG documents to establish the condition of the (i) fuel cladding, (ii) canister/cask confinement, and (iii) retrievability of the used fuel assemblies.

(b) Based on the results of the assessment of the canister/cask design, fabrication, and testing, a site-specific monitoring program to detect any degradation of the functional and structural integrity of the canister internals, and to ensure that the operation of the storage canister is within the required functional and operating limits.

The applicable ISGs associated with the following three aspects of the various DSC designs include:

1. *Fuel Cladding Considerations*: ISG-11, Cladding Considerations for the Transportation and Storage of Spent Fuel.
2. *Canister/Cask Confinement*: ISG-5, Confinement Evaluation; ISG-15, Materials Evaluation; ISG-18, The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage; and ISG-25, Pressure and Helium Leakage Testing of the Confinement Boundary of Spent Fuel Dry Storage Systems.
3. *Retrievability*: ISG-2, Fuel Retrievability.

The specific guidance and recommendations of these ISGs are discussed further in program element #2, Preventive Actions. The monitoring program tracks temperatures and radiation levels at selected locations such as canister surfaces close to the fuel assemblies and air inlets and outlets. Large, abrupt changes in levels of temperature and/or radiation could indicate degradation of fuel cladding, degradation of neutron-absorbing and gamma-shielding materials, helium leakage, or a breach in the confinement boundary of the canister.

IV.M5.2 Evaluation and Technical Basis

1. ***Scope of Program:*** This program applies to all DSCs that are sealed at both ends either by welding and/or bolting, and perform the function of confinement for used fuel. The various storage canisters include DSCs, MPCs, MSBs, and fuel basket assemblies. The program consists of monitoring, preferably continuous, of temperatures (including ambient temperature) and radiation levels at selected locations in the ISFSI. The temperature and radiation monitoring should cover at least a representative sample of the storage units at a site. This monitoring should be conducted in accordance with the guidance provided in 10 CFR 122(h)(4).

2. ***Preventive Actions:*** This AMP consists of an assessment of the various designs of storage canisters or casks to verify that they were designed, fabricated, erected, and tested in accordance with the guidance and recommendations of the applicable NRC ISG documents. Following guidance and recommendations in these ISG documents, listed below, is considered to be prevention or mitigation actions that provide assurance that the structural and functional integrity of the canister internals will be maintained during the period of extended operation.

   <u>Fuel Cladding Considerations:</u>

   ISG-11, "*Cladding Considerations for the Transportation and Storage of Spent Fuel,*" focuses on the acceptance criteria needed to provide reasonable assurance that commercial used fuel is maintained in the configuration that is analyzed in the SARs for used-fuel storage. To assure integrity of the cladding material, the following criteria should be met:

   1. For all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400°C (752°F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad). However, for low-burnup fuel, a higher short-term temperature limit may be used if the applicant can show by calculation that the best-estimate cladding hoop stress is equal to or less than 90 MPa (13,053 psi) for the temperature limit proposed.
   2. During loading operations, repeated thermal cycling (repeated heatup/cool-down cycles) may occur but should be limited to fewer than 10 cycles, with cladding temperature variations that are less than 65°C (117°F) each.
   3. For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).

High burnup fuel (i.e., fuel with burnups generally exceeding 45 GWd/MTU) may have cladding walls that have become relatively thin from in-reactor formation of oxides or zirconium hydride. For design basis accidents, where the structure integrity of the cladding is evaluated, the applicant should specify the maximum cladding oxide thickness and the expected thickness of the hydride layer (or rim). Cladding stress calculations should use an effective cladding thickness that is reduced by those amounts. The reviewer should verify that the applicant has used a value of cladding oxide thickness that is justified by the use of oxide thickness measurements, computer codes validated using experimentally measured oxide thickness data, or other means that the staff finds appropriate. Note that oxidation may not be of a uniform thickness along the axial length of the fuel rods.

ISG-11 states

"Based on staff's evaluation, it is expected that fuel assemblies with burnups less than 45 GWd/MTU are not likely to have a significant amount of hydride reorientation due to limited hydride content. Further, most of the low burnup fuel has hoop stresses below 90 MPa. Even if hydride reorientation occurred during storage, the network of reoriented hydrides is not expected to be extensive enough in low burnup fuel to cause fuel rod failures." (2nd paragraph, page 3)

"The staff believes that this guidance will allow all commercial spent fuel that is currently licensed by the Nuclear Regulatory Commission (NRC) for commercial power plant operations to be stored in accordance with the regulations contained in 10 CFR Part 72. However, cask vendors' requests for storage of spent fuel with burnup levels in excess of those levels licensed by the Office of Nuclear Reactor Regulation (NRR), or for cladding materials not licensed by NRR, may require additional justification by the applicant." (4th paragraph, page 3)

"The staff is currently reevaluating the technical basis for the transportation of spent fuel including assemblies with average assembly burnups exceeding 45 GWd/MTU. The staff is reviewing data and technical reports to further understand the mechanical and fracture toughness properties of spent fuel cladding in relation to the transportation of high burnup fuel under 10 CFR 71.55. Therefore, until further guidance is developed, the transportation of high burnup commercial spent fuel will be handled on a case-by-case basis using criteria given by 10 CFR 71.55, 10 CFR 71.43(f), and 10 CFR 71.51." (2nd paragraph, page 1)

"For high brnup cladding materials, cladding performance during hypothetical accident conditions of transport will require further information on the impact properties. Data is not currently available." (3rd paragraph, page 2)

The NRC has issued Requests for Additional Information (RAIs) for review of applications for revision to CoC of dry cask storage systems and the following RAI appeared repeatedly for high burnup fuels:

"Show that the mechanical properties calculated using the Geelhood and Beyer correlations apply to cladding radial hydrides. Compare the ductility of cladding with radial hydrides with the stress imposed during a side drop, i.e., hoop plus crush stress. Most of the DSCs have either been approved for storage of high burnup fuel or are asking for approval to transport high burnup fuel. The presence of radial hydrides is expected, since 1) the cladding will have had to undergo a drying cycle, and 2) most cladding, other than M5, has both hydrogen content >200 wppm, with hoop stresses increased by the larger increase in fission gases at high burnup."

Canister/Cask Confinement:

ISG-15, "*Materials Evaluation,*" provides specific guidance for evaluating material-related issues for used-fuel storage canisters under normal, off-normal, and accident conditions. To ensure an adequate margin of safety in the design basis of the storage canister, the evaluation should provide reasonable assurance that

- The physical, chemical, and mechanical properties of components important to safety meet their service requirements.
- Materials for components important to safety have sufficient requirements to control the quality of the raw material, handling, and fabrication and test activities.
- Materials for components important to safety are selected to accommodate the effects of, and to be compatible with, the ISFSI or MRS site characteristics and environmental conditions associated with normal, off-normal and accident conditions.
- The spent fuel cladding is protected from gross rupture and from conditions that could lead to fuel redistribution.
- DCSS must be designed to allow ready retrieval of spent fuel.

ISG-5, "*Confinement Evaluation,*" provides guidance for evaluating the design and analysis of the proposed cask confinement system for normal, off-normal, and accident conditions. To meet the regulatory requirements for radiation dose limits prescribed in 10 CFR 72, the confinement evaluation should ensure that the DCSS design satisfies the following acceptance criteria:

1. The cask design must provide redundant sealing of the confinement boundary. Typically, this means that field closure of the confinement boundary must have either two seal welds or two metallic O-ring seals.
2. To meet the regulatory requirements and general design criteria of Chapter 2 of the SRP (NUREG-1536), the design and construction of the confinement boundary should be in accordance with ASME Boiler and Pressure Vessel Code Section III, Subsection NB or NC, which defines the standards for all aspects of construction, including materials, design, fabrication, examination, testing, inspection, and certification of the components.
3. The applicant must specify the maximum allowed leakage rates for the total primary confinement boundary and redundant seals. However, this is unnecessary for storage casks having closure lids that are designed and tested to be "leak tight" as defined in "American National Standard for Leakage Tests

on Packages for Shipment of Radioactive Materials," ANSI N14.5-1997. The applicant's leakage analysis must demonstrate that an inert atmosphere will be maintained within the cask during the storage lifetime.

4. The applicant should describe the proposed monitoring capability and/or surveillance plans for mechanical closure seals. However, for casks with welded closures, the NRC staff has determined that no closure monitoring system is required. To show compliance with 10 CFR 72.122(h)(4), cask vendors have proposed, and the staff has accepted, routine surveillance programs and active instrumentation to meet the continuous monitoring requirements.
5. The cask must provide a non-reactive environment to protect fuel assemblies against fuel cladding degradation, which might otherwise lead to gross rupture. A non-reactive environment typically includes drying, evacuating air and water vapor, and backfilling with a non-reactive cover gas (such as helium).

ISG-25, "*Pressure and Helium Leakage Testing of the Confinement Boundary of Spent Fuel Dry Storage Systems,*" provides supplemental guidance for evaluating the helium leakage testing and ASME Code-required hydrostatic/pneumatic pressure testing that is specified for the DCSS confinement boundary. The DCSS confinement boundary for welded canisters consists of the canister shell, bottom plate, top lid, vent and drain port cover plates, and the interconnecting welds. Typically, the shell-to-bottom-plate and shell seam welds are prepared in the fabrication shop, while the lid-to-shell and vent port cover plate welds are prepared in the field after fuel loading. The confinement boundary is relied upon to (1) confine radioactive materials to a degree sufficient to meet 10 CFR 72 dose limits and (2) maintain a pressurized helium environment to ensure an inert atmosphere and, in some designs, to ensure adequate cooling of the used nuclear fuel. As discussed above in ISG-5(3), the applicant must specify the maximum allowed leakage rates for the total primary confinement boundary, and the allowable leakage rate must be evaluated for its radiological consequences and ability to maintain an inert atmosphere within the cask. If the entire confinement boundary is tested to be leak-tight in accordance with ANSI-N14.5 (i.e. 1.0×10^{-7} ref. cm^3/sec) and the canister lid-to-shell weld conforms to the criteria of ISG-18, then leakage is not considered credible and effluents are not required to be considered in a confinement dose analyses.

To maintain and ensure the integrity of the confinement boundary, it is examined and tested by a combination of methods, including helium leakage testing, ASME pressure testing, radiographic examination, ultrasonic examination, magnetic particle examination and /or PT examination. These tests verify that the boundary is free of cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce its structural integrity and confinement effectiveness. The volumetric and surface examinations of welds ensure that the welds comply geometrically with the design requirements, but can only detect flaws of a minimum size. The ASME Code pressure test provides additional assurance that the component has been properly fabricated by stressing the component to a minimum Code-required loading. The helium leakage test ensures that there are no flaws or leak paths that could result in significant release of the helium environment and radioactive contents. The following acceptance examinations and tests should be verified:

1. The examination of confinement welds should be performed in accordance with the guidance of ISG-15 and other acceptable practices, as appropriate.
2. The entire confinement boundary should be pressure-tested hydrostatically or pneumatically to 125 or 110 percent of the design pressure, respectively. The test pressure should be maintained for a minimum of 10 minutes prior to initiation of a visual examination for leakage, per the ASME Code. Also, if visual examination of inaccessible portions of the canister welds cannot be performed during the field ASME Code hydrostatic test, the results from the shop helium leakage test applied under ANSI-N14.5 standards are considered acceptable. However, the exception and its basis should be listed in the table of ASME code exceptions in the CoC.
3. A shop helium leakage test must be performed in accordance with ANSI N14.5 testing standards to demonstrate that the entire DCSS confinement body is free of defects that could lead to a leakage rate greater than the allowable design basis leakage rate specified in the confinement analyses. The requirements for the helium leakage test should be specified in the CoC to meet the requirements of 10 CFR 72.236(j) and (l).
4. The lid-to-shell welds and vent ports should be fabricated and helium leakage tested in accordance with the guidance of ISG-18, as applicable. The staff should note that only lid-to-shell welds are within the scope of leak-testing exceptions specified in ISG-18.
5. For bolted-closure casks, the entire confinement boundary should be similarly helium leak tested and pressure tested. The confinement boundary should be tested at the fabrication shop, with only a leakage test performed on the bolted lid closure seals (including drain and vent port seals) tested in-field by the DCSS user.

ISG-18, “The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage,” addresses the design and testing of the various closure welds, or “lid welds,” associated with the redundant closure of all-welded austenitic stainless steel canisters. 10 CFR 72.236(e) states that “the spent-fuel storage cask must be designed to provide redundant sealing of confinement systems.” For a welded canister design, the NRC staff has accepted closure designs employing redundant lids or covers, each with independent field welds. Thus, a potential leak path would have to sequentially breach two independent welds before the confinement system would be compromised.

As discussed above, the entire confinement boundary of storage canisters is examined and helium leak tested to meet the requirements of 10 CFR 72. For continued confinement effectiveness during storage, the bolted-closure canisters incorporate a helium monitoring system, whereas the welded-closure canisters rely on weld integrity. Consequently, at least one of the redundant welded closures must be helium leakage tested per the method of ANSI N14.5, except when the large, multi-pass weld joining an austenitic stainless steel canister shell to the canister lid is prepared and examined in accordance with guidance provided in ISG-15 and meets the Helium Leakage Test—Large Welded Exception Criteria of ISG-18R1. A “multi-pass weld” represents a weld with three or more individual layers of weld metal, in which each layer may be composed of a single weld bead or several adjacent weld beads of common thickness. A minimum of three layers minimizes the probability of a weld flaw propagating through the weld layers, resulting in a leakage path. Thus, a multi-pass weld provides reasonable assurance against the existence of a flaw of sufficient size to impair the structural strength or confinement capability of the weld. Therefore, helium leakage testing of such multi-pass welds is unnecessary, provided the weld is prepared and examined in accordance with ISG-15 and is in compliance with the technical review guidance of ISG-18.

<u>Retrievability:</u>

ISG-2, "Fuel Retrievability," provides guidance for determining whether the storage canister design satisfies the requirement of 10 CFR 72.122(l) that "storage systems must be designed to allow ready retrieval of used fuel . . . for further processing or disposal," and 10 CFR 72.236(m) that ". . . consideration should be given to compatibility with removal of the stored used fuel from a reactor site, transportation, and ultimate disposal by the Department of Energy." Thus, this guidance defines ready retrieval of used fuel as the ability to move the canister containing the fuel to either a transportation package or a location where the fuel can be removed, while maintaining the ability to handle individual fuel assemblies or canned fuel assemblies by normal means.

3. ***Parameters Monitored or Inspected:*** This program monitors temperatures (including ambient temperature) and radiation levels at selected location in the ISFSI. Increased levels of temperature and/or radiation dose rate could indicate degradation of cladding, degradation of neutron-absorbing and gamma-shielding materials, helium leakage, or a breach in the confinement boundary of the canister. Both neutron and gamma dose rates (mrem/h) and the total dose rate (mrem/h) are monitored for irradiated fuel actinides and fission-product isotopes such as Kr, Cs and I. The extent and frequency of monitoring, and sampling of storage canisters/casks included in the monitoring program, are based on results of the assessment of the design, fabrication, and testing of the canister/cask prior to long-term storage.

   The program also includes, but is not to be limited to, periodic calibration of electronic circuitry associated with the temperature monitoring devices, as specified in 10 CFR 72.44(c)(3)(ii), and periodic evaluation of temperature data sufficient to identify anomalous trends that could indicate degraded instrumentation or degradation in the cask system. Passive elements in the temperature and radiation measurement devices, such as sensing elements, should be periodically inspected to ensure that no degradation due to corrosion, wear, or cracking has occurred.

4. ***Detection of Aging Effects:*** On the basis of the results of the assessment of the canister/cask design, fabrication, and testing, the monitoring program and sample size are as follows: If the storage canisters/casks are in compliance with the guidelines of all the NRC ISG documents, the Canister Structural and Functional Integrity Monitoring Program includes temperature and radiation monitoring preferably continuously or at time intervals not exceeding 12 hours at the air outlet and other selected locations around the storage canister/cask, with provisions for more frequent measurements if

temperatures approach technical specification limits (i.e., within 10%). The monitoring program should cover at least a representative sample of storage canisters/casks based on their age and service history. All storage canisters/casks that do not meet the guidelines of the ISG-11 document should be included in the monitoring program.

If the canisters/casks do not meet some or all the guidelines of ISG-15, ISG-5, ISG-25, and ISG-18, the Canister Structural and Functional Integrity Monitoring Program should include continuous monitoring of temperature and radiation at the air outlet and other selected locations around the storage canister/cask. The sample size should be based on the results of the assessment of the canister/cask design, fabrication, testing, and service history, and may include all the storage units at the site.

This program includes measurement of gamma and neutron dose rates (mrem/h) and the total dose rate (mrem/h) at selected key locations such as cask top surface (lid center), cask side surface (e.g., peak burnup section of the fuel), and air inlet/outlet ducts. For concrete storage modules, representative locations are selected, such as the center of roof and side walls, and access doors. The measured dose rates are compared to the allowable dose rate limit at each location specified in the technical specifications. Note that the dose rate limit could be different, depending on location. For example, the dose rate limit for the cask top and side surfaces could be 100 and 200 mrem/h, respectively, and the dose rate limit for the air inlet and air outlet locations could be 350 and 100 mrem/h, respectively.

The program monitors both neutron and gamma dose rates and the total dose rate. Increased levels of neutron dose rate could indicate degradation of neutron-absorbing materials, and increased levels of gamma radiation dose rate could indicate degradation of shielding materials. They could also indicate a breach in the containment boundary of the DSC due to aging effects, such as cracking due to SCC or cyclic loads.

To ensure that the temperature and radiation monitoring devices will remain accurate during the long-term storage of the ISFSI, the devices should be periodically calibrated in accordance with plant QA requirements.

5. ***Monitoring and Trending:*** Trending should be established that could indicate either a problem with a cask or a faulty measurement device. Qualified personnel should periodically review the temperature and radiation data in accordance with 10 CFR 72.158. The data should be compared to baseline or predicted values and appropriate actions should be taken if any abnormal readings are noted. A review of temperature and radiation trends would detect an instrument problem long before there is an actual temperature problem involving the canister.

6. ***Acceptance Criteria:*** The measured temperature and radiation dose rates are compared to the established limits specified in the ISFSI technical specification for temperature and radiation dose rates (e.g., 195ºF and 100 mrem/h at the air outlet). Levels exceeding these limits or unexpected increases in levels should be investigated for potential degradation of the ISFSI components due to aging effects.

   When the operating limits for temperature and radiation dose rate are established in the technical specification, the surveillance limits for the recorded reading should be conservatively set such that there is sufficient margin between the two limits (i.e., operating limit and surveillance limit) to account for instrument uncertainties. Sufficient margins between the expected temperature and radiation dose rate readings and the calculated values in the ISFSI safety analyses should also be established.

7. ***Corrective Actions:*** Site QA procedures, review and approval processes, and administrative controls are implemented in accordance with the requirements of 10 CFR Part 72, Subpart G, and, for non-safety-related structures and components subject to an AMR, 10 CFR Part 50, Appendix B. As discussed in the appendix to this report, these requirements are found to be acceptable to address the corrective actions, confirmation process, and administrative controls.

8. ***Confirmation Process*****:** See element 7, above.

9. ***Administrative Controls*****:** See element 7, above.

10. ***Operating Experience*****:** Operating experience at existing ISFSI sites indicates that if there are multiple air outlets in a cask, measured temperatures can vary significantly between them because of wind and weather conditions. Any difference in temperature reading between the air outlets should be addressed in the ISFSI surveillance procedure.

    The NRC Information Notice 2011-10 identified thermal issues during loading of used fuel assemblies into a multipurpose canister within a transfer cask and vacuum drying. The operation was left unattended for the evening, and a cooling system, which circulated water in the annulus between the canister and transfer cask to keep cladding temperatures below allowable limits, was found to be inoperable the next morning. Although the thermal evaluation showed that cladding temperature limit was not exceeded during the absence of cooling, the NRC conducted a reactive team inspection at the utility site in September 2010, and the issues were also addressed during a design and quality assurance inspection of the cask vendor in October 2010. The NRC Regulatory Issue Summary 2006-22 discusses lessons learned from a dry cask storage campaign that included operational insight regarding the time limit established for the vacuum drying in the Technical Specification (TS) of the CoC of a dry cask design. A change in the sequence of operations that allowed the temperature of the fuel cladding to increase beyond the initial temperature of 215^{o}F, assumed in the basis of the SAR, would result in a shorter vacuum drying time than that specified in the TS.

IV.M5.3 References

10 CFR Part 72, Appendix G, Quality Assurance Criteria for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor Related Greater Than Class C Waste, Office of the Federal Register, National Archives and Records Administration, as amended June 9, 2008.

Duke Energy LLC, Oconee Nuclear Station Site-Specific Independent Spent Fuel Storage Installation Application for Renewed Site-Specific Material License (ML081280084), January 30, 2008.

NRC Information Notice 2011-10: Thermal Issues Identified during Loading of Spent Fuel Storage Casks, U.S. Nuclear Regulatory Commission, Washington, DC, May 2, 2011.

NRC Inspection Report Nos. 05000454/2010007; 05000455/2010007; and 07200068/2010002, Byron Station, Units 1 and 2; Reactive Team Inspection Involving Independent Spent Fuel Storage Installation Operations (ML103140226), U.S. Nuclear Regulatory Commission, Washington, DC, November 5, 2010.

NRC Inspection Report No. 72-1014/10-201 and Notice of Violations (ML110450157), U.S. Nuclear Regulatory Commission, Washington, DC, February 24, 2011.

NRC Interim Staff Guidance 2, Fuel Retrievability, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, Feb. 2010.

NRC Interim Staff Guidance 4, Cask Closure Weld Inspections, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, May 1999.

NRC Interim Staff Guidance 5, Confinement Evaluation, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, May 1999.

NRC Interim Staff Guidance 11, Cladding Considerations for the Transportation and Storage of Spent Fuel, Revision 3, U.S. Nuclear Regulatory Commission, Washington, DC, Nov. 2003.

NRC Interim Staff Guidance 15, Materials Evaluation, U.S. Nuclear Regulatory Commission, Washington, DC, January 2001.

NRC Interim Staff Guidance 18, The Design and Testing of Lid Welds on Austenitic Stainless Steel Canisters as the Confinement Boundary for Spent Fuel Storage, Revision 1, U.S. Nuclear Regulatory Commission, Washington, DC, Oct. 2008.

NRC Interim Staff Guidance 25, Pressure and Helium Leakage Testing of the Confinement Boundary of Spent Fuel Dry Storage Systems, U.S. Nuclear Regulatory Commission, Washington, DC, Aug. 2010.

NRC Regulatory Issue Summary 2006-22: Lessons Learned from Recent 10 CFR Part 72 Dry Cask Storage Campaign, U.S. Nuclear Regulatory Commission, Washington, DC, November 15, 2006.

Request for Additional Information for Review of the Model No. NUHOMS – MP197 Packaging, Docket No. 71-9302, U.S. Nuclear Regulatory Commission, Washington, DC, December 2, 2009.

Request for Additional Information for Review of the Model No. NUHOMS – MP197 Packaging, Docket No. 71-9302, U.S. Nuclear Regulatory Commission, Washington, DC, November 19, 2010.

Virginia Electric and Power Company, Surry Independent Spent Fuel Storage Installation (ISFSI) License Renewal Application, Docket No. 72-2, April 29, 2002.

V. APPLICATION OF AGING MANAGEMENT PROGRAMS

V.1 NUHOMS® Dry Spent-Fuel Storage

V.1.1 System Description

This section addresses the elements of the Standardized NUHOMS® horizontal cask system for dry storage of pressurized water reactor (PWR) or boiling water reactor (BWR) used nuclear fuel assemblies (Fig. V.1-1). The NUHOMS® system provides for the horizontal storage of used fuel in a DSC, which is placed in a concrete HSM. It can be installed at any reactor site or a new site where an ISFSI is required. Each NUHOMS® system model type is designated by NUHOMS-XXY. The two digits (XX) refer to the number of fuel assemblies stored in the DSC, and the character (Y) designates the type of fuel being stored, P for PWR or B for BWR. For some systems, a fourth character (T) is added, if applicable, to designate that the DSC is also intended for transportation in a 10 CFR 71-approved package. Also, two additional characters, HB, are added for systems that are used to store high-burnup fuels (e.g., NUHOMS-24PHB).

Figure V.1-1: NUHOMS® horizontal dry storage systems at San Onofre.

The NUHOMS® DSC is a welded canister that utilizes redundant multi-pass closure welds with no seal pressure monitoring in the system. After draining and drying, the canister is backfilled with helium to provide an inert environment.

The original system, NUHOMS-07P, was approved by the NRC in March 1986 for storage of seven used PWR fuel assemblies per DSC and HSM. The internal basket of the DSC for this system incorporates borated guide sleeves to ensure criticality safety during wet loading operations without credit for burnup (uranium deletion nonvolatile fission product buildup) or soluble boron. Later designs of the NUHOMS® system can hold 24 or 32 PWR fuels or 52 or 61 BWR fuels. Most of the standardized canister designs utilized borated guide sleeves to ensure criticality control during wet loading operations without credit for burnup or soluble boron. However, unlike these designs, no borated neutron-absorbing material is used in the standardized NUHOMS-24P basket design; it takes credit for burnup or soluble boron in the flooded DSC during wet loading or off-loading the fuel. The maximum heat load for the NUHOMS® DSCs is in the range of 18-24 kW.

The storage facility is mainly divided into two elements: the DSC and the HSM. The DSC (see Fig. V.1-2) is composed of three basic components: the internal basket assembly, the shell and the shielding plugs. They are described below.

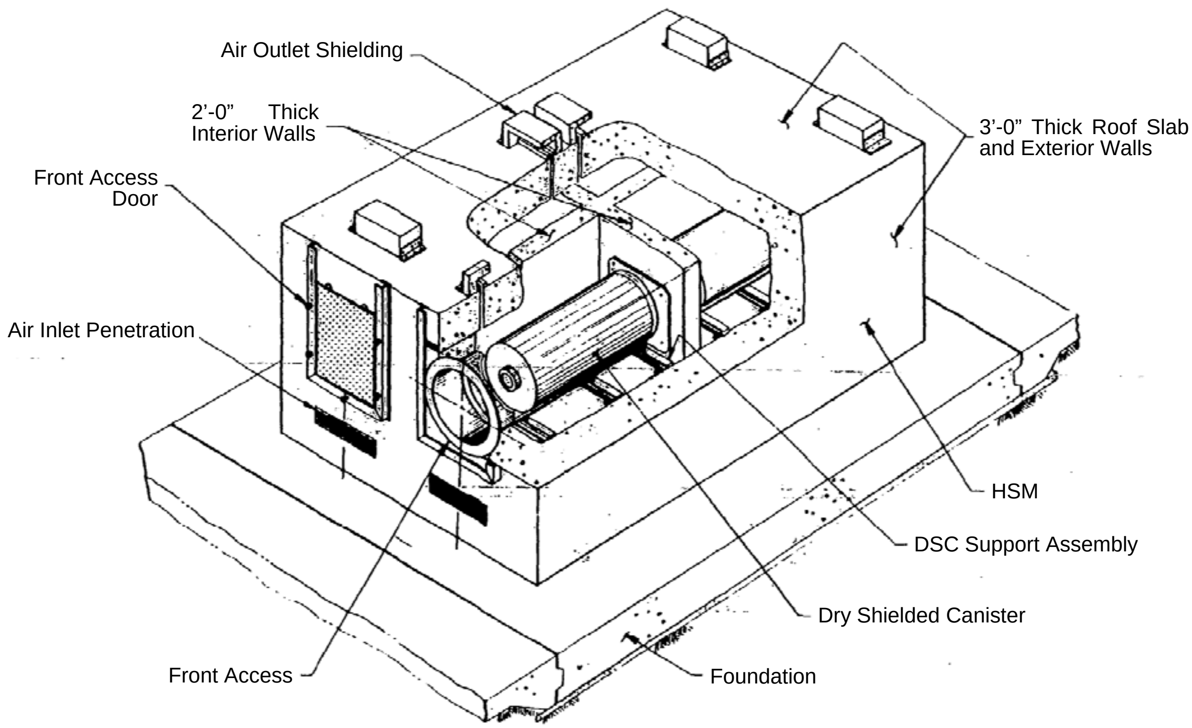


Figure V.1-2: A schematic of the NUHOMS® dry storage system.

V.1.1.1 HSM Description

The HSM is a low-profile prefabricated structure constructed from reinforced concrete and structural steel that provides protection for the DSC against tornado missiles and other potentially adverse natural phenomena, and also serves as the principal biological shield for the used fuel during storage. Major structures and systems of the HSM are the concrete structure, access door, DSC support assembly, and a ventilation system. However, the specific design details may vary for different ISFSIs. A schematic representation of the dry storage system is shown in Fig. V.1-2. The HSM is supported by a 3-ft-thick partially below-grade basemat (i.e., concrete pad). Each HSM is about 20 ft long, 9 ft wide, and 15 ft high, with 3-ft-thick concrete exterior walls and roof providing neutron and gamma shielding. The interior concrete walls between adjacent HSMs are 2 ft thick. Stainless steel heat-shielding plates are provided between the DSC and the HSM concrete to control the peak concrete temperatures and prevent concrete degradation during the worst design thermal conditions. The heat shield plates are anchored to the ceiling and walls; the details are shown in Fig. V.1-3. Each HSM contains one DSC and each DSC contains between 24 and 61 fuel assemblies.

The HSM is anchored to the foundation basemat to mitigate overturning and sliding effects under accident conditions such as seismic events, using dowel rods of a size and spacing consistent with the HSM wall vertical reinforcement. A concrete approach pad is adjacent to the basemat for

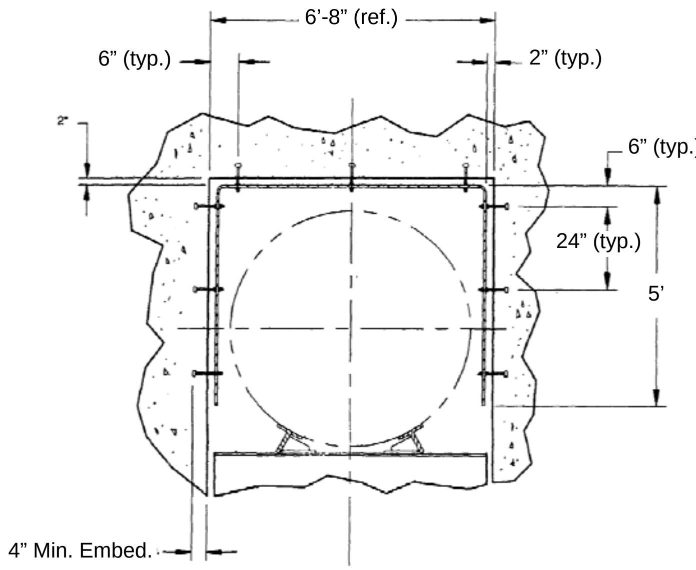


Figure V.1-3: Details of HSM heat shield

loading and retrieval of the DSCs. The approach pad is separated from the basemat by a construction joint. The differential settlement of the basemat and the concrete approach pad is limited to ensure proper retrieval of the DSC. The HSMs are typically built in units of 20 in a 2x10 array.

A shielded air inlet opening in the lower front wall of the structure and two shielded outlet vents in the roof, and associate pathways, are provided in each HSM for dissipation of the decay heat. The air flow diagram for a typical HSM design is shown in Fig. V.1-4. The ventilation air enters through the air inlet into a plenum formed by an interior shielding slab and a partial-height wall, and exits from two vertical and one horizontal opening in the plenum. The air flows around the DSC and exits through the exit vents in the HSM roof. Stainless steel inlet/outlet screens and frame are mounted at the air inlet and outlet openings of each HSM to prevent the entry of debris and birds/rodents that could compromise the heat removal capability of the HSM. Anchored concrete blocks are installed around the inlet and outlet vents for shielding purpose.

Each HSM has an access opening or docking flange in the front wall to accommodate transfer of DSCs from and into the shielded transfer cask. An embedded steel sleeve, with rail plates, is installed in the access opening to facilitate insertion and retrieval of the DSC into and from the HSM. The access opening is covered by a thick shielded access door. A steel frame attached to the front outside wall of the HSM supports the HSM door. In the early designs of HSMs, the 5-in-thick steel door contained 2-in-thick BISCO NS-3 shielding material. In the later designs, the 11-in-thick steel door contains 9.5-in concrete for shielding. The access door is typically sealed by welding the door to the steel door frame as confinement boundary after placement of the DSC in the HSM.

A supporting assembly, constructed from carbon steel structural product forms, supports the DSC inside the HSM. It consists of two rails supported at the front, rear and mid-length by steel beams supported by six legs; the three steel beams are also anchored to the sidewalls. Figure V.1-5 shows drawings of the side elevation and end view of the typical DSC support structure inside the HSM. In some HSM designs, the rails are supported at the front and rear by attaching to the front and rear walls, and mid-length by beam attached to the sidewalls. Stainless steel cover plates coated with a dry film lubricant are attached to the rails to provide a sliding surface (reduced friction) for DSC insertion and retrieval. In some designs, Nitronics 60 plates are welded to the cover plates because of this material's good high-temperature properties and resistance to oxidation, wear, and galling. Seismic restraints using steel plates or tubes are welded to the rear and front of the rails for retaining the DSC in place during seismic events. Threaded fasteners made from high-strength tempered steel are used for the DSC support assembly. A lightning protection system is also installed in the HSM. The system includes roof handrails, connectors, cable with lead shielding, and ground rods. Other items in the HSM include PVC pipes for drains and for electrical conduit, ladder and attachments, caulking, galvanized flashing, concrete nails, embedded steel plates and studs, expansion anchors, wedge anchors, and shell-type anchors.

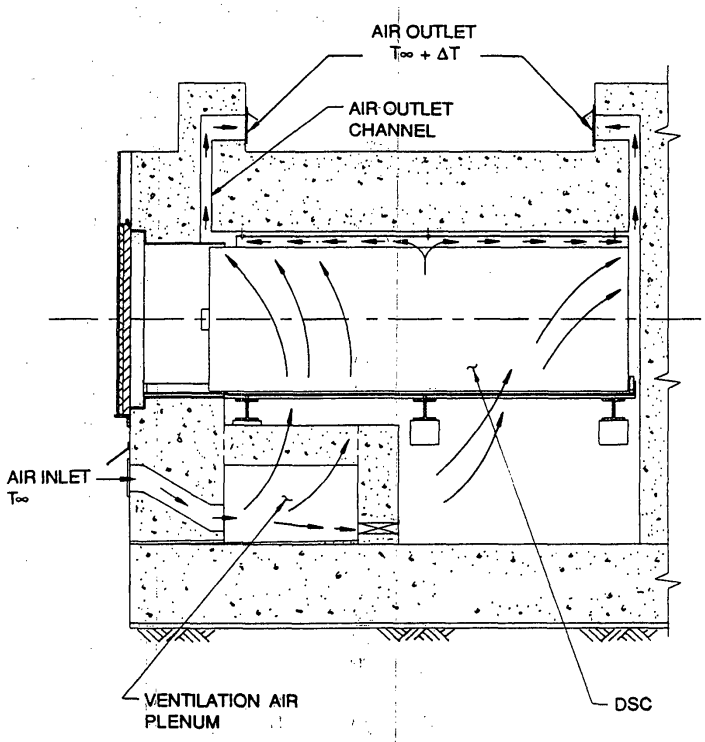


Figure V.1-4: Air flow diagram for a typical HSM design

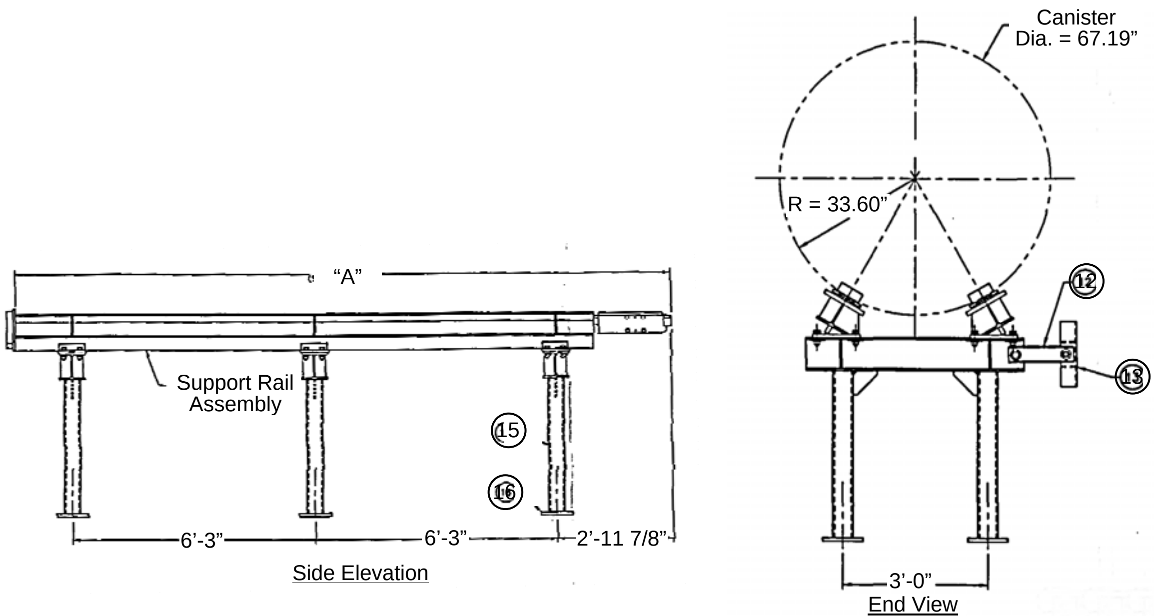
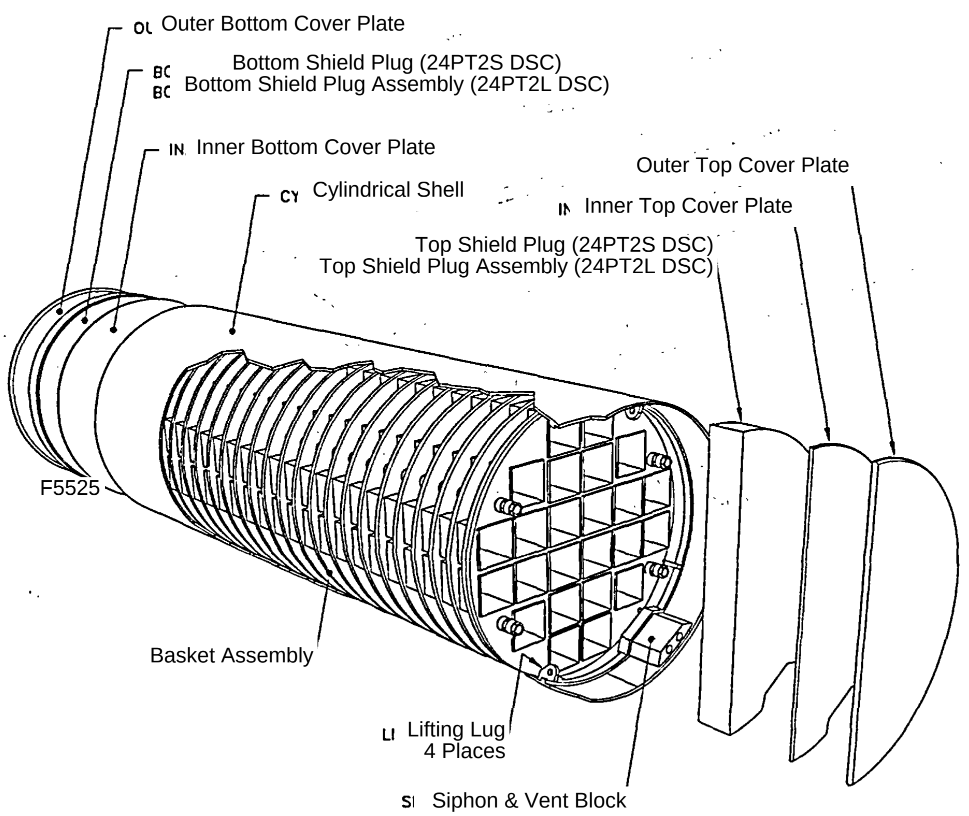


Figure V.1-5: Drawing showing the side elevation and end view of the DSC support structure.

The DSC is prevented from sliding longitudinally along the rail during a seismic event by axial seismic restraints. A steel tube member that seismically retains the DSC in place at the rear of the HSM is welded to each of the DSC support rails in the Phase 1 HSMs, whereas a steel plate is welded to each of the DSC support rails in the Phase 2 HSM design. After placement of the DSC, a removable seismic restraint is placed into slots in the access sleeve at the front of the HSM.

V.1.1.2 DSC Basket Assembly and Shell

The basket assembly is the key component of the DSC that provides structural support and criticality control for the fuel assemblies. It is an open structure consisting of square guide sleeves or guide tubes, circular spacer discs, and support rods. The guide sleeves surround each of the used-fuel assemblies. They are open at the ends and are inserted through the circular spacer discs, which provide lateral spacing and support for the fuel assemblies. The spacer discs are welded to the support rods, which run the length of the DSC interior. The various components of the different NUHOMS® DSC assembly designs are shown in Fig. V.1-6. In addition to the differences in the type and number of fuel assemblies stored in each canister, there are some minor differences in the configuration and design of the different canisters. Most of the canister designs utilize borated aluminum or boron carbide/aluminum alloy plate or BORAL composite neutron-absorbing material (poison) for necessary criticality control and heat conduction paths from the fuel assemblies to the canister shell. The NUHOMS-24P design, however, does not utilize borated guide sleeves for criticality safety.



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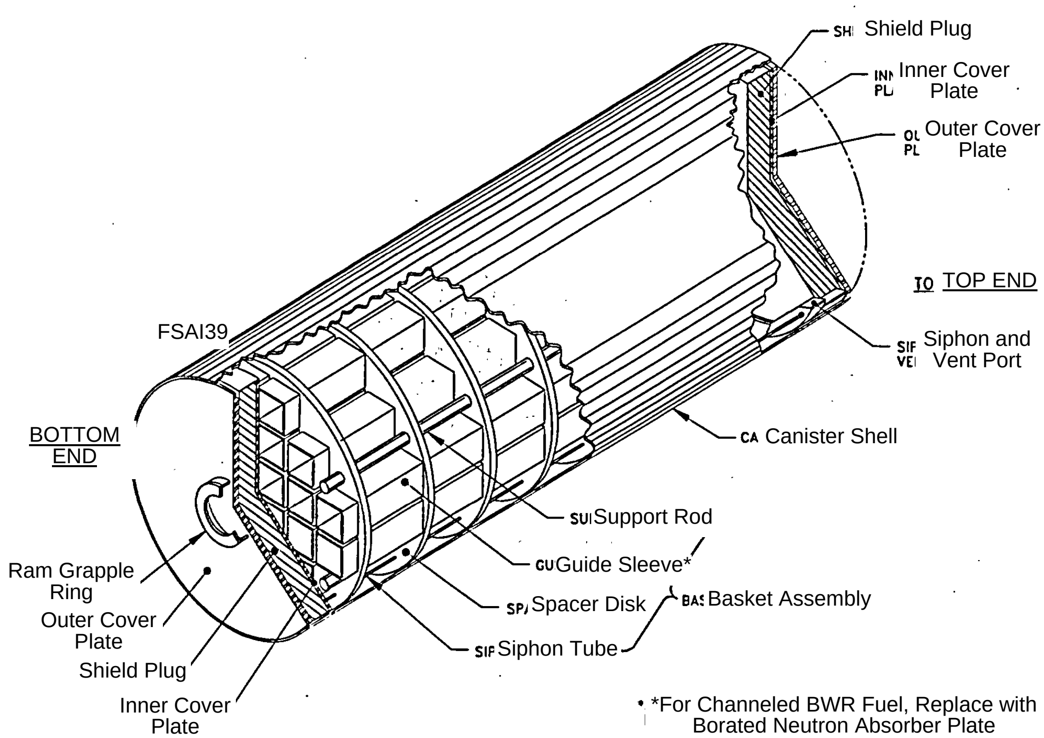
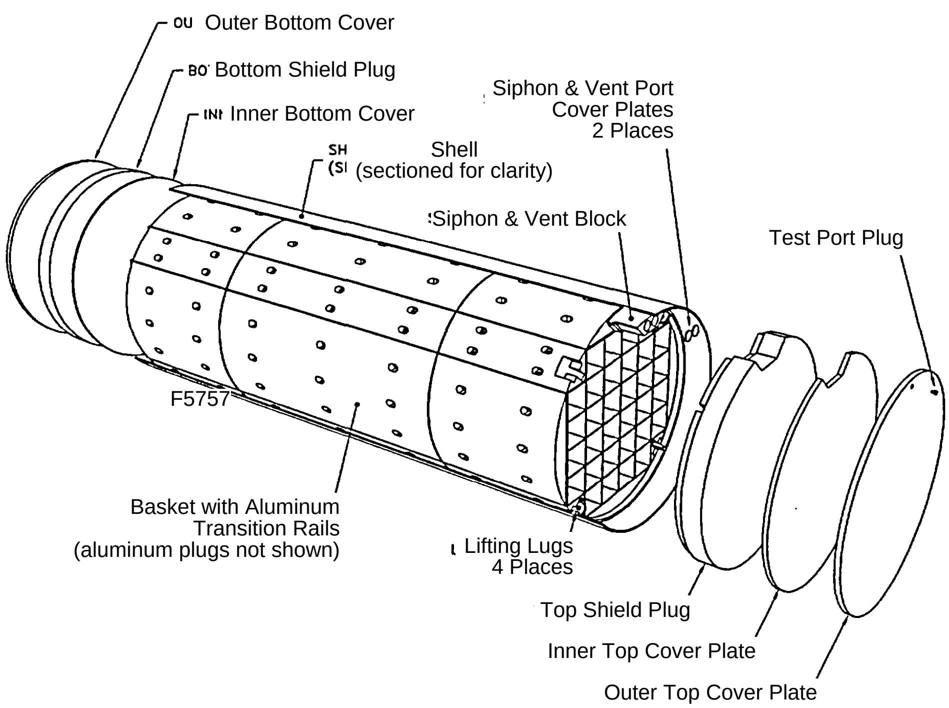
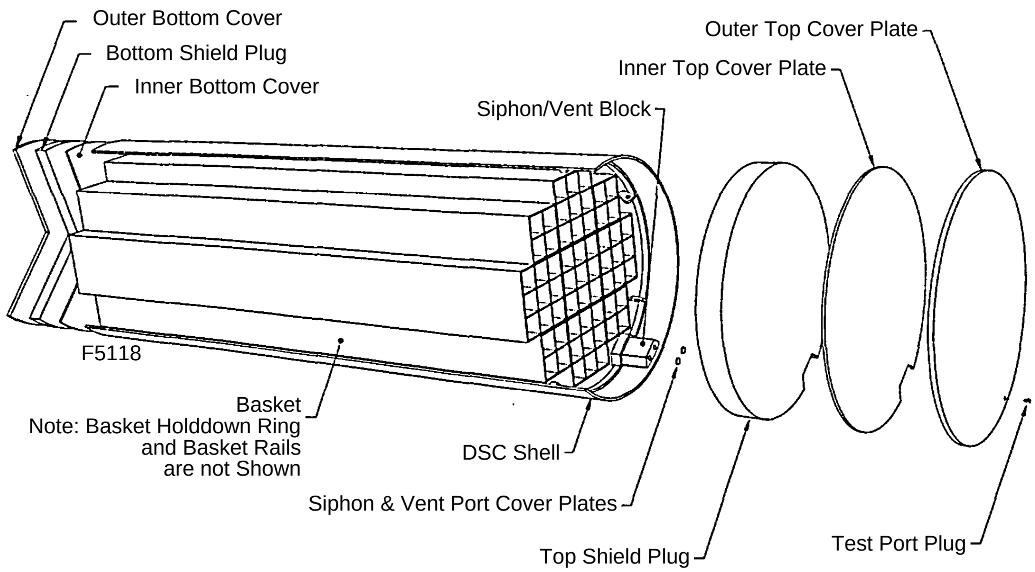


NUHOMS-24PHB



NUHOMS-32PT



NUHOMS-61BT

Figure V.1-6: Components of the various NUHOMS® DSC assembly designs.

Also, in the NUHOMS-24P canister design, square "over sleeves" are placed over the 12 interior guide sleeves at the ends (i.e., the open span between the 1st and 2nd spacer discs, and also between the 7th and 8th spacer discs). In the NUHOMS-32PT canister design, the space between the guide sleeve grid assembly and the DSC shell is bridged by transition rail structure. The transition rail consists of solid aluminum segments that transfer mechanical loads to the DSC shell and support the fuel assembly grid, and provide a thermal conduction path from the fuel assemblies to the canister shell wall. The fuel assembly grid in the NUHOMS-61BT canister consists of five compartments of nine guide sleeves/tubes each and four compartments of four guide sleeves/tubes each. Welded boxes that also retain the neutron absorber plates between the compartments hold the four or nine sleeve compartments together. The five 3x3 compartments are arranged in a cross and the 2x2 compartments are located at the four corners. These fuel compartments are held in place by basket rails and hold-down rings. Also, damaged fuel is stored in the four 2x2 corner compartments. End caps are installed on both ends of the 16 guide sleeves that hold damaged fuel assemblies.

The majority of the components in the DSC basket assembly are made of stainless steel and a few components are made of aluminum alloys. In some designs, carbon steel with an aluminum coating is also used for spacer discs and support rods. The DSC shell (body) is a stainless steel cylinder that consists of a rolled and welded plate that is 12.7–15.9 mm (0.5–0.625 inch) thick. The DSC shell serves as a portion of the confinement barrier.

V.1.1.3 Shell Shielding Plugs

Shielding plugs (also called lids or covers) are provided at both ends of each DSC for biological shielding. The bottom end shielding consists of stainless steel inner plate and outer plate, which encapsulates the lead shielding material in the "long-cavity design." The long-cavity DSC design is used to accommodate used-fuel assemblies with control components. The "short-cavity design" uses a thick carbon steel plate for shielding and is utilized to store used fuel without control components.

The top shielding plug is welded to the DSC shell to form the inner confinement barrier. In later designs, the top shielding plug was revised to a 2-piece version that consists of an aluminum-coated carbon steel shielding plug encased in lead and a stainless steel inner cover plate placed on top of the shielding plug. The inner cover plate is welded to the shell to form an inner confinement barrier. In the “short cavity” version, the top shielding plug consists of a thick aluminum-coated carbon steel plate. The pressure and confinement boundaries for NUHOMS-32PT DSC are shown in Fig. V.1-7. All designs do not include the helium leak test port plug. A stainless steel top cover plate is placed on top of the top shielding plug to provide a redundant confinement boundary after the drying and helium backfill operations are complete. The top cover plate is welded to the DSC shell. Also, a rolled ring is attached to the top cover for handling purposes.

V.1.1.4 Vent and Siphon Ports

Each DSC has two stainless steel penetration ports at the top shielding plug for venting and draining the canister before it is sealed (Fig. V.1-7). Vent and siphon (or drain) penetration ports open to the DSC interior just below the top shielding plug. The siphon penetration incorporates a tube that continues to the bottom of the DSC interior cavity. Prior to installation of the top cover plate, the penetrations are sealed, by welding, as portions of the confinement barrier.

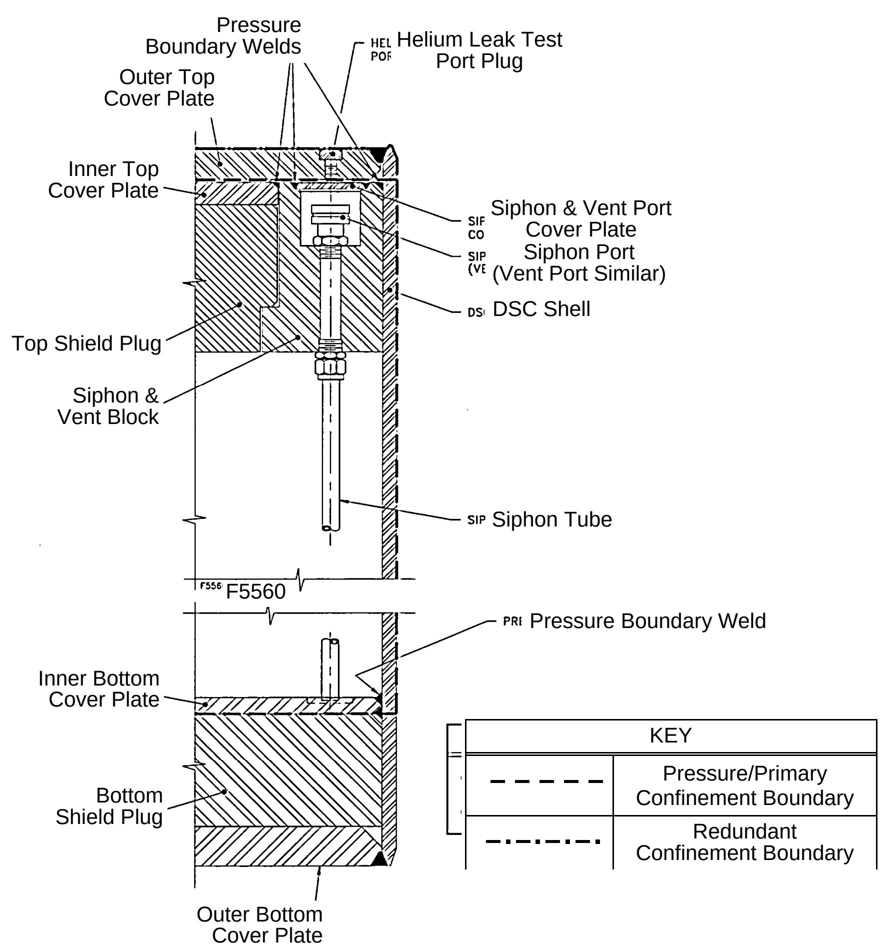


Figure V.1-7: Pressure and confinement boundaries for NUHOMS-32P1-T DSC.

V.1.2 Design Codes and Service Life

The DSCs are designed, fabricated, and inspected in accordance with ASME Boiler and Pressure Vessel Code rules for Class 1 components and core support structures (Section III, Subsections NB and NG). The HSM is designed in accordance with ACI-349 Code including load combination (dead load, live loads, and temperature). The DSC support structures and other miscellaneous structural steel for the HSMs are designed, fabricated, and constructed in accordance with AISC Code. The thermal cycling analysis of the HSM concrete and DSC support structure is based on one cycle per day or 18,250 cycles for a 50-year service life. The original fatigue analyses of the DSC shell (in accordance with ASME Code Section III NB-3222.4) and DSC support structure (in accordance with ASME Code Section III NF-3331.1) indicate that no consideration of fatigue is required for the 50-year service life.

The heat generation is limited to 0.3–1.2 kW per fuel assembly or 18–24 kW per canister. For example, on the basis of storage of 24 PWR assemblies per DSC with a nominal burnup of 40,000 MWD/MTU, an initial enrichment of 4.0 wt % U-235 and a nominal decay period of ten years yield a thermal load of 0.66 kW/assembly. The passive cooling system of the HSM is designed to ensure that peak cladding temperatures are less than 340°C (644°F) during long-term storage for average normal ambient temperatures of 21°C (70°F). The design temperature is 204.4°C (400°F) for the DSC internal structures and 247°F at the DSC outside surface under normal operating conditions (i.e., assuming 21°C ambient air). The highest temperature is at the top surface of the DSC.

The DSC is designed for a maximum dose rate of 200 mrem/h at the surface of the top and bottom end shielding plugs. The HSM is designed for an average dose rate of 20 mrem/h at the surface of the module, decreasing to a negligible level at the site boundary.

The estimated gamma dose in the BISCO NS-3 encased in the access door of a NUHOMS-24 system has been calculated to be approximately 1.8×10^5 rads for a service life of 60 years (service limit of the BISCO NS-3 is about 1.5×10^{10} rads). The estimated integrated neutron fluence in the HSM concrete has been calculated to be about 1.44×10^{14} n/cm^2 for a service life of 60 years (service limits for concrete for fast and thermal neutron exposure are 1.6×10^{17} and 1.5×10^{19} n/cm^2, respectively).

V.1.3 Current Inspection and Monitoring Program

The current inspection program for the NUHOMS® system in the original or extended license period (20- or 40-year extension) involves visual inspection of the entire exterior HSM concrete surfaces, shielding blocks, steel members, painting, caulking, concrete approach pad, storm water drainage system, and lightning protection system to ensure that no significant degradation of the HSMs occurs.

The interior components of the HSM are inspected in place or remotely using a camera and/or fiber optic technology by removal of the access doors. The components inspected may include interior concrete walls and floors, heat shielding plates and anchors, shielding blocks, DSC exterior surfaces (including welds), DCS support assembly, ventilation paths and shielding blocks, and various anchorages. Note that the DSC closure weld in the top cover plate can be inspected in place after removal of the HSM access door. However, inspection of the closure weld at the other end of the HSM (i.e., in the bottom plate of the DSC) may require withdrawing the DSC into the fuel transfer cask.

The current program also includes daily surveillances of the air inlets and outlets to ensure no obstruction occurs that could potentially overheat the components inside the HSM and the concrete structures. Facility procedures are in place for these inspections and surveillances.

In addition, the current program includes monitoring of area radiation levels for compliance with dose limits. Increased levels could indicate a breach of the confinement barriers of the DSC. Dose rates are measured at predetermined HSM locations.

The AMPs to manage aging effects for specific structures and components, the materials of construction, and environment of the NUHOMS® HSM and DSC are given in Tables V.1.A and V.1B. In these tables, the DCSS components listed in the Structure and/or Component column are classified as "A", "B", or "C" according to importance to safety, as described in Section I.2.

V.1.4 References

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Table V.1.A NUHOMS® Dry Spent-Fuel Storage: Horizontal Storage Module (HSM)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-1	Concrete: HSM walls, roof, and floor; inlet and outlet vents shield blocks (A)	RS, SS, HT	Reinforced Concrete	Air – inside the module, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general, >200°F local) and long-term exposure to gamma radiation	Site-specific AMP The implementation of 10 CFR 72 requirements and ASME Section XI, Subsection IWL, would not enable identification of the reduction of strength and modulus of elasticity due to elevated temperature and gamma radiation. Thus, for any portions of concrete in an HSM that exceed specified limits for temperature and gamma radiation, further evaluations are warranted. For normal operation or any other long-term period, Subsection CC-3400 of ASME Section III, Division 2, specifies that the concrete temperature limits shall not exceed 66°C (150°F) except for local areas, such as around penetrations, which are not allowed to exceed 93°C (200°F). Also, a gamma radiation dose of 10^{10} rads may cause significant reduction of strength. If significant equipment loads are supported by concrete exposed to temperatures exceeding 66°C (150°F) and/or gamma dose above 10^{10} rads, an evaluation is to be made of the ability to withstand the postulated design loads. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	Site-specific AMP Further evaluation, if temperature and gamma radiation limits are exceeded
V.1.A-2	Concrete (accessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-3	Concrete (inaccessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a site-specific AMP is needed to manage cracking and expansion due to reaction with aggregate of concrete in inaccessible areas. A site-specific AMP is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates per ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) for potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Further evaluation to determine whether a site-specific AMP is needed
V.1.A-4	Concrete (accessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Chapter IV.S1, “Structures Monitoring Program”	Generic program
V.1.A-5	Concrete (inaccessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Further evaluation is required to determine if a site-specific AMP is needed to manage loss of strength due to concrete interaction with aluminum in inaccessible areas.	Further evaluation to determine whether a site-specific AMP is needed
V.1.A-6	Concrete (accessible areas): Interior and above-grade exterior (A or B)	RS, SS, HT	Reinforced Concrete	Air – inside the module, uncontrolled or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, “Structures Monitoring Program”	Generic program

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-7	Concrete: (inaccessible areas) Below-grade exterior; basemat (pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Ground-water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program" Inaccessible Concrete Areas: For facilities with non-aggressive groundwater/soil, i.e., pH >5.5, chlorides <500 ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilities with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm), and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program
V.1.A-8	Concrete: Below-grade exterior; basemat (pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Ground-water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.1.A-9	Concrete (accessible areas): Exterior above- and below-grade; basemat (pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-10	Concrete (inaccessible areas): Basemat (pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for facilities that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a site-specific AMP is needed. A site-specific AMP is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Code, Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a site-specific AMP is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental U.S. is shown in ASTM C33-90, Fig. 1.	Further evaluation, for facilities located in moderate to severe weathering conditions
V.1.A-11	Concrete (inaccessible areas): Basemat (pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for facilities that are located in moderate to severe weathering conditions (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a site-specific AMP is needed. A site-specific AMP is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Code, Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a site-specific AMP is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental U.S. is shown in ASTM C33-90, Fig. 1.	Further evaluation, for facilities located in moderate to severe weathering conditions

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-12	Concrete: Interior and above-grade exterior (B)	RS, SS, HT	Reinforced Concrete	Air – inside the module, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.1.A-13	Concrete (inaccessible areas): Basemat (pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Ground-water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program" Inaccessible Concrete Areas: For facilities with non-aggressive groundwater/soil; i.e., pH >5.5, chlorides <500 ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilities with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm), and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program
V.1.A-14	Concrete (inaccessible areas): Basemat (pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Ground-water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-15	Concrete (accessible areas): Exterior above- and below-grade; basemat (pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.1.A-16	Concrete (inaccessible areas): Exterior above- and below-grade; basemat (concrete pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a site-specific AMP is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in inaccessible areas. A site-specific AMP is not required if (1) there is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Further evaluation, if leaching is observed in accessible areas that impact intended function
V.1.A-17	Concrete: Below-grade; basemat (concrete pad) and approach slab (ramp) (B)	SS	Reinforced Concrete	Soil and water – flowing under foundation	Reduction of strength, cracking due to differential settlement, and erosion of porous concrete sub-foundation	Chapter IV.S1, "Structures Monitoring Program." If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of that system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement
V.1.A-18	Concrete: All (A or B)	RS, SS, HT	Reinforced Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter IV.S1, "Structures Monitoring Program." If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of that system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-19	Moisture barriers (caulking, sealants, and expansion joint fillers) (C)	SS Not ITS	Elastomers, rubber and other similar materials	Air – outdoor	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.1.A-20	Storm drainage system (drain pipes and other components) (C)	SS Not ITS	PVC and other materials	Air – outdoor	Loss of drainage function due to blockage, wear, damage, erosion, tear, cracks, or other defects	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components"	Generic program
V.1.A-21	Transfer cask docking: Flange and access opening sleeve, axial seismic restraints (B)	SS	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components"	Generic program
V.1.A-22	Transfer cask docking: Flange and access opening sleeve, axial seismic restraints (B)	SS	Steel	Air – indoor, uncontrolled	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components"	Generic program
V.1.A-23	Heat Shielding Plates and Anchors (B)	HT	Stainless Steel, Steel	Air – inside the module	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components"	Generic program
V.1.A-24	DSC Support Structure: Structural beams, rails, plates, bolts and nuts, including welds, and various anchorages/ embedments (B)	SS	Steel, Stainless Steel	Air – indoor, uncontrolled	Loss of material due to corrosion and wear; cracking; and coating degradation	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components"	Generic program

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-25	DSC Support Structure: Structural beams, frames, and anchorages (B)	SS	Steel, Stainless Steel	Air – indoor, uncontrolled	Cumulative fatigue damage due to cyclic loading	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.1.A-26	Ventilation Air Openings: Air inlet and outlet screens and frames (A)	HT	Steel	Air – inside the module, uncontrolled or Air – outdoor	Loss of material and coating degradation due to corrosion and wear	Chapter IV.M2, "Ventilation System Surveillance Program"	Generic program
V.1.A-27	Ventilation Air Openings: Air inlet and outlet screens and frames (A)	HT	Steel	Air – inside the module, uncontrolled or Air – outdoor	Reduced heat convection capacity due to blockage	Chapter IV.M2, "Ventilation System Surveillance Program"	Generic program
V.1.A-28	Shielded access door: Access door support frame, access ring, and anchorages (B)	RS, SS	Steel	Air – inside the module, uncontrolled or Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components"	Generic program
V.1.A-29	Shielded access door: Shielding material (A)	RS	BISCO NS-3	Embedded between steel plates	Degradation of shielding material due to radiation exposure	Degradation of radiation-shielding materials is a TLAA to be evaluated for the period of extended operation. See Section III.5, "Time-Dependent Degradation of Radiation-Shielding Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-30	HSM/foundation basemat anchorage (inaccessible area): Dowel rods (B)	SS	Steel	Air – uncontrolled (dry to wet conditions)	Loss of material due to general, pitting, and crevice corrosion	Further evaluation is required to determine if a site-specific AMP is needed to manage loss of material due to corrosion. A site-specific AMP is not required if a TLAA is performed to manage aging effects of corrosion for the period of extended operation. See Section IV.3, "Corrosion Analysis of Metal Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1of NUREG-1927.	Further evaluation to determine whether a site-specific AMP is needed
V.1.A-31	Coatings (if applied) (C)	SS Not ITS	Coating	Air – inside the module, uncontrolled or Air – outdoor	Loss of coating integrity due to blistering, cracking, flaking, peeling, or physical damage	Chapter IV.S2, "Protective Coating Monitoring and Maintenance Program"	Generic program
V.1.A-32	Lightning protection system (C)	SS Not ITS	Various materials	Air – outdoor	Loss of lightning protection due to wear, tear, damage, surface cracks, or other defects	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.1.A-33	Electrical equipment subject to 10 CFR 50.49 EQ requirements (B)	Monitoring system	Various metallic and polymeric materials	Adverse localized environment caused by heat, radiation, oxygen, moisture, or voltage	Various degradation phenomena/ various mechanisms	EQ is a TLAA to be evaluated for the period of extended operation. See Section III.6, "Environmental Qualification of Electrical Equipment," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.1.A-34	Handrail and Bracing (B)	SS	Steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components"	Generic program

Table V.1.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.A-35	Cathodic protection systems (B)	Cathodic protection of reinforcing steel	Various materials	Embedded in concrete	Reduction of cathodic protection effect on bond strength due to degradation of cathodic protection current	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Notes:

1. The HSM is anchored to the foundation slab (concrete pad) to mitigate overturning and sliding effects using dowel rods of a size and spacing consistent with the HSM wall vertical reinforcement. The rods could be corroded by water, moisture, or aggressive chemicals through the crevices between the HSM walls and the concrete pad.
2. External precast shielding blocks are placed over HSM air outlets to reduce direct and streaming radiation dose. Internal shielding blocks are placed around air inlets to reduce direct and streaming radiation dose. The shielding blocks are anchored to the embedded base plate on roof and floor slab.

Table V.1.B NUHOMS® Dry Spent-Fuel Storage: Dry Shielded Canister (DSC)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.1.B-1	DSC: Shell (including welds) (A)	CB, HT, SS, FR	Stainless Steel	Air – inside the HSM, uncontrolled (external), Helium (internal)	Cumulative fatigue damage due to cyclic loading	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.1.B-2	DSC confinement boundary: Shell, outer top cover plate, outer bottom cover plate, and welds (A)	CB, HT, SS, FR	Stainless Steel	Air – inside the HSM, uncontrolled (external), Helium (internal)	Cracking and leakage due to stress corrosion cracking when exposed to moisture and aggressive chemicals in the environment	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components" Chapter IV.M3, "Welded Canister Seal and Leakage Monitoring Program"	Generic program
V.1.B-3	DSC Internals: Basket assembly (Spacer disks, support rods, guide sleeves, transition rail structure, basket rails and hold-down rails); Shielding plugs and inner cover plates; Vent/siphon block (A)	CC, CB, HT, RS, SS, FR	Stainless Steel, aluminum-coated carbon steel, borated aluminum or boron carbide/ composite	Helium	Degradation of heat transfer, radiation shield, criticality control, confinement boundary, or structural support functions of the DSC internals due to extended exposure to high temperature and radiation.	Chapter IV.M5, "Canister Structural and Functional Integrity Monitoring Program"	Generic program
V.1.B-4	DSC Internals: Fuel basket; Neutron absorber panels (A)	CC	Borated aluminum or boron carbide/ aluminum alloy plate or BORAL composite	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of neutron-absorbing materials is a TLAA to be evaluated for the period of extended operation. See Section III.4, "Time-Dependent Degradation of Neutron-Absorbing Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA

V.2 HI-STORM 100 System

V.2.1 System Description

This section addresses the elements of the HI-STORM 100 system for dry storage of used nuclear fuel. The basic HI-STORM system (Fig. V.2-1) consists of interchangeable MPCs providing a confinement boundary for BWR or PWR used nuclear fuel, a storage overpack providing a structural and radiological boundary for long-term storage of the MPC placed inside it, and a transfer cask for transfer of a loaded MPC from a used-fuel storage pool to the storage overpack. All MPC designs have a nominal external diameter of 1.74 m (68.375 in.) and the maximum overall length is 4.84 m (190.5 in.). The maximum weight of fully loaded MPCs, however, varies because of the differing storage contents; the maximum weight is approximately 44.5 tons. The MPCs are designed for maximum and minimum temperatures of 385°C (725°F) and -40°C (-40°F); internal pressures of 689.5, 758.4, and 1,379.0 kPa (100, 110, and 200 psi) under normal, off-normal, and accident conditions, respectively; and maximum permissible peak fuel cladding temperatures of 400°C (752°F) for short- and long-term normal operations and 570°C (1058°F) for off-normal and accident conditions and during MPC drying.

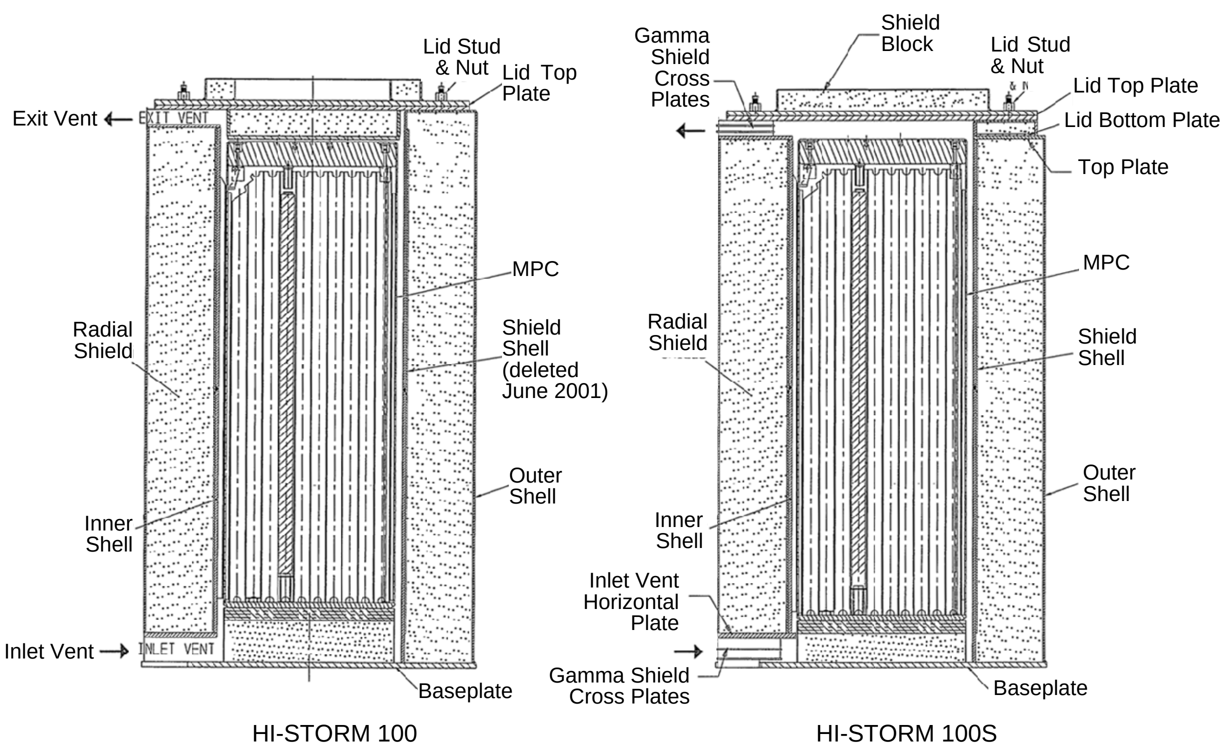

Figure V.2-1: Cross-sectional views of an MPC inserted into HI-STORM 100 and 100S storage overpacks.

A base HI-STORM overpack design is capable of storing each type of MPC. The overpack inner cavity can accommodate canisters with an inner-shell diameter of 1.87 m (73.5 in.) and a cavity height of 4.86 m (191.5 in.). The overpack inner shell is provided with channels distributed around the inner cavity to present an available inside diameter of 1.77 m (69.5 in.). The channels provide guidance for MPC insertion and removal, and a flexible medium to absorb some of the impact during a tip-over. They also allow flow of cooling air through the overpack. The outer diameter of the overpack is 3.37 m (132.5 in.), and the overall height is 6.08 m (239.5 in.). The design life of the HI-STORM 100 System is 40 years.

There are three base HI-STORM overpack designs: HI-STORM 100, HI-STORM 100S, and HI-STORM 100S Version B. The significant differences among the three are overpack height, MPC pedestal height, location of the air outlet ducts, and the vertical alignment of the inlet and outlet air ducts. The HI-STORM 100S Version B overpack design does not include a concrete-filled pedestal to support the MPC. Instead, the MPC rests upon a steel plate that maintains the MPC sufficiently above the inlet air ducts to prevent direct radiation shine through the ducts. Cross-sectional views of the storage system with an MPC inserted into HI-STORM 100 and HI-STORM 100S overpacks, respectively, are presented in Fig. V.2-1. Similar information for the HI-STORM 100S Version B overpack is provided in Fig. V.2-2. The HI-STORM 100S system is either 5.89 or 6.17 m (232 or 243 in.) high, and the HI-STORM 100S Version B system is 5.54 or 5.82 m (218 or 229 in.) high.

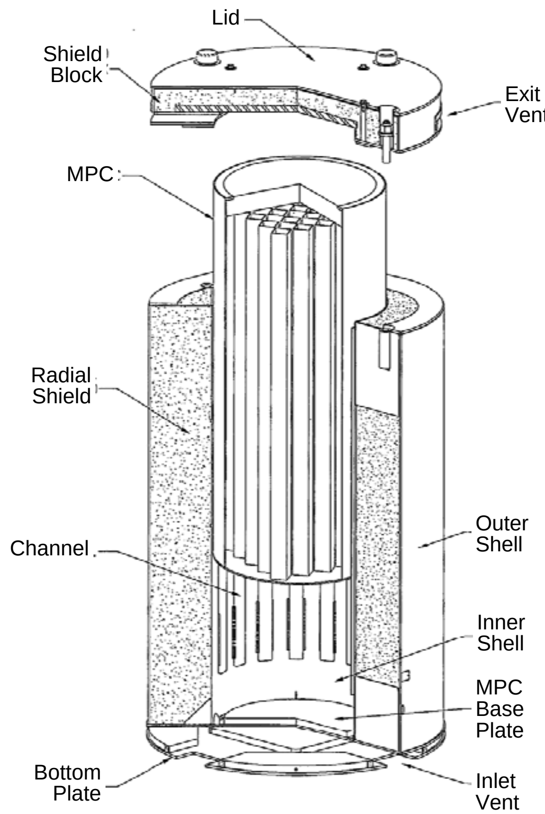

Figure V.2-2: HI-STORM 100 System using a HI-STORM 100S Version B overpack.

Another version of the HI-STORM 100A and 100SA overpacks, which are equipped with lugs to anchor the overpack to the ISFSI pad, is generally deployed at those sites where the postulated seismic events exceed the maximum limit permitted for free-standing installation. The anchored version of the HI-STORM system differs only in the diameter of the overpack baseplate and the presence of holes and associated anchorage hardware. The HI-STORM 100S version B overpack design is not deployed in the anchored configuration at this time.

V.2.1.1 Multipurpose Canisters

The MPCs are welded cylindrical structures, shown in cross-sectional views in Figs. V.2-3. The outer diameter of each MPC is fixed. Each used-fuel MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, a canister shell, a lid, and a closure ring. A cross-sectional elevation view of a fuel basket for the MPC-68 series is shown in Fig. V.2-4. The number of used nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics. There are eight MPC models, distinguished by the type and number of fuel assemblies authorized for loading: MPC-24 (including MPC-24E and MPC-24EF), MPC 32 (including MPC-32F), and MPC-68 (including MPC-68F and MPC-68FF). The fuel storage cells in the MPC-24 series are physically separated from one another by a “flux trap,” for criticality control. Flux traps are not used in the MPC-32 and MPC-68 series. The PWR MPC-32/32F and MPC-68 designs do not use flux traps; their design includes credit for soluble boron in the MPC water during wet fuel loading and unloading operations for criticality control.

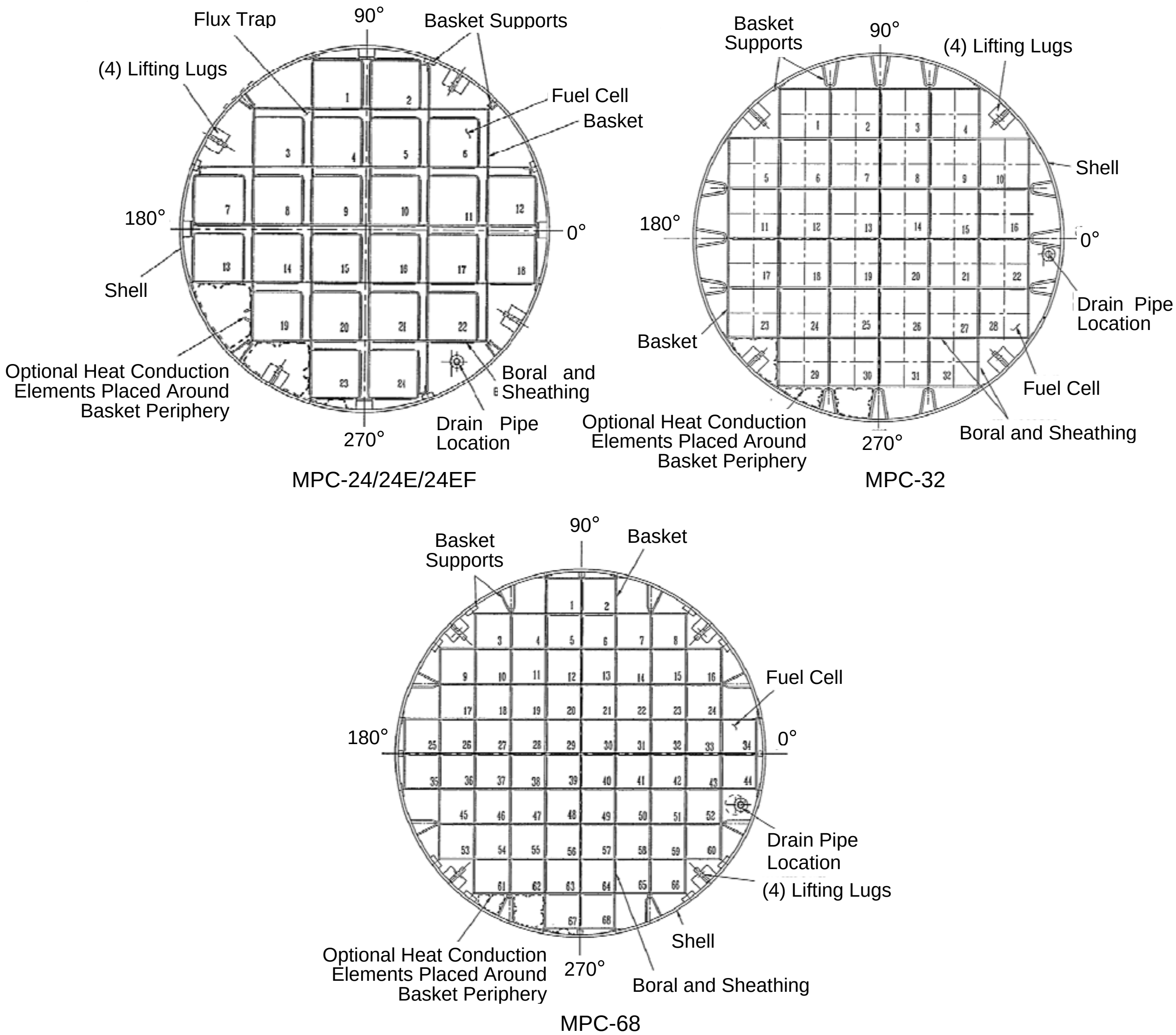

Figure V.2-3: Cross-sectional views of different MPC designs.

The MPC fuel baskets that do not use flux traps (namely, MPC-68, MPC-68F, MPC-68FF, MPC-32, and MPC-32F) are constructed from an array of plates welded to each other at their intersections. In the flux-trap type fuel baskets (MPC-24, MPC-24E, and MPC-24EF), angle sections are interposed onto the orthogonally configured plate assemblage to create the required flux-trap channels. The MPC fuel basket is positioned and supported within the MPC shell by a set of basket supports welded to the inside of the MPC shell. In the early-vintage MPCs fabricated, certified, and loaded under the original HI-STORM 100 design, optional heat conduction elements (fabricated from thin aluminum alloy 1100) may have been installed between the periphery of the basket, the MPC shell, and the basket supports. The heat-conduction elements are installed along the full length of the MPC basket except at the drainpipe location to create a nonstructural thermal connection that facilitates heat transfer from basket to shell. The aluminum heat conduction elements are not installed in later version of the HI-STORM 100.

For fuel assemblies that are shorter than the design basis length, upper and lower fuel spacers, as appropriate, maintain the axial position of the fuel assembly within the MPC basket. The upper fuel spacers are threaded into the underside of the MPC lid as shown in Fig. V.2-4. The lower fuel spacers are placed in the bottom of each fuel basket cell. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 50.8–63.5 mm (2.0–2.5 in.) is provided to account for the irradiation and thermal growth of the fuel assemblies. The actual length of fuel spacers is determined on a site-specific or fuel-assembly-specific basis.

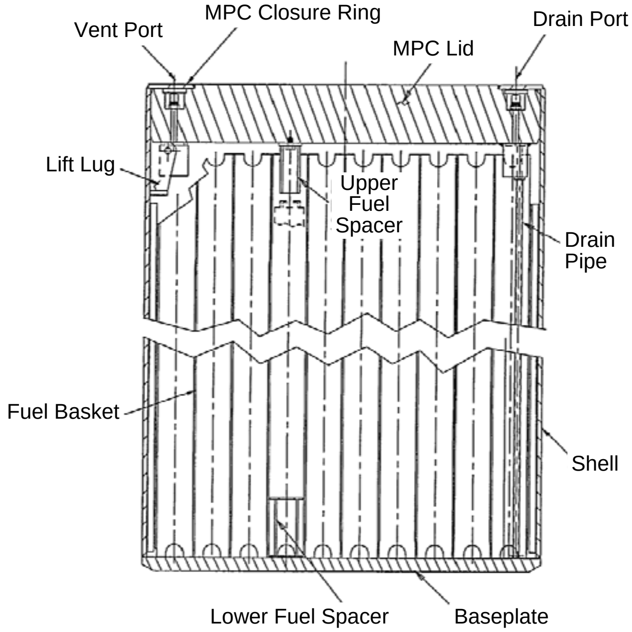

Figure V.2-4: Cross-section elevation view of MPC.

All structural components in MPCs are made of a material designated by the manufacturer as “Alloy X.” Candidate Alloy X materials include Types 304, 304L, 316, and 316LN austenitic stainless steels. Any steel component in an MPC may be fabricated from any Alloy X material; however, the various sections of the 12.7-mm (0.5-in.)-thick cylindrical MPC shell must all be fabricated from the same type of Alloy X stainless steel. All MPC components that are likely to come in contact with used-fuel pool water or the ambient environment (with the exception of neutron absorber, aluminum seals on vent and drain port caps, and optional aluminum heat conduction elements) are constructed from stainless steel. Thus, there are no concerns regarding potential interactions between coated carbon steel materials and the various MPC operating environments.

Lifting lugs attached to the inside surface of the MPC canister shell (Fig. V.2-4) serve to permit placement of the empty MPC into the HI-TRAC transfer cask and also serve to axially locate the MPC lid prior to welding. They are not used to handle a loaded MPC because the MPC lid is installed prior to any handling of a loaded canister. The MPC lid is a circular plate (fabricated from one piece or two pieces—split top and bottom) edge-welded to the MPC outer shell. In the two-piece lid design, only the top piece comprises a part of the enclosure vessel’s pressure boundary; the bottom piece is attached to the top piece with a non-structural, non-pressure-retaining weld and acts as a radiation shield. The lid is equipped with vent and drain ports that are utilized to remove moisture and air from the MPC and backfill the MPC with helium. The vent and drain ports are covered and seal welded before the closure ring is installed (Fig. V.2-5). The closure ring is a circular ring edge-welded to the MPC shell and lid; details are shown in Fig. V.2-6. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with used nuclear fuel to be lifted by threaded holes in the MPC lid.

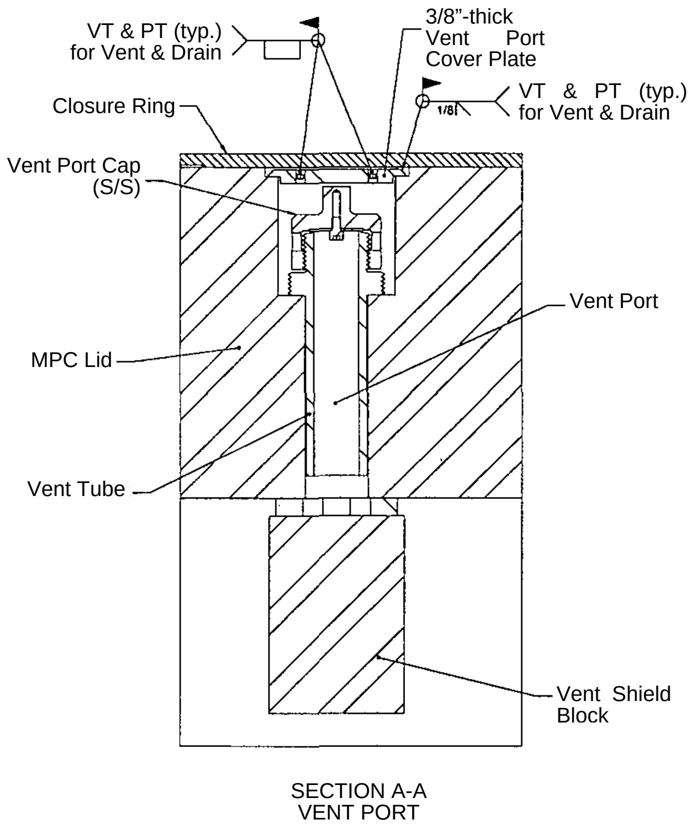

Figure V.2-5: MPC vent port details

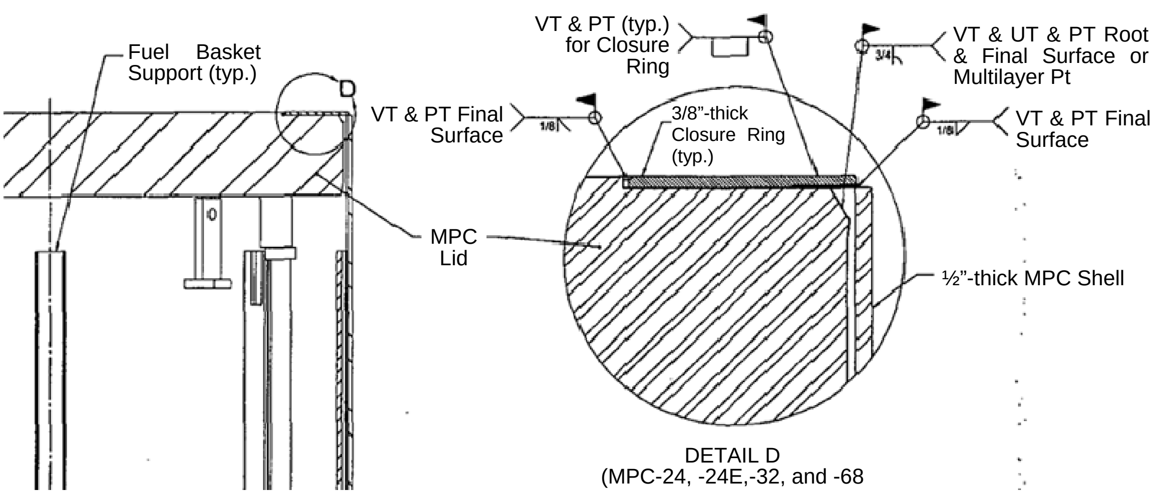

Figure V.2-6: MPC closure details showing the MPC shell, MPC lid, and closure ring.

The MPC does not require any valves, gaskets or mechanical seals for confinement. Figure V.2-7 shows the MPC confinement boundary. All components of the confinement boundary are safety significant, and are fabricated entirely of stainless steel. The MPC confinement boundary components are designed and fabricated in accordance with the ASME Code Section III, Subsection NB. The primary confinement boundary is defined by the outline formed by the sealed, cylindrical enclosure of the MPC shell (including any associated axial or circumferential welds) welded to the baseplate at the bottom, the MPC lid welded around the top circumference to the shell wall, and the port cover plates welded to the lid. As required by 10 CFR 72.236(e), the MPC incorporates a redundant closure system consisting of the closure ring welded to the lid and the MPC shell, as shown in Fig. V.2-4. All welds associated with the MPC confinement boundary are shown in Fig. V.2-8. The welds between MPC lid and vent or drain port cover plate are helium leak tested. The weld between lid and MPC shell is not required to be helium leak tested because (a) it is multipass (more than a 2-pass) weld, (b) root pass, cover pass and at least one in-between pass are inspected by either ultrasonic testing (UT) or liquid penetrant testing (PT), and (c) the minimum detectable flaw size is demonstrated to be less than the critical flaw size as calculated in accordance with ASME Section XI methodology. A shield lid is bolted to the top of the MPC lid, and provides radiation shielding (Fig. V-2-9).

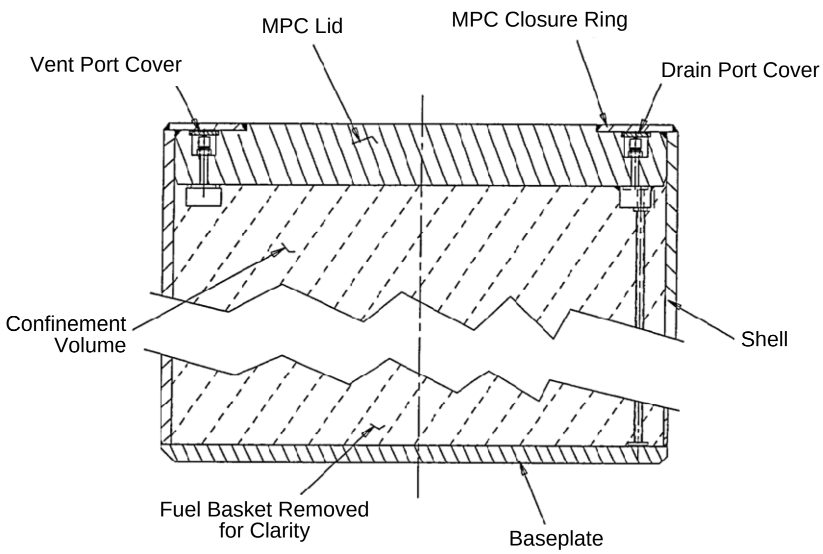

Figure V.2-7: MPC confinement boundary.

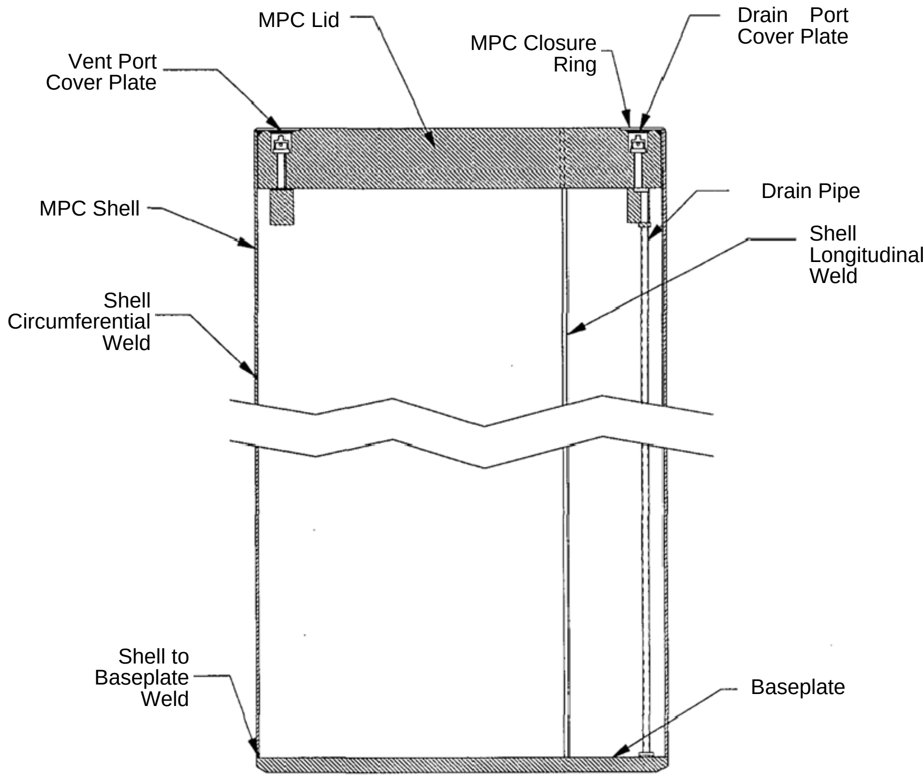

Figure V.2-8: Weld associated with the MPC confinement boundary.

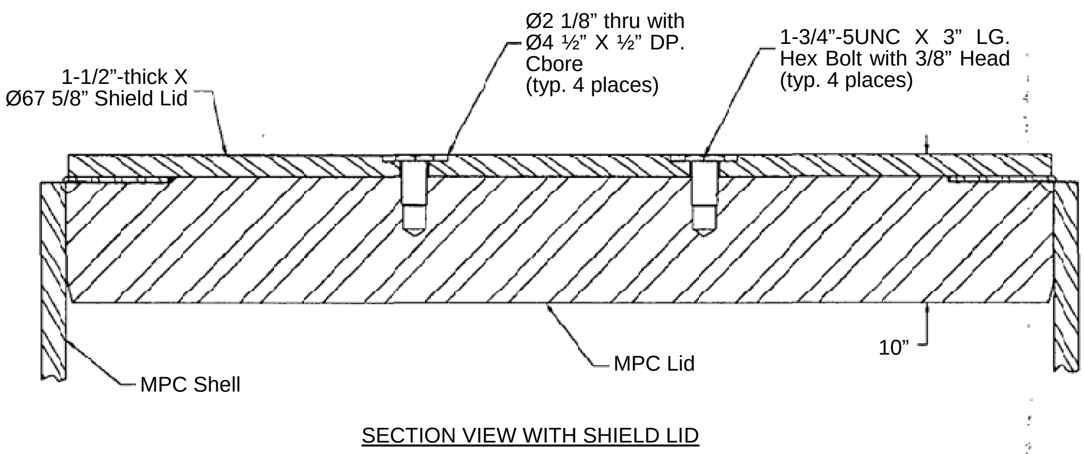

Figure V.2-9: Cross sectional view of the MPC lid, closure ring, and shield lid.

The helium backfill gas plays an important role in the MPC thermal performance. It fills all the spaces between solid components and provides an improved conduction medium relative to air for dissipating decay heat in the MPC. Furthermore, the pressurized helium environment within the MPC sustains a closed-loop thermo-siphon action, removing used-nuclear-fuel decay heat by upward flow of helium through the storage cells. This internal convection heat dissipation process is illustrated in Fig. V.2-10.

V.2.1.2 Overpacks

The HI-STORM overpacks are rugged, heavy-walled cylindrical vessels. Figure V.2-11 shows cross-sectional views of the HI-STORM 100 and 100S overpacks. The HI-STORM 100A and 100SA overpack designs are the anchored variant of the HI-STORM 100 and -100S designs. The HI-STORM 100A and 100SA systems differ only in the diameter of the overpack baseplate and the presence of bolting holes and associated anchorage hardware. The main structural function of the storage overpack is provided by carbon steel, and the main shielding function is provided by plain concrete. The plain concrete, enclosed by cylindrical inner and outer steel shells, a thick baseplate, and a top plate, is specified to provide the necessary shielding properties (dry density) and compressive strength. The overpack lid has appropriate concrete shielding to provide neutron and gamma attenuation in the vertical direction.

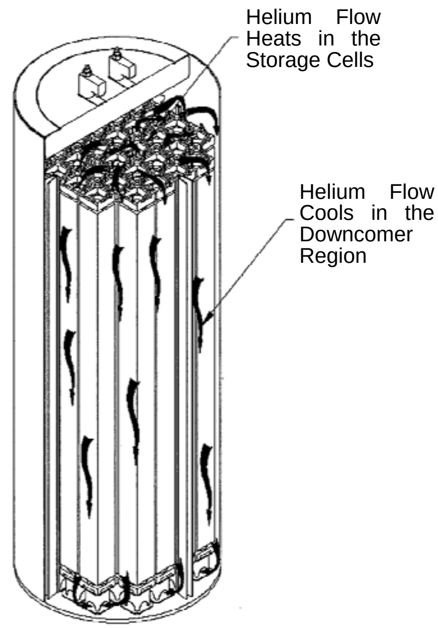

Figure V.2-10: MPC internal helium circulation.

The vertical annulus between the MPC and the inner shell of the overpack facilitates an upward flow of air by buoyancy forces, drawing ambient air from the inlet vents and releasing it from the outlet vents at the top of the HI-STORM storage system. The annulus ventilation flow cools the hot MPC surfaces and safely transfers decay heat to the outside environment. This overpack cooling process is illustrated in Fig. V.2-12.

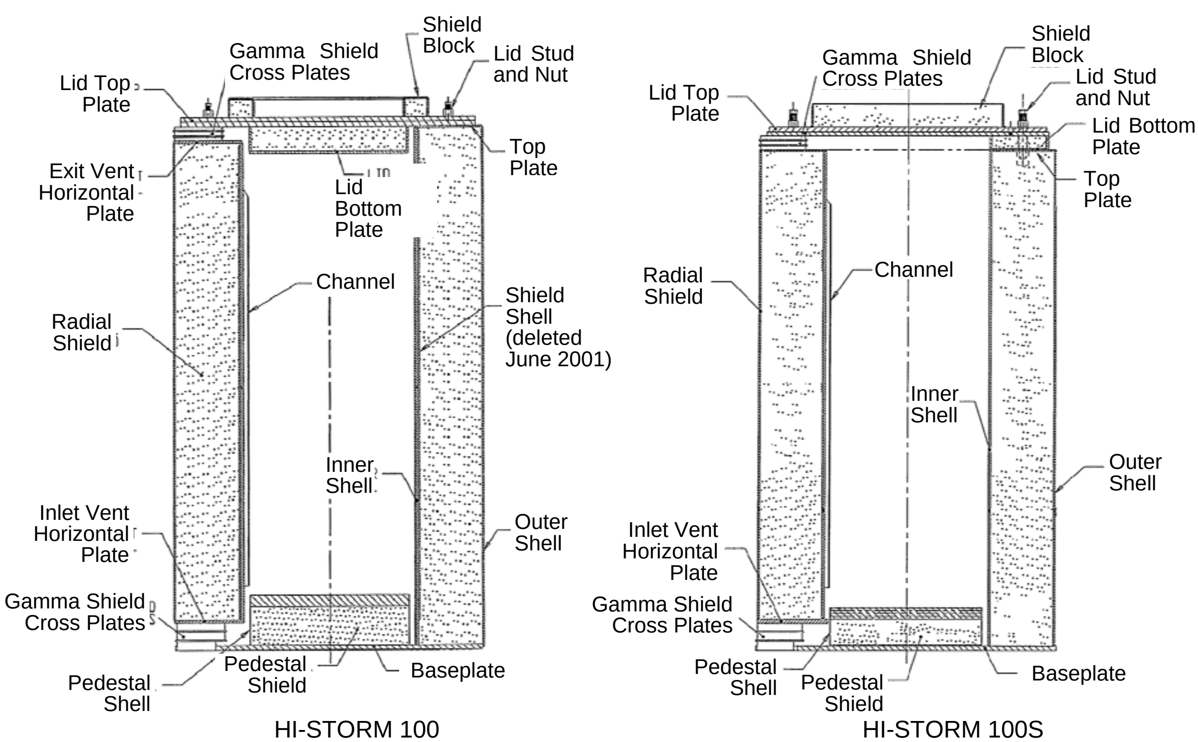

Figure V.2-11: Cross-sectional views of the HI-STORM 100 and 100S overpacks.

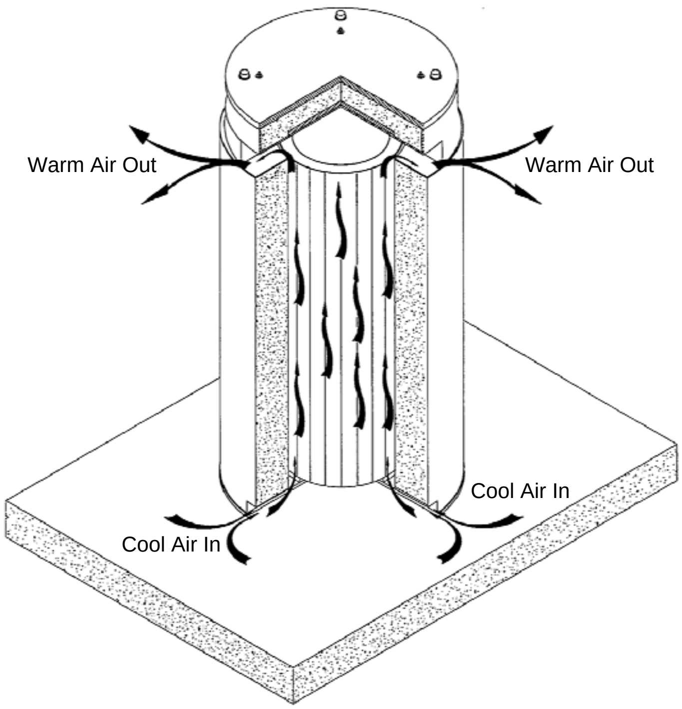

Figure V.2-12: Ventilation cooling of a HI-STORM storage system.

The principal function of the concrete is to provide shielding against gamma and neutron radiation. However, it also imparts a large thermal inertia to the HI-STORM overpack, allowing it to moderate the rise in temperature of the system under hypothetical conditions when all ventilation passages are assumed to be blocked. The high thermal inertia characteristics of the HI-STORM concrete also control the temperature of the MPC in the event of a postulated fire accident at the ISFSI. Although the annular concrete mass in the overpack shell is not a structural member, it does act as an elastic/plastic filler of the intershell space.

The HI-STORM overpack has air ducts to allow for passive natural convection cooling of the contained MPC. A minimum of four air inlets and four air outlets are located at the lower and upper extremities of the storage system, respectively. The locations of the air outlets in the HI-STORM 100 and the HI-STORM 100S (including Version B) designs differ in that the outlet ducts for the HI-STORM 100 overpack are located in the overpack body and are aligned vertically with the inlet ducts at the bottom of the overpack body. The air outlet ducts in the HI-STORM 100S and 100S Version B are integral to the lid assembly and are not in vertical alignment with the inlet ducts. A screen to reduce the potential for blockage covers the air inlets and outlets.

Four threaded anchor blocks, located at 90° arcs around the circumference of the top of the overpack lid, are provided for lifting. The anchor blocks are integrally welded to the radial plates, which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (HI-STORM 100) or the inlet air duct horizontal plates (HI-STORM 100S). The HI-STORM 100S Version B overpack design incorporates partial-length radial plates at the top of the overpack to secure the anchor blocks and uses both gussets and partial-length radial plates at the bottom of the overpack for structural stability. The overpack may also be lifted from the bottom using specially designed lifting transport devices, including hydraulic jacks, air pads, Hillman rollers, or other designs based on site-specific needs and capabilities.

As discussed earlier, the HI-STORM overpack is a steel weldment, which makes it a relatively simple matter to extend the overpack baseplate, form lugs (or "sector lugs"), and then anchor the cask to the reinforced concrete structure of the ISFSI. The sector lugs are bolted to the ISFSI pad using anchor studs that are made of a creep-resistant, high-ductility, environmentally compatible material. The typical HI-STORM/ISFSI pad fastening detail is shown in Fig. V.2-13. The lateral load-bearing capacity of the HI-STORM/pad interface is many times greater than the horizontal sliding force exerted on the cask under the postulated design basis earthquake seismic event. Thus, the potential for lateral sliding of the HI-STORM 100A system during a seismic event is precluded, as is the potential for any bending action on the anchor studs. The sector lugs in the HI-STORM 100A are typically made of the same steel material as the baseplate and the shell (SA516-Gr. 70), which helps ensure high-quality fillet welds used to join the lugs to the body of the overpack.

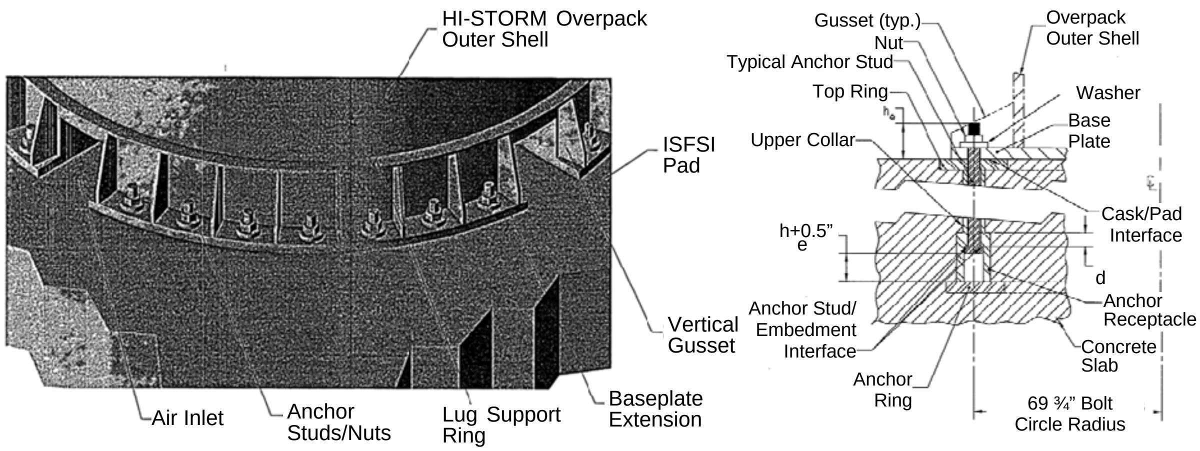

Figure V.2-13. Anchoring details for the HI-STORM 100A and 100SA overpacks.

V.2.1.3 Shielding Materials

The HI-STORM 100 System is provided with shielding to ensure that the radiation and exposure requirements in 1OCFR72.104 and 1OCFR72.106 are met. This shielding is an important factor in minimizing the personnel doses from the gamma and neutron sources in the used nuclear fuel in the MPC during loading, handling, transfer, and storage. The fuel basket structure of edge-welded composite boxes and neutron absorber panels attached to the fuel storage cell vertical surfaces provides the initial attenuation of gamma and neutron radiation emitted by the radioactive used fuel. The MPC shell, baseplate, lid and closure ring provide additional thicknesses of steel to further reduce the gamma flux at the outer canister surfaces.

In the HI-STORM storage overpack, the primary shielding in the radial direction is provided by concrete and steel. In addition, the storage overpack has a thick circular concrete slab attached to the lid, and the HI-STORM 100 and 100S have a thick circular concrete pedestal upon which the MPC rests. This concrete pedestal is not necessary in the HI-STORM 100S Version B overpack design. These slabs provide gamma and neutron attenuation in the axial direction. The thick overpack lid and concrete shielding integral to the lid provide additional gamma attenuation in the upward direction, reducing both direct radiation and skyshine. Several steel plate and shell elements provide additional gamma shielding as needed in specific areas, as well as incremental improvements in the overall shielding effectiveness. To reduce the radiation streaming through the overpack air inlets and outlets, gamma shield cross plates are installed in the ducts to scatter the radiation (Fig. V.2-14). The configuration of the gamma shield cross plates is such that the increase in the resistance to flow in the air inlets and outlets is minimized. This scattering acts to significantly reduce the local dose rates adjacent to the overpack air inlets and outlets. The inlet air ducts for the HI-STORM 100S Version B are shorter in height but larger in width.

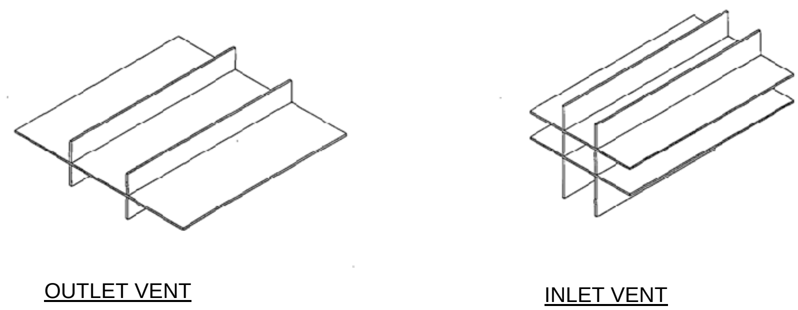

Figure V.2-14: Gamma shield cross plates for HI-STORM 100 and 100S overpacks.

Neutron absorbers: BORAL and METAMIC® neutron absorber panels are used and are completely enclosed in Alloy X stainless steel sheathing that is stitch welded to the MPC basket cell walls along their entire periphery. The edges of the sheathing are bent toward the cell wall to make the edge weld. Thus, the neutron absorber is contained in a tight, welded pocket enclosure. The shear strength of the pocket weld joint, which is an order of magnitude greater than the weight of a fuel assembly, guarantees that the neutron absorber and its enveloping sheathing pocket will maintain their as-installed position under all loading, storage, and transient conditions. In addition, the pocket joint detail ensures that fuel assembly insertion or withdrawal into or out of the MPC basket will not lead to a disconnection of the sheathing from the cell wall.

Neutron shielding: Neutron attenuation in the HI-STORM overpack is provided by the thick walls of concrete contained in the steel vessel, lid, and pedestal (only for the HI-STORM 100 and 100S overpack designs). The concrete composition has been specified to ensure its continued integrity at the long-term temperatures required for used nuclear fuel storage.

Gamma shielding material: For gamma shielding, the HI-STORM 100 storage overpack primarily relies on massive concrete sections contained in a robust steel vessel. A carbon steel plate, the shield shell, is located adjacent to the overpack inner shell to provide additional gamma shielding. Carbon steel supplements the concrete gamma shielding in most portions of the storage overpack, most notably the pedestal (HI-STORM 100 and 100S overpack designs only) and the lid.

It appears that there is a very limited concern about fatigue damage to HI-STORM 100 system structural components due to thermal loading, but it was not possible to find supporting analysis or numbers in the FSAR. The system design is such that there is no physical interference (either radial or axial) between fuel basket, MPC shell and overpack due to free thermal expansion. Thermal stresses in the MPC shell due to differential thermal expansion are small. The FSAR states that thermal gradient in the basket has been minimized, but no data for the corresponding stresses are given.

The storage facility is divided into two elements: the MPC, including basket assembly, and the storage overpack (HI-STORM).

The MPC design has the following six objectives:

1. Provide confinement
2. Dissipate heat
3. Withstand large impact loads
4. Provide unrestrained free-end expansion
5. Maintain geometric spacing to avoid criticality
6. Avoid significant impairment of retrievability of stored used fuel

The storage overpack design has the following objectives:

1. Provide a missile barrier and radiological shielding
2. Provide cooling for the MPC by providing flow paths for natural convection
3. Provide kinematic stability to the MPC, which is a free-standing component
4. Act as an energy absorber for the MPC in the event of a tip-over accident

V.2.2 Design Codes and Service Life

The design life of the HI-STORM 100 System is 40 years. The design life considers the effects of environmental exposure, material degradation, corrosion, structural fatigue effects, helium atmosphere, cladding temperatures, and neutron-absorber boron depletion throughout the design life. Section III of the ASME Boiler and Pressure Vessel Code is the governing code for the structural design of the MPC and the steel structure of the overpack. The MPC confinement boundary is designed in accordance with Section III, Subsections NB Class 1. The MPC fuel basket and basket support are designed in accordance with Section NG Class 1. The overpack steel structure and anchor studs are designed in accordance with ASME code Section III, Subsection NF, Class 3.

ACI 349 is the governing code for the plain concrete in the overpack. ACI 318.1-85 is the code utilized to determine the allowable compressive strength of the plain concrete.

If the Zero Period Accelerations (ZPAs) at the surface of the concrete pad exceed the threshold limit for free-standing HI-STORM, the cask must be installed in an anchored configuration (HI-STORM 100A). The embedment design for the HI-STORM 100A (and 100SA) shall comply with Appendix B to ACI-349-97. A later Code edition may be used provided a written reconciliation is performed.

V.2.3 Current Inspection and Monitoring Program

The HI-STORM FSAR states that visual inspection of the vent screens is required to ensure that the air inlets and outlets are free from obstruction (or alternatively, temperature monitoring may be utilized). The FSAR further states that if an air temperature monitoring system is used in lieu of visual inspection of the air inlet and outlet vents, the thermocouples and associated temperature monitoring instrumentation shall be maintained and calibrated in accordance with the user's QA program commensurate with the equipment's safety classification and designated QA category.

Other maintenance includes reapplication of corrosion-inhibiting materials on accessible external surfaces and periodic visual inspection of overpack external surfaces.

Radiation monitoring of the ISFSI in accordance with 10 CFR72.104(c) provides ongoing evidence and confirmation of shielding integrity and performance. If increased radiation doses are indicated by the facility monitoring program, additional surveys of overpacks shall be performed to determine the cause of the increased dose rates. The neutron absorber panels installed in the MPC baskets are not expected to degrade under normal long-term storage conditions. No periodic verification testing of neutron poison material is required.

The AMPs to manage aging effects for specific structures and components, the materials of construction, and environment of the HI-STORM systems are given in Tables V.2.A and V.2B. In these tables, the DCSS components listed in the Structure and/or Component column are classified as “A”, “B”, or “C” according to importance to safety, as described in Section I.2.

V.2.4 References

10 CFR 72.104, Criteria for Radioactive Materials in Effluents and Direct Radiation from an ISFSI or MRS, Office of the Federal Register, National Archives and Records Administration, 2011.

ACI 349-85, Code Requirements for Nuclear Safety Related Concrete Structures, American Concrete Institute, Detroit, MI, 1985.

ACI 318.1-89 and ACI 318.1R-89, Building Code Requirements for Structural Plain Concrete (Revised) and Commentary (Revised), American Concrete Institute, Detroit, MI, 1992.

ASME Boiler & Pressure Vessel Code, Section III, Subsection NB, American Society of Mechanical Engineers, New York, 1995 (with Addenda through 1997).

ASME Boiler & Pressure Vessel Code, Section III, Subsection NF, American Society of Mechanical Engineers, New York, 1995 (with Addenda through 1997).

ASME Boiler & Pressure Vessel Code, Section III, Subsection NG, American Society of Mechanical Engineers, New York, 1995 (with Addenda through 1997).

EPRI 1021048, Industry Spent Fuel Storage Handbook, Electric Power Research Institute, Palo Alto, CA, July 2010.

FCRD-USED-2011-00136, Gap Analysis to Support Extended Storage of Used Nuclear Fuel, Rev. 0, U.S. Department of Energy, Washington, DC, January 31, 2012.

HI-2002444, Final Safety Analysis Report for the HI-STORM 100 Cask System, Revision 8, USNRC Docket No. 72-1014, Holtec International, Marlton, NJ, January 18, 2010.

NEI 98-01, Industry Spent Fuel Storage Handbook, Nuclear Energy Institute, Washington, DC, May 1998.

NUREG-1571, Information Handbook on Independent Spent Fuel Storage Installations, U.S. Nuclear Regulatory Commission, Washington, DC, 1995.

Response to NRC Request for Additional Information on License Amendment Request No. 8 to Holtec International HI-STORM 100 Certificate of Compliance No. 1014, Holtec International, Marlton, NJ, November 4, 2010.

Table V.2.A HI-STORM 100 System: Storage Overpack and Pad

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-1	Storage overpack: Outer and inner shell (including guidance channels for MPC insertion and retrieval), baseplate, covers for plain concrete shielding blocks, sector lugs, lid studs and nuts, top ring, upper collar, anchor ring, anchor studs and receptacle (A or B)	SS, HT, RS, FR	Carbon Steel	Normal air or marine environment – outdoor	Loss of material due to general corrosion, pitting, crevice corrosion	Chapter IV.S1,"Structures Monitoring Program" Chapter IV.S2, "Protective Coating Inspection and Maintenance Program"	Generic program
V.2.A-2	Storage overpack: Overpack concrete radiation shield (between inner and outer shells); pedestal shield; and overpack lid shield (A)	RS, SS	Plain Concrete	Radiation and elevated temperature	Reduction of strength and modulus of concrete and degradation of shielding performance due to long-term exposure to high temperature and gamma radiation	Site-specific AMP The compressive strength and shielding performance of plain concrete is maintained by ensuring that the minimum concrete density is achieved during construction and the allowable concrete temperature and radiation limits are not exceeded. Further evaluations are warranted if these specified limits for temperature and gamma radiation are exceeded during service. Subsection CC-3400 of ASME Section III, Division 2, specifies that for normal operation, concrete temperature shall not exceed 66°C (150°F) for a long period and for accident conditions, concrete temperature shall not exceed 93°C (200°F) for short periods. Also, a gamma radiation dose of 10^{10} rads may cause significant reduction of strength. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are incorporated in the design calculations.	Site-specific AMP Further evaluation, if temperature and gamma radiation limits are exceeded

Table V.2.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-3	Storage overpack: Overpack concrete radiation shield (between inner and outer shells); pedestal shield; and overpack lid shield (A)	RS, SS	Plain Concrete	Radiation and elevated temperature	Reduction of strength and modulus of concrete and degradation of shielding performance due to reaction with aggregate of concrete in inaccessible areas	Further evaluation is required to determine if a site-specific AMP is needed to manage cracking and expansion due to reaction with aggregate of concrete in inaccessible areas. A site-specific AMP is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) for potentially reactive aggregates, aggregate concrete reaction is not significant if it is demonstrated that the in-place concrete can perform its intended function.	Site-specific AMP Further evaluation to determine whether a site-specific AMP is needed
V.2.A-4	Ventilation air openings: Air ducts, screens, gamma shield cross plates (A)	HT	Steel	Air – inside the module, uncontrolled or Air – outdoor or marine environment	Loss of material and coating degradation due to corrosion and wear	Chapter IV.M2, "Ventilation System Surveillance Program"	Generic program
V.2.A-5	Ventilation air openings: Air ducts, screens, gamma shield cross plates (A)	HT	Steel	Air – inside the module, uncontrolled or Air – outdoor	Reduced heat convection capacity due to blockage	Chapter IV.M2, "Ventilation System Surveillance Program."	Generic program
V.2.A-6	Anchor Studs (for anchored cask) (A)	SS	SA-193, SA-354, SA-479, SA-540, SA-564, SA-574, SA-638	Normal air or marine environment – outdoor	Loss of preload due to self loosening; loss of material due to corrosion; cracking due to stress corrosion cracking	Chapter IV.S1,"Structures Monitoring Program"	Generic program

Table V.2.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-7	Anchor Studs (for anchored cask) (A)	SS	SA-193, SA-354, SA-479, SA-540, SA-564, SA-574, SA-638	Air – outdoor	Cumulative fatigue damage due to cyclic loading	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.2.A-8	Lifting Anchor Blocks (A)	FR	Steel	Air – outdoor	Loss of material due to general corrosion, pitting, crevice corrosion	Chapter IV.S1,"Structures Monitoring Program" Chapter IV.S2, "Protective Coating Inspection and Maintenance Program."	Generic program
V.2.A-9	Concrete Overpack and Pad (accessible areas): Above-grade (B)	SS, HT	Plain Concrete, Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.2.A-10	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a site-specific AMP is needed to manage cracking and expansion due to reaction with aggregate of concrete in inaccessible areas. A site-specific AMP is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) for potentially reactive aggregates, aggregate concrete reaction is not significant.	Further evaluation to determine whether a site-specific AMP is needed
V.2.A-11	Concrete Overpack and Pad (accessible areas): Above-grade (B)	SS, HT	Plain Concrete, Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.2.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-12	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Further evaluation is required to determine if a site-specific AMP is needed to manage loss of strength due to concrete interaction with aluminum in inaccessible areas.	Further evaluation to determine whether a site-specific AMP is needed
V.2.A-13	Concrete Overpack and Pad (accessible areas): Above-grade (A or B)	RS, SS, HT	Plain Concrete, Reinforced Concrete	Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.2.A-14	Concrete Pad: (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program" Inaccessible Concrete Areas: For facilities with non-aggressive ground-water/soil, i.e., pH >5.5, chlorides <500 ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilities with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm), and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program

Table V.2.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-15	Concrete Pad: (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.2.A-16	Concrete Overpack and Pad (accessible areas): Above-grade (A or B)	RS, SS, HT	Plain Concrete, Reinforced Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.2.A-17	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for facilities that are located in moderate to severe weathering areas (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a site-specific AMP is needed. A site-specific AMP is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Code Section III Division 2), and subsequent inspections of accessible areas did not reveal degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a site-specific AMP is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental U.S. is shown in ASTM C33-90, Fig. 1.	Further evaluation, for facilities located in moderate to severe weathering areas

Table V.2.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-18	Concrete Overpack and Pad: Above-grade (A or B)	RS, SS, HT	Plain Concrete, Reinforced Concrete	Air – inside the module, uncontrolled or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.2.A-19	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program" Inaccessible Concrete Areas: For facilities with non-aggressive ground-water/soil, i.e., pH >5.5, chlorides <500 ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilities with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm), and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program
V.2.A-20	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.2.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-21	Concrete Pad (accessible areas): Above-grade (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.2.A-22	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a site-specific AMP is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in inaccessible areas. A site-specific AMP is not required if (1) there is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Further evaluation, if leaching is observed in accessible areas that impact intended function
V.2.A-23	Concrete Pad: Below-grade (B)	SS	Reinforced Concrete	Soil and water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter IV.S1, "Structures Monitoring Program." If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement
V.2.A-24	Concrete Pad: All (B)	SS	Reinforced Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter IV.S1, "Structures Monitoring Program." If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement

Table V.2.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-25	Concrete Overpack and Pad: All (A or B)	RS, SS, HT	Plain Concrete, Reinforced Concrete	Air – inside the module, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Site-specific AMP The implementation of 10 CFR 72 requirements and ASME Section XI, Subsection IWL would not enable identification of the reduction of strength and modulus of elasticity due to elevated temperature. Thus, for any portions of the concrete pad that exceed specified temperature limits, further evaluations are warranted. Subsection CC-3400 of ASME Code Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	Site-specific AMP Further evaluation, if temperature limits are exceeded
V.2.A-26	Coatings (if applied) (C)	SS Not ITS	Coating	Air – inside the module, uncontrolled or Air – outdoor	Loss of coating integrity due to blistering, cracking, flaking, peeling, or physical damage	Chapter IV.S2, “Protective Coating Monitoring and Maintenance Program”	Generic program
V.2.A-27	Moisture Barriers (caulking, sealants, and expansion joint fillers) (C)	SS Not ITS	Elastomers, rubber and other similar materials	Air – outdoor	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter IV.S1, “Structures Monitoring Program”	Generic program

Table V.2.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.A-28	Lightning Protection System (C)	SS Not ITS	Various materials	Air – outdoor	Loss of lightning protection due to wear, tear, damage, surface cracks, or other defects	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.2.A-29	Electrical Equipment subject to 10 CFR 50.49 EQ requirements (B)	Monitoring system	Various metallic and polymeric materials	Adverse localized environment caused by heat, radiation, oxygen, moisture, or voltage	Various degradation/ various mechanisms	EQ is a TLAA to be evaluated for the period of extended operation. See Section III.6, "Environmental Qualification of Electrical Equipment," for acceptable methods for meeting acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.2.A-30	Cathodic Protection Systems (B)	Cathodic protection of rein-forcing steel	Various materials	Embedded in concrete	Reduction of cathodic protection effect on bond strength due to degradation of cathodic protection current	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.2.B HI-STORM 100 System: Multipurpose Canister (MPC)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.2.B-1	MPC: Baseplate, shell, lid, port cover, closure ring, and associated welds; Fuel basket and fuel spacer (A)	CB, CC, HT, SS, FR	Stainless Steel: 304 SS, 304LN SS, 316 SS, 316LN SS	Air – inside the storage overpack, uncontrolled (external), Helium (internal)	Cumulative fatigue damage due to cyclic loading	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.2.B-2	MPC: Baseplate, shell, lid, closure ring, and associated welds; shield lid and bolting (A)	CB, CC, HT, SS, FR	Stainless steel: 304 SS, 304LN SS, 316 SS, 316LN SS	Air – inside the storage overpack, uncontrolled (external)	Cracking and leakage due to stress corrosion cracking when exposed to moisture and aggressive chemicals in the environment	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components" Chapter IV.M3, "Welded Canister Seal and Leakage Monitoring Program"	Generic program
V.2.B-3	MPC Internals: Fuel basket, fuel spacer, basket support; heat conduction elements; drain pipe, vent port; neutron absorber panels (enclosed in stainless steel sheathing) (A)	CC, CB, HT, SS, FR	Stainless Steel, aluminum alloy, borated aluminum or boron carbide/-aluminum alloy plate or BORAL composite	Helium	Degradation of heat transfer, radiation shield, criticality control, confinement boundary, or structural support functions of the MPC internals due to extended exposure to high temperature and radiation.	Chapter IV.M5, "Canister Structural and Functional Integrity Monitoring Program"	Generic program
V.2.B-4	MPC Internals: Fuel basket neutron absorber panels (enclosed in stainless steel sheathing) (A)	CC	Borated aluminum or boron carbide/-aluminum alloy plate or BORAL composite	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of neutron-absorbing materials is a TLAA to be evaluated for the period of extended operation. See Section III.4, "Time-Dependent Degradation of Neutron-Absorbing Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA

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V.3 Transnuclear Metal Spent-Fuel Storage Cask

V.3.1 System Description

The metal spent-fuel cask was developed by Transnuclear Inc. (TN) to transport/store irradiated spent-fuel assemblies at an ISFSI. The TN spent-fuel storage cask is a vertical canister with bolted lid closure and two metallic O-rings forming the seal. As a storage cask, it provides confinement, shielding, criticality control and passive heat removal independent of any other facility structures or components. There are three types of TN metal spent-fuel storage casks: TN-32, TN-40HT, and TN-68. The TN-32 cask, approved for use at Surry, North Anna and McGuire ISFSIs, accommodates 32 intact PWR fuel assemblies. Each fuel assembly is assumed to have a maximum initial enrichment not to exceed 3.85 w/o U-235 in uranium. Further assumptions limit the fuel to a maximum of 40,000 MWD/MTU burn-up, a minimum decay time of 7 years after reactor discharge and a maximum decay heat load of 0.847 kW per assembly for a tota1 of 27.1 kW for a cask.

The TN-40 cask, approved for use at the Prairie Island site-specific-licensed ISFSI, accommodates up to forty (40) 14 x 14 PWR fuel assemblies with or without fuel inserts. The maximum allowable initial enrichment of the fuel to be stored in a TN-40 cask is 5.0 weight percent U-235. The maximum bundle average burn-up, maximum decay heat, and minimum cooling time are 60,000 MWd/MTU, 0.80 kW per assembly, and 12 years, respectively. The cask is designed for a maximum heat load of 32 kW. Only undamaged fuel will be stored in the TN-40 casks.

The TN-68 cask, approved for use at the Peach Bottom general-licensed ISFSI, accommodates up to 68 BWR fuel assemblies. The maximum allowable initial lattice-average enrichment varies from 3.7 to 4.7 wt% U-235, depending on the B-10 areal density in the basket neutron absorber plates. The maximum bundle average burnup, maximum decay heat, and minimum cooling time are, respectively, 40 GWd/MTU, 0.312 kW/assembly, and 10 years for a 7x7 fuel, and 60 GWd/MTU, 0.441 kW/assembly, and 7 years for all other fuel. The cask is designed for a maximum heat load of 30 kW. Damaged fuel that can be handled by normal means may be stored in eight peripheral compartments fitted with damaged-fuel end caps designed to retain gross fragments of fuel within the compartment.

The following description of the TN dry storage casks is based on the USNRC SER for TN-32 (USNRC 1996) and the TN SARs for TN-32 (Transnuclear Inc. 2002, Transnuclear Inc. 2004) and TN-68 (Transnuclear Inc. 2005).

The TN-32 cask body is a right circular cylinder composed of the following components: confinement vessel with bolted lid closure, basket for fuel assemblies, gamma shield, pressure monitoring system, weather cover, trunnions, and neutron shield (Fig. V.3-1). The confinement boundary components are shown in Fig. V.3-2 and key dimensions for the TN-32 cask body are provided in Fig. V.3-3.

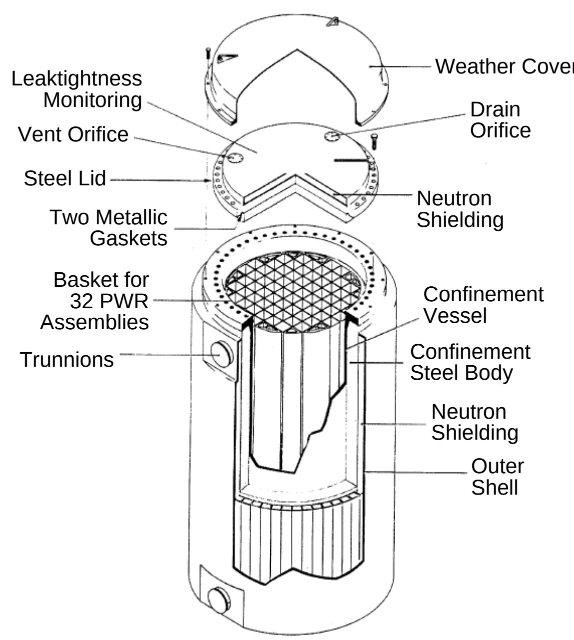

Figure V.3-1: Components of the Transnuclear dry shielded canister assembly

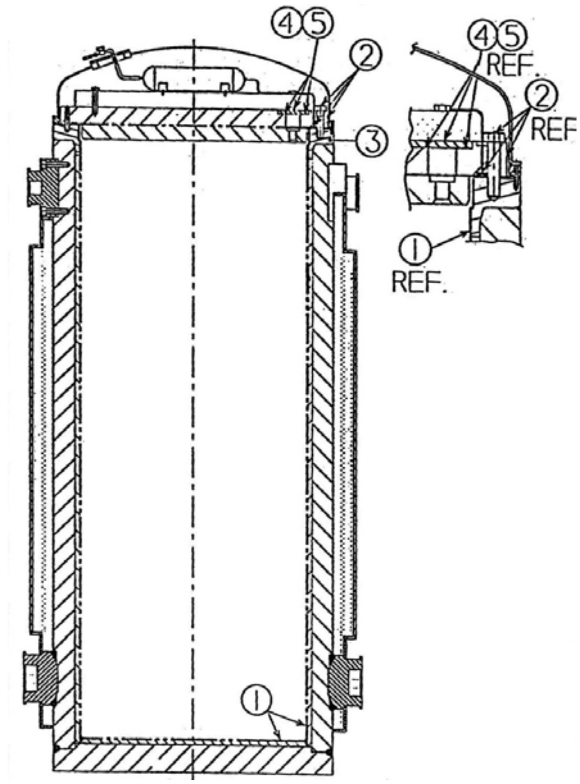

Figure V.3-2: Transnuclear storage cask confinement boundary components.

(a) Figure not to scale. Features exaggerated for clarity.
(b) Phantom line (— -- — --) indicates confinement boundary.
(c) Confinement boundary components are listed below:

1. Cask body inner shell
2. Lid assembly outer plate, closure bolts and inner o-ring
3. Bolting flange
4. Vent port cover plate, bolts and seals
5. Drain port cover plate, bolts and seals

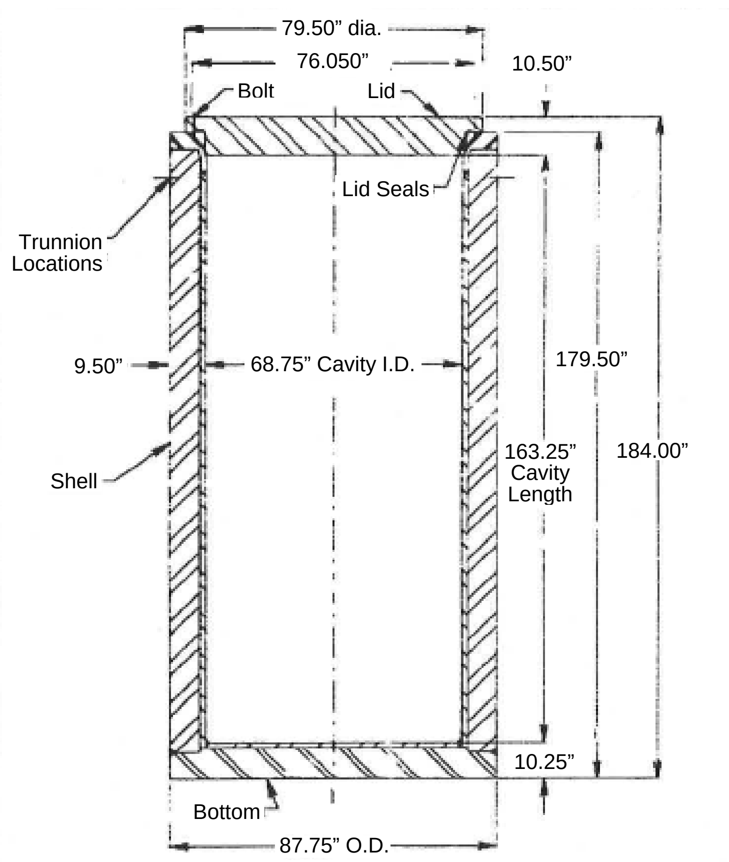

Figure V.3-3: TN-32 cask body key dimensions.

V.3.1.1 Containment Vessel, Closure Lid, and Pressure Monitoring System

The confinement vessel is made of welded cylindrical carbon steel (SA-203. Gr. A) inner shell, 38 mm (1.50 in.) thick, with an integrally welded carbon steel bottom closure. A flange forging is welded to the top of the inner shell to accommodate a flanged 4.5-in.-thick carbon steel lid closure fastened to the flanged forging with 48 bolts. In the TN-32 cask, the inner shell and bottom are lined with carbon steel. The inner shell has a sprayed metallic aluminum coating for corrosion protection. Surrounding the outside of the containment vessel wall is a steel gamma shield with a wall thickness of 8.0 inches. The bottom of the gamma shield has a thickness of 8.75 inches. The cask is sealed with one carbon steel closure lid, with a thickness of 10.5 inches, to the top flange of the containment vessel. The closure lid is secured to the cask body by 48 bolts.

The closure lid uses a double-barrier seal system with two metallic O-rings (Helicoflex seals) forming the seal. The seals are made of stainless steel with silver plating. The annular space between the metallic O-rings is connected to a pressure monitoring system (PMS) placed between the lid and the protective cover, also called weather cover (see Fig. V.3-4). Pressure in the tank of the PMS is maintained above the pressure in the cask cavity to prevent either flow of fission gases out of, or air into, the cask cavity, which, under normal storage conditions, is pressurized above atmospheric pressure with helium. The transducers/switches monitor the pressure in the annular space between the metallic O-rings to provide an indication of seal failure before any release is possible. Two identical transducers/switches are provided to ensure a functional system through redundancy.

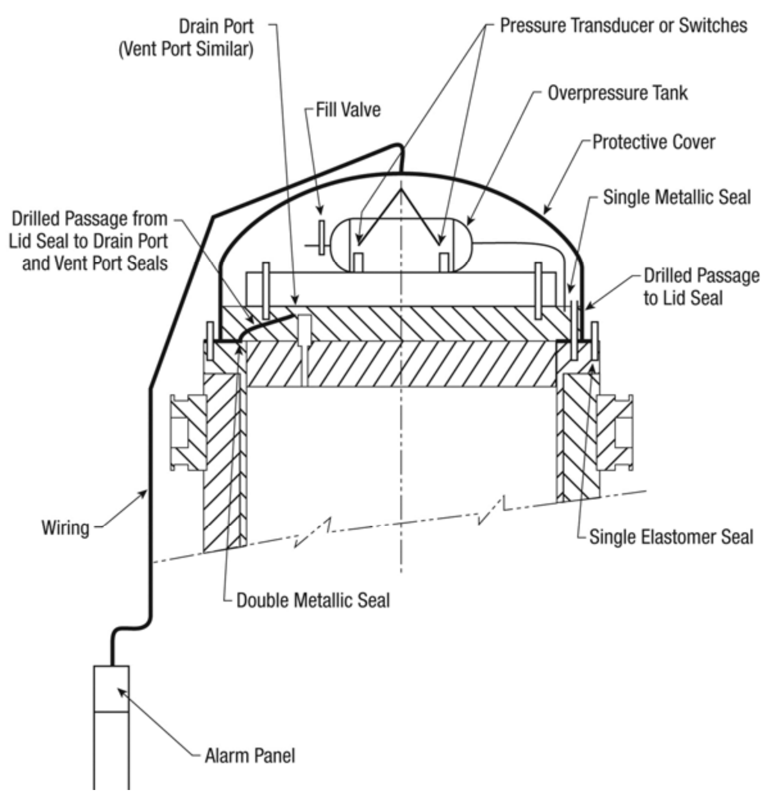

Figure V.3-4: TN-32 cask seal pressure monitoring system.

The cask body has four carbon steel trunnions that are welded to the gamma shield. Two of these are located near the top of the cylindrical steel forging, spaced 180 degrees apart, and are used for lifting the cask. The remaining two trunnions are 180 degrees apart and located near the bottom of the cask. The lower trunnions are used to rotate the unloaded cask between vertical and horizontal positions. The lifting trunnions have an effective diameter of 220 mm (8.67 in.) and are hollow to permit installation of neutron-shielding material and eliminate a path for neutron streaming.

The cask lid has three confinement access ports: drain port, vent port and overpressure system port. The drain and vent ports are covered by a bolted stainless steel closure plate having a double-barrier seal system with two metallic O-rings (Helicoflex seals) forming the seal, similar to the one used for the lid closure. The overpressure port is also covered by a bolted stainless steel closure plate but has a single metallic O-ring forming the seal. The closure lid has drilled interseal passageways connecting the annular space between the seals at each port to the annular space between the closure-lid seals, as shown in Fig. V.3-4. The cavity drain line penetrates the closure lid and terminates in the bottom of the cask cavity. This line is used to drain water from the cask cavity after underwater fuel loading. It is also used during the drying and helium back-filling of the cask cavity. The drain valve is of the quick-disconnect type and was not analyzed as part of the primary containment system. The cavity vent valve is identical to the drain valve. Overall, the cask is 5131 mm (202 in.) long and 2591 mm (102 in.) in diameter. The cask weighs approximately 115.5 tons (230,990 pounds) when loaded.

The all-metal Helicoflex seal used in the TN metal casks has a built-up structure with inner and outer liners, and a central helical energizing spring [see Figs. V.3-5 (a) and (b)]. Sealing is accomplished by plastic flow of the outer liner against the mating sealing surfaces. The helical spring aids in keeping a sufficient load against the outer liner to follow temperature fluctuations and small deformations. Helicoflex seals can be manufactured to meet leak-tight or better sealing criteria. Leakage rates of less than 1 x $l0^{-9}$ atm-cm/s (helium) can be maintained using seals with a slightly larger wire gauge for the internal spring than those for "standard" sealing. The seal is not generally considered reusable **(Warrant et al. 1989)**. The seals' rated service temperature is 280°C (536°F), per NUREG/CR-6886.

The confinement vessel has a cylindrical cavity with an inert gas atmosphere. The cavity is 1753 mm (69.0 in.) in diameter and 4140 mm (163 in.) long and holds a fuel basket with 32 compartments, each 221 mm (8.70 in.) square, to locate and support the PWR fuel assemblies. A PWR assembly typically consists of zircaloy fuel rods containing uranium dioxide fuel pellets. The fuel rods are assembled into a square array, spaced and supported laterally by grid structures with top and bottom fittings for vertical support and handling. The basket assembly also transfers heat from the fuel assembly to the cask body wall and provides neutron absorption to satisfy nuclear criticality requirements, especially during loading and unloading operations that occur underwater. During storage, with the cavity dry and sealed from the environment, criticality control measures within the cask are not necessary because of the low reactivity of the fuel in the dry cask and the assurance that no water can enter the cask during storage.

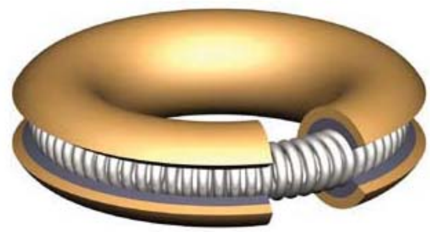

Figure V.3-5(a): Helicoflex seal

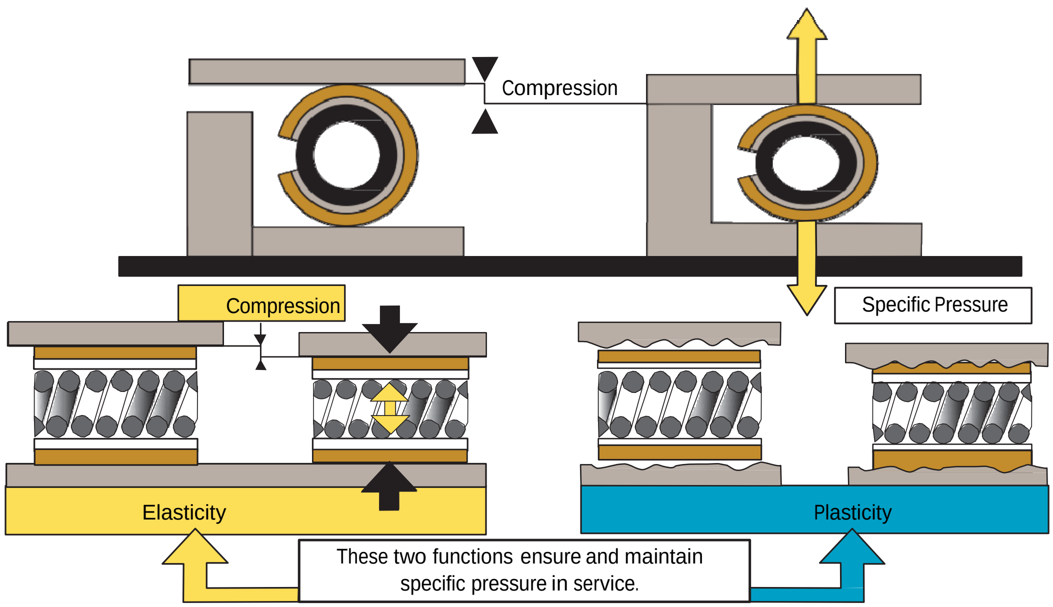

Figure V.3-5(b): A sketch illustrating the sealing concept of the Helicoflex seal

V.3.1.2 Fuel Basket Assemblies and Shieldings

The fuel cavities in the basket are formed by a sandwich of aluminum plates, BORAL plates, and stainless steel boxes. The stainless steel fuel-compartment box sections are attached by a series of stainless steel plugs that pass through the 12.7-mm (0.5-in.)-thick aluminum plates and the 1.02-mm (0.04-in.)-thick poison plates and are fusion-welded to both adjacent stainless steel box sections. The aluminum provides the heat conduction paths from the fuel assemblies to the cask cavity wall. The poison material provides the necessary criticality control. The basket is guided into the cask body and held in place by aluminum rails that run the axial length of the cask body, as shown in Fig. V.3-6.

Surrounding the outside of the confinement vessel wall is a steel gamma shield (SA-516, Grade 70) with a wall thickness of 203 mm (8.0 in.), as shown in Fig. V.3-7. The top end of the gamma shield is welded to the confinement vessel flange. The bottom end of the gamma shield is made of the same material and has a thickness of 222 mm (8.75 in.). The bolted closure lid provides the gamma shielding at the upper end of the cask body. Neutron emissions from the stored fuel are attenuated by a neutron shield, consisting of a borated polyester resin compound, enclosed in long aluminum boxes that surround the gamma shield. The resin compound is 114 mm (4.50 in.) thick and is cast into long, slender aluminum containers, which are held in place by a 13-mm (0.50-in.)-thick painted SA 516 Gr 70 steel shell constructed of two half-cylinders. Neutron emissions from the top of the cask are attenuated by a 102-mm (4.0-in.)-thick polypropylene disc, encased in a 6.35-mm (0.25-in.)-

thick steel shell and placed on the top of the closure lid. There is no neutron shielding provided on the bottom of the cask.

The inside surfaces of the inner shell and bottom have a sprayed metallic coating of aluminum for corrosion protection. The external surfaces of the cask are metal, sprayed or painted for ease of decontamination and corrosion protection. The neutron shield, PMS, and shield cap are placed on top of the cask after fuel is removed from the spent-fuel pool and loaded into the cask. A stainless steel overlay is applied to the O-ring seating surfaces on the body for corrosion protection. A protective cover, 9.5 mm (0.375 in.) thick, with a Viton O-ring is bolted to the top of the cask body to provide weather protection for the lid penetrations and other components, as shown in Fig. V.3-4.

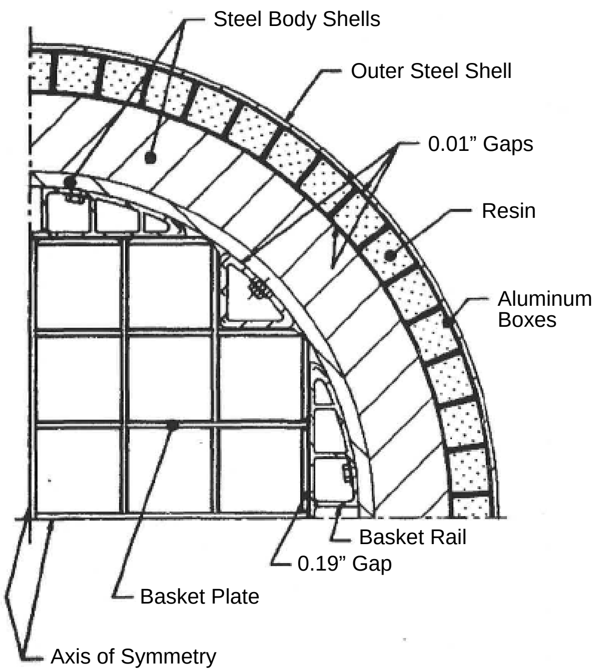

Figure V.3-6: Radial cross-section of TN-32 cask showing basket, basket rails and gamma and neutron shields.

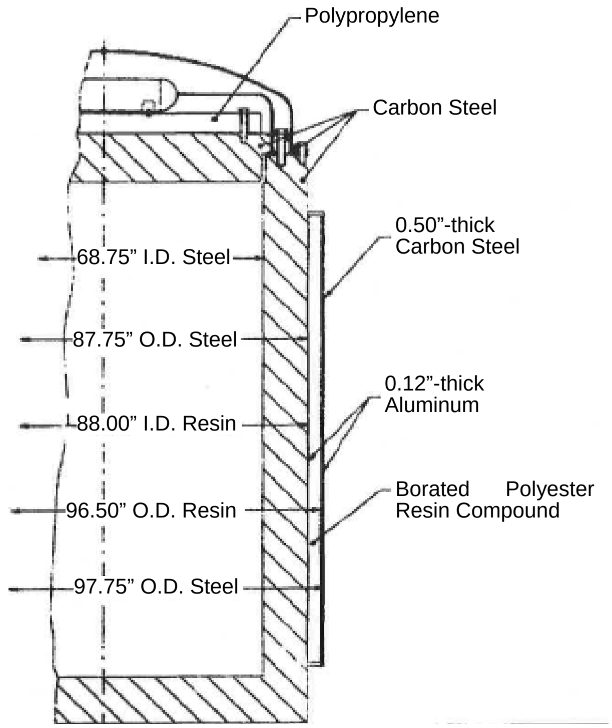

Figure V.3-7: TN-32 cask shielding configuration

The heat rejection capability of the cask maintains the maximum fuel rod clad temperature below 348°C (658°F), on the basis of normal operating conditions with a 27.1 kW decay heat load, 38°C (100°F) ambient air, and full insolation. The fuel assemblies are stored in an inert helium gas atmosphere. The cast shielding features of the cask are designed to maintain the maximum combined gamma and neutron surface dose rate at less than 200 mrem/hr under normal operating conditions.

V.3.1.3 Concrete Pad and Operating Experience

The casks are stored on a 3-ft-thick reinforced concrete slab in a free-standing, vertical orientation. Typically, two or three concrete pads are utilized at an ISFSI with each pad containing an array of several casks arranged in two rows. One possible configuration for a dry storage installation is shown in Fig. V.3-8. The operating experience for TN-32 casks includes several instances of chipped external coatings on the casks and corrosion of lid bolts and outer metallic seals due to intrusion of rainwater in the vicinity of the seal at the Surry site operated by Dominion. It was determined that the Conax connector seal for the electrical connector in the cask protective cover was leaking because of improper installation of the connectors. To reduce the likelihood of protective-cover leakage, the pressure sensing instrumentation was relocated outside the cover. This relocation required routing pressure sensor tubing through the side of the cover and mounting the pressure switches on the side of the cask. The original openings for the Conax connectors in the top of the protective cover were welded closed **(Virginia Electric 2002)**. Dominion has backfitted these covers to preclude leakage. Future covers will incorporate the backfit modification.

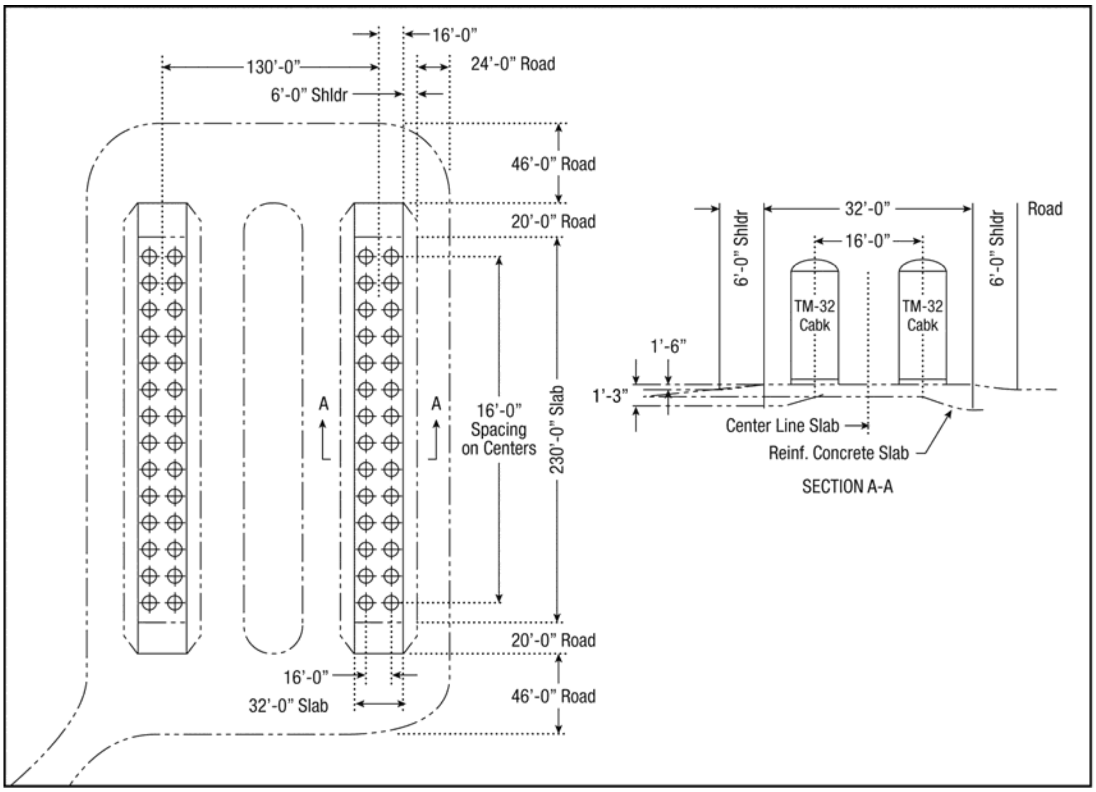

Figure V.3-8: Typical vertical storage of Transnuclear metal dry shielded canisters. Note that for TN-32, the spacing on centers is 16 ft.

V.3.2 Codes and Service Life

The minimum design life of TN-32 cask is 25 years. The ASME Boiler and Pressure Vessel Code is the governing code for design, fabrication, examination, and acceptance criteria of TN cask components. The containment vessel is designed to the ASME Code, Section III, Subsection NB. The basket assembly and other safety-related components (trunnions, neutron shieldings, protective cover) are designed to the ASME Code, Section III, Subsection NF and American Welding Society (AWS) Structural Welding Codes.

There is no concrete or reinforced concrete in the TN casks, except the concrete pad. The concrete pad is designed in accordance with ACI codes and standards and is designed with a nominal design concrete compressive strength of 3000 psi at 28 days.

The TN-32 cask is designed for 0.12 g horizontal ground motion, 0.08 g vertical ground motion, and 360 mph wind speed.

V.3.3 Current Inspection and Monitoring Program

Typical maintenance tasks involve occasional recalibration of seal monitoring instrumentation, visual inspection, and repainting the casks with protective coatings. Transnuclear Inc. suggests no special maintenance techniques. The metallic O-rings are designed to maintain their sealing capability until the cask is opened. If a drop in pressure in the overpressure system indicates a leak, all the gaskets can be replaced. The overpressure system has two identical pressure transducers/switches for redundancy. Replacements are necessary if they malfunction.

The aging management of TN casks in Surry ISFSI relies on the Dry Storage Cask Inspection Activities Program during the period of extended operation. The scope of program involves 1) the continuous pressure monitoring of the in-service dry storage casks, 2) the quarterly visual inspection of all dry storage casks that are in-service at the Surry ISFSI, 3) a visual inspection of the TN storage cask seal cover area, which is to be performed prior to the end of the original operating license period, and 4) the visual inspection of the normally inaccessible areas of casks in the event they are lifted in preparation for movement or an environmental cover is removed for maintenance.

The AMPs to manage aging effects for specific structures and components, the materials of construction, and environment of the TN metal spent-fuel storage cask are given in Tables V.3.A and V.3.B. In these tables, the DCSS components listed in the Structure and/or Component column are classified as “A”, “B”, or “C” according to importance to safety, as described in Section I.2.

V.3.4 References

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Table V.3.A Transnuclear Metal Spent-Fuel Storage Cask (A)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.3.A-1	Gamma Shield (A)	RS, HT, SS	Carbon Steel	Air – outdoor	Loss of material due to corrosion	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program
V.3.A-2	Top Neutron Shield (A)	CC	Polypro-pylene (encased in carbon steel)	Radiation and elevated temperature	Degradation of shielding properties due to exposure to high temperature and gamma and neutron radiation	Degradation of neutron-absorbing materials is a TLAA to be evaluated for the period of extended operation. See Section III.4, "Time-Dependent Degradation of Neutron-Absorbing Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.3.A-3	Top Neutron Shield Enclosure & Bolts (A)	SS	Carbon Steel; Low-Alloy Steel	Air, Leaking Rainwater –under the protective cover (external)	Loss of material due to corrosion	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program
V.3.A-4	Radial Neutron Shield (A)	RS	Borated Polyester (encased in aluminum)	Elevated temperatures and gamma radiation in air environment[1]	Degradation of shielding material due to radiation exposure	Degradation of radiation-shielding materials is a TLAA to be evaluated for the period of extended operation. See Section III.5, "Time-Dependent Degradation of Radiation-Shielding Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.3.A-5	Radial Neutron Shield Box Assembly (A)	RS	Aluminum; SA 516 Gr 70 steel	Air – outdoor	Loss of material due to corrosion	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program
V.3.A-6	Outer Shell (A)	SS	Carbon Steel	Air – outdoor	Loss of material due to corrosion	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program
V.3.A-7	Trunions (includes welds) (A)	SS	Carbon Steel	Air – outdoor	Loss of material due to corrosion	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program

Table V.3.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.3.A-8	Concrete Pad (A or B)	SS	Reinforced Concrete	Air (external)	Cracking, loss of bond, loss of material due to corrosion of embedded steel, expansion from reaction with aggregates, aggressive chemical attack, freeze-thaw and settlement	Chapter IV.S1,"Structural Monitoring Program" Inaccessible Concrete Areas: For facilities with non-aggressive ground-water/soil; i.e., pH >5.5, chlorides <500 ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations.	Generic program
V.3.A-9	Concrete (accessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.3.A-10	Concrete (accessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Further evaluation is required to determine if a site-specific AMP is needed to manage loss of strength due to concrete interaction with aluminum in inaccessible areas.	Further evaluation to determine whether a site-specific AMP is needed
V.3.A-11	Cathodic Protection Systems (B)	Cathodic protection of rein-forcing steel	Various materials	Embedded in concrete	Reduction of cathodic protection effect on bond strength due to degradation of cathodic protection current	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.3.B Transnuclear Metal Spent-Fuel Storage Cask (B)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.3.B-1	Inner Shell, Flange and Bottom (A)	CB, SS, RS, HT	Carbon Steel	Limited air (external), Helium (internal)	Cumulative fatigue damage/fatigue	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.3.B-2	Lid and Bolting (A)	CB, SS, RS, HT	Carbon Steel, low-alloy steel	Air, Leaking Rainwater –under the protective cover (external), Helium (internal)	Cumulative fatigue damage/fatigue	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2. "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.3.B-3	Cover Plates and Bolting: Drain, vent and overpressure port (A)	CB, SS, RS, HT	Stainless Steel, low-alloy steel	Air, Leaking Rainwater –under the protective cover (external), Helium (internal)	Cumulative fatigue damage/fatigue	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.5.B-4	Flange (A)	CB, SS, RS, HT	Carbon steel	Air-outdoor (external), Helium (internal)	Loss of material due to corrosion	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program
V.3.B-5	Lid and Bolting (A)	CB, SS, RS, HT	Carbon Steel, low-alloy steel	Air, Leaking Rainwater –under the protective cover (external), Helium (internal)	Loss of material due to corrosion,	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program
V.3.B-6	Bolting for Cover Plates: Drain, vent and overpressure port (A)	CB, SS, HT	Low-alloy steel	Air, Leaking Rainwater –under the protective cover (external), Helium (internal)	Loss of material due to corrosion	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program

Table V.3.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.3.B-7	Helicoflex Seals (includes stainless steel cladding on sealing surface): Lid, drain, vent and overpressure port closures (A)	CB	Aluminum, Silver, Stainless Steel, Ni-base Alloys	Air, Leaking Rainwater –under the protective cover (external), Helium (internal)	Loss of material due to corrosion (for Al only)	Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic program
V.3.B-8	Helicoflex Seals (includes stainless steel cladding on sealing surface): Lid, drain, vent and overpressure port closures (A)	CB	Aluminum, Silver, Stainless Steel, Ni-base Alloys	Air, Leaking Rainwater –under the protective cover (external), Helium (internal)	Loss of sealing forces due to stress relaxation and creep of the metallic O-rings, corrosion and loss of preload of the closure bolts	Chapter IV.M4, "Bolted Canister Seal and Leakage Monitoring Program"	Generic program
V.3.B-9	Pressure Monitoring System: Pressure sensor inner and outer housing and associated elastomer seals and bolts. (C)	Monitor-ing system	Steel, elasto-mers, rubber and similar materials	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.3.B-10	Confinement Vessel Inner Components Fuel Basket, top and bottom fittings, aluminum and stainless steel plates, neutron absorber plates, stainless steel plugs, basket rails, drain pipe (A)	CC, SS, HT, RS, FR	Stainless Steel, Alumi-num, BORAL composite	Helium	Degradation of heat transfer, radiation shield, criticality control, or structural support function due to extended exposure to high temperature and radiation	Chapter IV.M5, "Canister Structural and Functional Integrity Monitoring Program"	Generic program

Table V.3.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.3.B-11	Confinement Vessel Inner Components Fuel basket neutron absorber panels (A)	CC	BORAL; borated polyester resin	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of neutron-absorbing materials is a TLAA to be evaluated for the period of extended operation. See Section III.4, “Time-Dependent Degradation of Neutron-Absorbing Material,” for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA

[1] Small gaps may exist where metal-to-metal or metal-to-polymer subcomponents interface. These gaps contain either air or helium gases. The limited amount of oxygen in these locations will be depleted and oxidation will be arrested. Therefore, oxidation at these interfaces is not a concern.

V.4 NAC International S/T Storage Casks

V.4.1 System Description

NAC International (formerly Nuclear Assurance Corporation) has developed a number of cask and canister systems for the dry storage of spent nuclear fuel (SNF). Of these, three are listed in 10 CFR 72.214 as currently being approved under a general license for the storage of spent fuel under the conditions specified in their CoCs. These three storage systems, all of which are canister plus overpack designs, are NAC-MPC, NAC-UMS, and NAC-MAGNASTOR. In addition, NAC International developed four earlier stand-alone casks, all variations of the same design. These are the NAC-S/T, NAC-C28 S/T, NAC-I28 S/T, and NAC-STC casks. The NAC-I28 S/T design is presently being used at the Surry nuclear plant under a site-specific license. General licenses were also granted for the NAC-S/T and NAC-C28 S/T designs in 1990, but these licenses expired on August 17, 2010, and they are not presently in use. The NAC-STC cask is not presently licensed for spent-fuel storage, but it is licensed under CoC No. 9235 for fuel transport. Summary descriptions of all seven of these storage systems are given here.

V.4.1.1 NAC-S/T, C28 S/T, I28, and STC

NAC International, Inc., has developed four variations of the NAC S/T (storage/transfer) cask, namely the NAC-S/T (CoC 72-1002), the NAC-C28 (CoC 72-1003), NAC-I28 (CoC 72-1020), and NAC-STC (CoC 72-1013). All of these designs are stand-alone casks without the need for an overpack. Only the NAC-S/T and NAC-C28 casks are listed as approved spent-fuel storage casks under the current edition of 10 CFR 72.214, and their licenses expired on August 31, 2010. The NAC-I28 cask is currently approved for storage of spent fuel at Surry 1 and 2 under a site-specific license (Docket No. 72-2), and the NAC-STC cask is licensed for spent-fuel transport under CoC 71-9235. Because of the similarities in the design of these four casks, they will be described together here. Selected design parameters are summarized in Table V.4-1, and the configuration of the basic NAC-S/T cask is shown in Fig. V.4-1.

All four cask designs are comprised of a pair of concentric stainless steel cylinders separated by a poured-in chemical lead gamma radiation shield. A solid neutron shield surrounds the outer shell, which, in turn, is encased in a stainless steel shell approximately 6.35 mm (0.25 in.) thick. The fuel baskets are designed to hold 26 or 28 fuel assemblies. Six trunnions can be attached to the casks—four around the top and two on each side at the bottom.

Gamma and neutron radiation shielding is provided by lead, stainless steel, and Bisco NS4-FR, a poured-in-place solid borated synthetic polymer that surrounds the outer shell along the cavity region. The bottom and lid are also made of lead encased in a stainless steel shell. A stainless steel and Bisco neutron shield cap is placed on top of the cask after fuel loading to further reduce radiation.

The cask designs are sealed to maintain an inert helium atmosphere using a closure lid with a double-barrier seal system and two metallic O-ring seals. There are four access ports in the cask: (1) a cavity drain port, (2) a cavity vent port, (3) an inter-seal test line port, and (4) a pressure monitoring port.

The fuel basket is a right circular cylinder configuration with 26 or 28 aluminum fuel tubes that are separated and supported by an aluminum and stainless steel grid of spacers and tie bars. Sheets of

BORAL are attached to the outside of the tubes to absorb neutrons. Impact limiters made of aluminum honeycomb inside a stainless steel shell are attached to the top and bottom of both casks during transport and storage.

Table V.4-1 Parameters for Selected NAC International Dry Storage Casks. All are stand-alone casks without an overpack and all have bolted primary containment boundary closures.

Parameter	NAC-S/T	NAC-C28 S/T	NAC-I28 S/T	NAC-STC
Fuel Type	PWR	PWR	PWR	PWR/BWR
No. of Assemblies	26	28	28	26
Maximum Heat Load (kilowatts)	26	20	15.6	7.7-22.1[a]
Minimum Cooling Time (years)	5	10	10	5-19[a]
Maximum Fuel Burnup (MWd/ton)	35,000	35,000	35,000	32,000-45,000[a]
Storage/Transport Cask:				
Length [m (in.)]	4.66 (183.3)	4.66 (183.3)	4.66 (183.3)	4.90 (193)
Cavity Height [m (in.)]	4.22 (166)	4.22 (166)	4.22 (166)	4.19 (165)
Outer Diameter [m (in.)]	2.39 (94)	2.39 (94)	2.39 (94)	2.51 (99)
Inner Diameter [m (in.)]	1.65 (64.8)	1.65 (64.8)	1.65 (64.8)	1.80 (71)
Outer SS Shell Thick. [mm (in.)]	66.8 (2.63)	66.8 (2.63)	66.8 (2.63)	67.3 (2.65)
Inner SS Shell Thick. [mm (in.)]	38.1 (1.5)	38.1 (1.5)	38.1 (1.5)	38 (1.5)
Pb γ Shield Thick. [mm (in.)]	81.3 (3.2)	81.3 (3.2)	81.3 (3.2)	94 (3.7)
Neutron Shield Thick. [mm (in.)]	178 (7.0)	178 (7.0)	178 (7.0)	140 (5.5)
Base Thickness [mm (in.)]	224 (8.8)	224 (8.8)	224 (8.8)	348 (13.7)
Top Neutron Shield Cap Thick-ness [mm (in.)]	152.4 (6.0)	96.5 (3.8)	76 (3.0)	230 (9.0)
Top Lid Thickness [mm (in.)]	215.9 (8.5)	215.9 (8.5)	215.9 (8.5)	133 (5.25)
Loaded Weight [tonne (tons)]	95.3 (105.1)	<113 (<125)	98 (108)	107 (117)
Empty Weight [tonne (tons)]	73.6 (81.1)	75 (83)		
Currently Licensed for Storage	No	No	Yes[b]	No[c]
NRC Part 72 Docket	72-1002	72-1003	72-1020 72-2[b]	72-1013 (71-9235)[c]
Facilities Where Used	-	-	Surry 1, 2	-

[a] Depending upon fuel type.
[b] Site-specific license for use at Surry 1 and 2.
[c] Licensed for transport under CoC No. 9235, Rev. 12, Docket No. 71-9235, U.S. Nuclear Regulatory Commission, October 5, 2010.

As indicated above, only the NAC-I28 cask is presently being used for spent-fuel storage under a site-specific license at the Surry nuclear plant. The NAC-STC cask is currently licensed for the transport of spent PWR nuclear fuel as well as fuel from the LaCrosse BWR under CoC 71-9235.

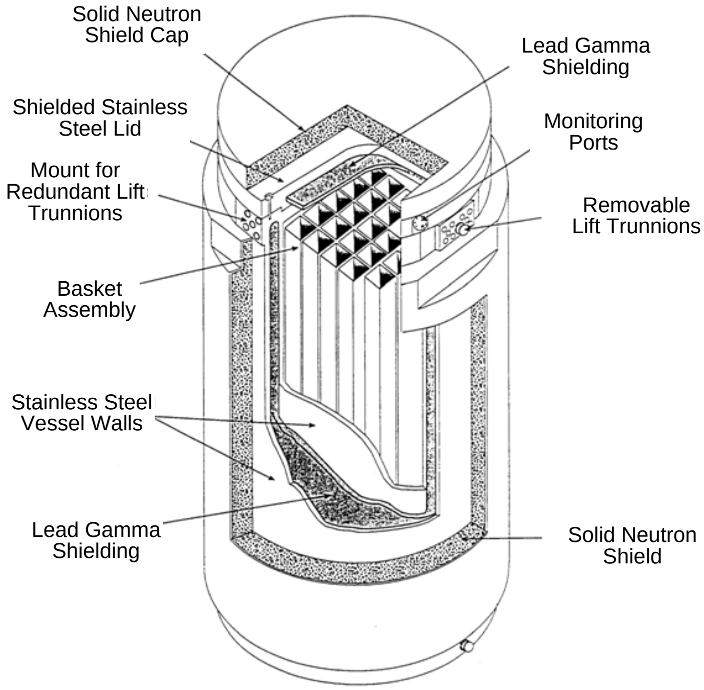

Figure V.4-1: NAC-S/T Metal Storage Cask (NEI 98-01, 1998)

V.4.1.2 NAC-MPC

The NAC-MPC system, made by the Nuclear Assurance Corporation, is a metal DCSS designed to store intact PWR fuel assemblies. Certificate of Compliance No. 1025 for this system was originally issued on April 10, 2000, and was most recently amended (Amendment 4) on October 27, 2004. The principal components of the NAC-MPC storage system are the transportable storage canister (TSC), the vertical concrete cask, and the transfer cask. As the name implies, the NAC-MPC system is designed for both the transport and storage of SNF, and the dual-purpose TSC is licensed for transport in the NAC-STC transportation cask under CoC 71-9235. The NAC-MPC system is presently in use at the Yankee Rowe and Haddam Neck (Connecticut Yankee) nuclear power plants. Design parameters for the NAC-MPC systems for these two plants are given in Table V.4-2.

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 constitute the confinement boundary. The stainless steel fuel basket is a right circular cylinder configuration with up to 36 fuel tubes (for Yankee Rowe Class fuel) and up to 26 fuel tubes (for Haddam Neck 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, using this storage system.

For the Yankee Rowe Class MPC, an alternative fuel basket design with enlarged fuel tubes in the four corner locations has also been authorized. In this alternative configuration, the BORAL sheet and stainless steel cover are removed from each side of the fuel tube in the four corner locations.

Aluminum heat transfer disks are spaced midway between the support disks and are the primary path for conducting heat from the spent-fuel assemblies in the TSC wall.

The vertical concrete cask serves as the storage overpack for the TSC and 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 vertical concrete cask has an annular air passage to allow the natural circulation of air around the TSC. The air inlet and outlet vents take non-planar paths to the vertical concrete cask cavity to minimize radiation streaming. The spent-fuel decay heat is transferred from the fuel assemblies to the tubes in the fuel basket and through the heat-transfer disks to the TSC wall. Heat flows by convection from the TSC wall to the circulating air, as well as by radiation from the TSC wall to the vertical concrete cask liner. The heat flow to the circulating air from the TSC wall and the vertical concrete cask liner is exhausted through the air outlet vents. The top of the vertical concrete cask is closed by a shield plug, consisting of a carbon steel plate for gamma shielding and solid neutron shielding material, covered by a carbon steel lid. The lid is bolted in place and has tamper-indicating seals on two of the bolts.

The transfer cask provides shielding during TSC movements between work stations, the vertical concrete cask, or the transport cask. It is a multi-wall (steel/lead/BISCO NS-4-FR/steel) design and has a bolted top retaining ring to prevent a loaded canister from being inadvertently removed through the top of the transfer cask. Hydraulically operated retractable bottom shield doors on the transfer cask are used during unloading operations. To minimize contamination on the TSC, clean water is circulated in the gap between the transfer casks and the TSC during spent-fuel pool loading operations.

V.4.1.3 NAC-UMS

The NAC Universal Multi-Purpose Canister 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 cask, and a transfer cask. The NAC-UMS system received 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 NAC-UMS System is presently in use at the Maine Yankee, Palo Verde, Catawba and McGuire nuclear plants. Design parameters for the NAC-UMS system are given in Table V.4-2.

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.

Table V.4-2 Selected Parameters for NAC-MPC, NAC-UMS, and NAC-MAGNASTOR Dry Storage Systems (adapted from EPRI 1021048).

Parameter	NAC-MPC		NAC-UMS		NAC-MAGNASTOR	
Fuel Type	Yankee Rowe PWR	Haddam Neck PWR	PWR	BWR	PWR	BWR
No. of Assemblies	36	24–26	24	56	37	87
Maximum Heat Load (kilowatts)	12.5	17.5	23	23	35.5	33
Minimum Cooling Time (years)	8–24	6	5–15	5–26	4	4
Maximum Fuel Burnup (MWd/ton)	36,000	43,000	60,000	45,000	60,000	60,000
<u>Dual-Purpose Canister:</u>						
Length (m)	3.11	3.86	4.45–4.87		4.69–4.87	
(in.)	(122.5)	(151.8)	(175.1–191.8)		(184.8–191.8)	
Cavity Height (m)	2.88	3.61	4.15–4.57		4.38–4.56	
(in.)	(113.5)	(142)	(163.3–180.0)		(172.5–179.5)	
Outer Diameter [m (in.)]	1.79 (70.6)		1.70 (67.1)		1.83 (72)	
Inner Diameter [m (in.)]	1.76 (69.4)		1.67 (65.8)		1.80 (71)	
Wall Thickness [mm (in.)]	15 (0.6)		15 (0.6)		13 (0.5)	
Base Thickness [mm (in.)]	25 (1.0)	46 (1.8)	46 (1.8)		69.9 (2.75)	
Structural Lid Thickness [m (in.)]	76 (3.0)		76 (3.0)		229 (9.0)	
Loaded Weight (tonne)	24.8 (27.4)	29.8 (32.9)	32.0–34.5		46.0	46.4
(tons)			(35.3–38.0)		(50.75)	(51.25)
<u>Transfer Cask:</u>						
Length (m)	3.39	4.14	4.5–4.89		4.84	
(in.)	(133.4)	(162.9)	(177.3–192.6)		(190.62)	
Outer Diameter (m)			2.16		2.24	
(in.)			(85.3)		(88)	
Loaded Weight with water (tonne)	61.45	78.34	90.6–97.20		104.1	
(tons)	(67.7)	(86.4)	(99.9–107.1)		(114.8)	
<u>Storage Cask:</u>						
Length [m (in.)]	3.25 (128)		3.45 (136)		3.45 (136)	
Outer Diameter (m)	4.06		5.31–5.70		5.54–5.72	
(in.)	(160)		(209.2–224.5)		(218.3–225.3)	
Loaded Weight (tonne)			140.0–146.9		145.15–145.6	
(tons)			(154.4–162.0)		(160.0–160.5)	
Currently Licensed for Storage	Yes		Yes		Yes	
NRC Part 72 Docket	72–1025		72–1015		72–1031	
Facilities Where Used	Yankee Rowe	Haddam Neck	Maine Yankee Palo Verde Catawba, McGuire		McGuire	

The storage overpack, designated the vertical concrete cask, 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 vertical concrete cask configurations of different lengths for PWR and site-specific contents and two vertical concrete cask configurations of different lengths for BWR contents.

The transfer cask is used for the vertical transfer of the TSC between work stations and the vertical concrete cask or the 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 vertical concrete cask or UMS transportation cask. Figure V.4-2 shows the transfer configuration in which a transfer cask transfers a loaded TSC to a vertical concrete cask for the UMS system.

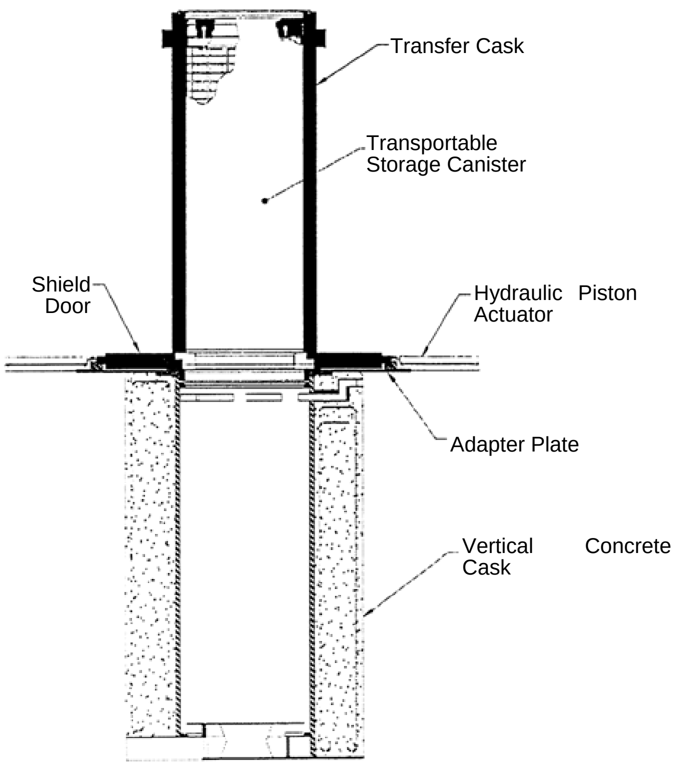

Figure V.4-2: NAC-UMS Dual-Purpose Storage System, CoC #72-1015 (NEI 98-01, May 1998).

V.4.1.4 NAC-MAGNASTOR

The NAC-MAGNASTOR System is a dual-purpose (storage and transport) canister system with a maximum capacity of 37 PWR fuel assemblies or 87 BWR assemblies. The storage component includes a TSC with a welded closure, a concrete storage cask, and a transfer cask. The NAC-MAGNASTOR system received 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 first two of a series of 20 NAC-MAGNASTOR systems were delivered to the Mcguire nuclear plant in late 2010. Design parameters for the NAC-MAGNASTOR system are given in Table V.4-2.

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 closure lid is positioned inside the TSC on the lifting lugs above the fuel basket assembly following fuel loading. After the closure lid is placed on the TSC, the TSC is moved to a workstation, and the closure lid is welded to the TSC. The vent and drain ports are penetrations through the lid, which provide access for auxiliary systems to drain, dry, and backfill the TSC. The drain port has a threaded fitting for installing the drain tube. The drain tube extends the full length of the TSC and ends in a sump in the base plate. The vent port also provides access to the TSC cavity for draining, drying, and backfilling operations. Following completion of backfilling, the port covers are installed and welded in place.

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 structural steel inner liner and base. The reinforced concrete wall and steel liner provide the neutron and gamma radiation shielding for the stored spent fuel. Inner and outer reinforcing steel (rebar) assemblies are encased within the concrete. The reinforced concrete wall provides the structural strength to protect the TSC and its contents in natural-phenomena events such as tornado wind loading and wind-driven missiles and during non-mechanistic tip-over events. The concrete surfaces remain accessible for inspection and maintenance over the life of the cask, so that any necessary restoration actions may be taken to maintain shielding and structural conditions. The concrete cask provides an annular air passage to allow the natural circulation of air around the TSC to remove the decay heat from the contents. The lower air inlets and upper air outlets are steel-lined penetrations in the concrete cask body. Each air inlet/outlet is covered with a screen. The weldment baffle directs the air upward and around the pedestal that supports the TSC. Decay heat is transferred from the fuel assemblies to the TSC wall by conduction, convection, and radiation. Heat is removed by conduction and convection from the TSC shell to the air flowing upward through the annular air passage and exhausting out through the air outlets. The passive cooling system is designed to maintain the peak fuel cladding temperature below acceptable limits during long-term storage. The concrete cask thermal design also maintains the bulk concrete temperature below the American Concrete Institute limits under normal operating conditions. The inner liner of the concrete cask incorporates standoffs that provide lateral support to the TSC in side-impact accident events. A carbon steel and concrete lid is bolted to the top of the concrete cask. The lid reduces skyshine radiation and provides a cover to protect the TSC from the environment and postulated tornado missiles.

The transfer cask provides shielding during TSC movements between work stations, the concrete cask, or the transport cask. It is a multiwall (steel/lead/Bisco NS-4-FR/steel) design with retractable (hydraulically operated) bottom shield doors that are used during loading and unloading operations. During TSC loading and handling operations, the shield doors are closed and secured. After placement of the transfer cask on the concrete cask or transport cask, the doors are retracted using hydraulic cylinders and a hydraulic supply. The TSC is then lowered into a concrete cask for storage or into a transport cask for offsite shipment. Sixteen penetrations, eight at the top and eight at the bottom, are available to provide a water supply to the transfer cask annulus. Penetrations not used for water supply or draining are capped. The transfer cask annulus is isolated using inflatable seals located between the transfer cask inner shell and the TSC near the upper and lower ends of the transfer cask. During TSC closure operations, clean water is added through these penetrations into the annulus region to remove heat generated by the spent-fuel contents. The cooling-water circulation is maintained through completion of TSC activities and is terminated to allow movement of the transfer cask for TSC transfer operations. A similar process of clean-water circulation is used during in-pool fuel loading to minimize contamination of the TSC outside surfaces. The transfer cask penetrations can also be used for the introduction of forced air or gas at the bottom of the transfer cask to achieve cooling of the TSC contents in case of the failure of the cooling water system. Alternatively, the loaded TSC may be returned to the spent-fuel pool for in-pool cooling.

A rendering of the MAGNASTOR storage configuration and a cutaway of the storage overpack are provided in Fig. V.4-3.

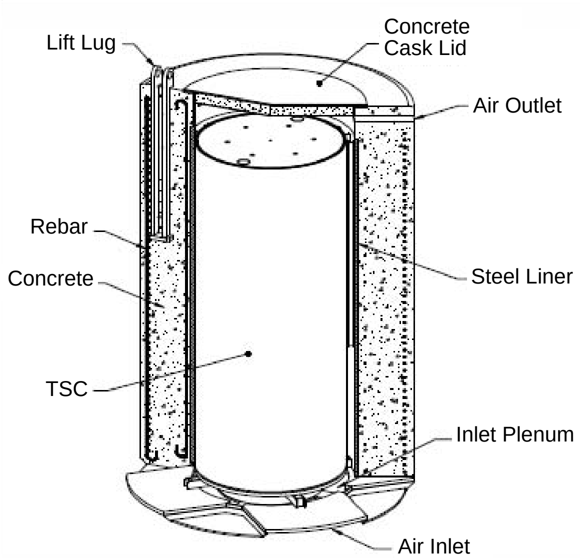

Figure V.4-3: NAC-MAGNASTOR Dual-Purpose Storage/Transport System, CoC # 72-1031 (Pennington, 2005).

V.4.2 Design Codes and Service Life

The NAC-UMS Maine Yankee canister and fuel basket assembly are designed, fabricated, and inspected in accordance with ASME Boiler and Pressure Vessel Code Section III rules for Class 1 components and core support structures and are Code stamped “N” and “NPT.” The NAC-MPC and MAGNASTOR canisters and fuel basket structures are designed and fabricated in general compliance with the ASME Code, Section III, Division 1, Subsections NB and NG, respectively. However, some exceptions are taken, and the components are not code stamped. The American Concrete Institute Specifications ACI 349 (1985) and ACI 318 (1995) govern the concrete cask design and construction, respectively, for all three of these storage systems.

The design basis for temperature for the Maine-Yankee UMS storage system is a 76°F maximum average yearly temperature. The 3-day average ambient temperature is 106°F or less, and the allowed temperature extremes, averaged over a 3-day period, are greater than -40°F and less than 133°F. The design basis earthquake seismic acceleration levels at the top surface of the ISFSI pad are 0.38g in the horizontal direction and 0.253g in the vertical direction. The maximum heat load is 23 kW for both PWR and BWR fuel. The NAC-UMS system is designed and analyzed for a 50-year service life.

For the NAC-MPC storage system, the maximum average yearly temperature is 75°F. The design basis 3-day average ambient temperature is 100°F or less, and the allowed temperature extremes, averaged over a 3-day period, are greater than -40°F and less than 125°F. The design basis earthquake seismic acceleration levels at the top surface of the ISFSI pad are 0.25g in the horizontal direction and 0.167g in the vertical direction. The maximum heat load is 12.5 kW for Yankee Rowe PWR fuel and 17.5 kW for Haddam Neck PWR fuel. The NAC-MPC system is designed and analyzed for a minimum 50-year service life.

For the NAC-MAGNASTOR storage system, the maximum average yearly temperature is 76°F. The design basis 3-day average ambient temperature is 106°F or less, and the allowed temperature extremes, averaged over a 3-day period, are greater than -40°F and less than 133°F. The maximum design basis earthquake acceleration at the ISFSI pad top surface to prevent cask tip-over is ≤0.37g in the horizontal direction and ≤0.25g in the vertical direction. The maximum heat load is 33 kW for BWR fuel and 35.5 kW for PWR fuel. The design life for the NAC-MAGNASTOR system is 50 years.

The maximum surface dose rates for the NAC-UMS concrete cask are not to exceed 50 mrem/hour (neutron + gamma) on the side (on the concrete surfaces), 50 mrem/hour (neutron + gamma) on the top, and 100 mrem/hour (neutron + gamma) at air inlets and outlets. For the NAC-MPC concrete cask, the corresponding limits are 50 mrem/hour (neutron + gamma) on the side (on the concrete surfaces), 55 mrem/hour (neutron + gamma) on the top, and 200 mrem/hour (neutron + gamma) average of the measurements at the air inlets and outlets. The dose rates for the NAC-MAGNASTOR concrete cask are not to exceed 95 mrem/hour gamma and 5 mrem/hour neutron on the vertical concrete surfaces and 450 mrem/hour (neutron + gamma) on the top.

V.4.3 Current Inspection and Monitoring Program

For the NAC canister designs, the canister shield-lid-to-shell weld is performed in the field following fuel assembly loading. The canister is then pneumatically pressure tested, although limited accessibility for leakage inspections precludes an ASME Code-compliant hydrostatic test. The shield lid-to-shell weld is also leak tested to the leak-tight criteria of ANSI N14.5. The vent port and drain port cover welds are examined by root and final liquid penetrant (PT) examination. If the weld is completed in a single weld pass, only a final surface PT examination is performed. The vent port and drain port cover welds are not pressure tested, but are tested to the leak-tight criteria of ANSI N14.5. The structural lid enclosure weld is not pressure tested, but is examined by progressive PT or ultrasonic testing and final surface PT.

The assembled storage systems at the ISFSI are subject to daily air inlet and outlet temperature monitoring, which may be done directly or remotely. A visual inspection must be performed if a decline in thermal performance is noted. All NAC storage systems in use at an ISFSI must be inspected within 4 hours after the occurrence of an off normal, accident or natural-phenomena event in the area of the ISFSI. This inspection should specifically verify that all the concrete cask inlets and outlets are not blocked or obstructed. At least one-half of the inlets and outlets on each concrete cask must be cleared of blockage or debris within 24 hours to restore air circulation. The concrete cask and canister must also be inspected if they experience a drop or a tipover. Following a natural-phenomena event, the ISFSI site must be inspected to verify that the concrete casks have not been repositioned so as to result in higher dose rates at the ISFSI boundary.

Additionally, thermal testing is to be performed for the first NAC-UMS system placed in service with a heat load ≥10 kW and the first NAC-MAGNASTOR system with a heat load ≥30 kW. A letter report summarizing the results of the measurements with respect to analyses of the actual canister content must then be submitted to the NRC in accordance with 10 CFR 72.4 within 60 days of placing the loaded cask on the ISFSI pad. The report is to include a comparison of the calculated mass flow of the storage system at the loaded heat load to the measured mass flow. A report is not required for the systems that are subsequently loaded, provided that the performance of the first system placed in service with a heat load of ≥10 kW for the NAC-UMS storage system or ≥30 kW for the NAC-MAGNASTOR system is demonstrated by the comparison of the calculated and measured mass flow rates.

The AMPs to manage aging effects for specific structures and components, the materials of construction, and environment of the NAC International S/T storage cask are given in Tables V.4.A and V.4.B. In these tables, the DCSS components listed in the Structure and/or Component column are classified as “A”, “B”, or “C” according to importance to safety, as described in Section I.2.

V.4.4 References

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Certificate of Compliance No. 1003 for the NAC C28 S/Y Cask, Rev. 0, Docket No. 72-1003, ML033020125, U.S. Nuclear Regulatory Commission, Washington, DC, August 17, 1990.

Certificate of Compliance No. 71-9235 for the NAC STC Cask, Rev. 12, Docket No. 71-9235, ML102780430, U.S. Nuclear Regulatory Commission, Washington, DC, October 5, 2010.

Certificate of Compliance No. 72-1015 for NAC-UMS Spent Fuel Storage Cask, Appendix A, “Technical Specifications, Amendment 5,” ML090120459, U.S. Nuclear Regulatory Commission, Washington, DC, January 12, 2009.

Certificate of Compliance No. 72-1025 for NAC-MPC Spent Fuel Storage Cask, Appendix A, “Technical Specifications, Amendment 4,” ML043020537, U.S. Nuclear Regulatory Commission, Washington, DC, October 27, 2004.

Certificate of Compliance No. 72-1031 for NAC-MAGNASTOR Spent Fuel Storage Cask, Appendix A, "Technical Specifications, Amendment 0," ML090350566, U.S. Nuclear Regulatory Commission, Washington, DC, February 4, 2009.

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Renewed Materials License No. SNM-2501, Surry Independent Spent Fuel Storage Installation, Appendix A, "Technical Specifications for Safety," Docket 72-2, ML050600021, U.S. Nuclear Regulatory Commission, Washington, DC, February 25, 2005.

Rigby, D. R., Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel, United States Nuclear Waste Technical Review Board, December 2010.

Virginia Electric and Power Company, Surry Independent Spent Fuel Storage Installation Final Safety Analysis Report Revision 16, Docket No. 72-2, License No. SNM-2501, June 2004.

Table V.4.A NAC International S/T Storage Casks: Storage Overpack (NAC-MPC, NAC-UMS, and NAC-MAGNASTOR) and Pad

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.8.A-1	Storage overpack: steel inner liner, base, shield plug, and lid (A or B)	SS, HT, RS, FR	Carbon or low-alloy steel	Outside air or marine environment	Loss of material due to general corrosion, pitting, crevice corrosion	Chapter IV.S1,"Structures Monitoring Program" Chapter IV.S2, "Protective Coating Monitoring and Maintenance Program"	Generic program
V.4.A-2	Storage overpack: Overpack concrete radiation shield, pedestal shield, and overpack lid shield (A)	RS, SS	Reinforced Concrete	Radiation and elevated temperature	Reduction of strength and modulus of concrete and degradation of shielding performance due to long-term exposure to high temperature and gamma radiation	Site-specific AMP The compressive strength and shielding performance of reinforced concrete is maintained by ensuring that the minimum concrete density is achieved during construction and the allowable concrete temperature and radiation limits are not exceeded. Further evaluations are warranted if these specified limits for temperature and gamma radiation are exceeded during service. Subsection CC-3400 of ASME Code Section III, Division 2, specifies that for normal operation, concrete temperature shall not exceed 66°C (150°F) for a long period and for accident conditions, concrete temperature shall not exceed 93°C (200°F) for short periods. Also, a gamma radiation dose of 10^{10} rads may cause significant reduction of strength. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are incorporated in the design calculations.	Site-specific AMP Further evaluation, if temperature and gamma radiation limits are exceeded

Table V.4.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.4.A-3	Storage overpack: Overpack concrete radiation shield, pedestal shield, and overpack lid shield (A)	RS, SS	Reinforced Concrete	Radiation and elevated temperature	Reduction of strength and modulus of concrete and degradation of shielding performance due to reaction with aggregate of concrete in inaccessible areas	Further evaluation is required to determine if a site-specific AMP is needed to manage cracking and expansion due to reaction with aggregate of concrete in inaccessible areas. A site-specific AMP is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) for potentially reactive aggregates, aggregate concrete reaction is not significant.	Site-specific AMP Further evaluation to determine whether a site-specific AMP is needed
V.4.A-4	Ventilation air openings: Air ducts, screens, gamma shield cross plates (A)	HT	Carbon or low-alloy steel	Air – inside the module, uncontrolled or Air – outdoor or marine environment	Loss of material and coating degradation due to corrosion and wear; cracking due to stress corrosion cracking	Chapter IV.M2, “Ventilation Surveillance Program.”	Generic program
V.4.A-5	Ventilation air openings: Air ducts, screens, gamma shield cross plates (A)	HT	Carbon or low-alloy steel	Air – inside the module, uncontrolled or Air – outdoor	Reduced heat convection capacity due to blockage	Chapter IV.M2, “Ventilation Surveillance Program.”	Generic program
V.4.A-6	Anchor Studs (for anchored cask) (A)	SS	SA-193, SA-354, SA-479, SA-540, SA-564, SA-574, SA-638	Normal air or marine environment – outdoor	Loss of preload due to self loosening; loss of material due to corrosion; cracking due to stress corrosion cracking	Chapter IV.S1, “Structures Monitoring Program”	Generic program

Table V.4.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.4.A-7	Anchor Studs (for anchored cask) (A)	SS	SA-193, SA-354, SA-479, SA-540, SA-564, SA-574, SA-638	Air – outdoor	Cumulative fatigue damage due to cyclic loading	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, “Fatigue of Metal and Concrete Structures and Components,” for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.4.A-8	Lifting Lugs and Trunnions (A)	FR	Steel	Air – outdoor	Loss of material due to general corrosion, pitting, crevice corrosion	Chapter IV.S1,”Structures Monitoring Program” Chapter IV.S2, “Protective Coating Inspection and Maintenance Program”	Generic program
V.4.A-9	Concrete Overpack and Pad (accessible areas): Above-grade (B)	SS, HT	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter IV.S1, “Structures Monitoring Program”	Generic program
V.4.A-10	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a site-specific AMP is needed to manage cracking and expansion due to reaction with aggregate of concrete in inaccessible areas. A site-specific AMP is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) for potentially reactive aggregates, aggregate concrete reaction is not significant.	Further evaluation to determine whether a site-specific AMP is needed

Table V.4.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.4.A-11	Concrete (accessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.4.A-12	Concrete (inaccessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Further evaluation is required to determine if a site-specific AMP is needed to manage loss of strength due to concrete interaction with aluminum in inaccessible areas.	Further evaluation to determine whether a site-specific AMP is needed
V.4.A-13	Concrete Overpack and Pad (accessible areas): Above-grade (A or B)	RS, SS, HT	Plain Concrete, Reinforced Concrete	Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program" Inaccessible Concrete Areas For facilities with non-aggressive ground-water/soil; i.e., pH >5.5, chlorides <500ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilities with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm), and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program

Table V.4.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.4.A-14	Concrete Pad: (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.4.A-15	Concrete Overpack and Pad (accessible areas): Above-grade (A or B)	RS, SS, HT	Plain Concrete, Reinforced Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.4.A-16	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for facilities that are located in moderate to severe weathering areas (weathering index >100 day-inch/yr) (NUREG-1557) to determine if a site-specific AMP is needed. A site-specific AMP is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Code Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a site-specific AMP is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental U.S. is shown in ASTM C33-90, Fig. 1.	Further evaluation, for plants located in moderate to severe weathering areas
V.4.A-17	Concrete Overpack and Pad: Above-grade (A or B)	RS, SS, HT	Plain Concrete, Reinforced Concrete	Air – inside the module, uncontrolled, or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.4.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.4.A-18	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.4.A-19	Concrete Pad (accessible areas): Above-grade (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.4.A-20	Concrete Pad (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a site-specific AMP is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in inaccessible areas. A site-specific AMP is not required if (1) there is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Further evaluation, if leaching is observed in accessible areas that impact intended function
V.4.A-21	Concrete Pad: Below-grade (B)	SS	Reinforced Concrete	Soil and water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter IV.S1, "Structures Monitoring Program." If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement

Table V.4.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.8.A-22	Concrete Pad: All (B)	SS	Reinforced Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter IV.S1, "Structures Monitoring Program." If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement
V.4.A-23	Concrete Overpack and Pad: All (A or B)	RS, SS, HT	Plain Concrete, Reinforced Concrete	Air – inside the module, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Site-specific AMP The implementation of 10 CFR 72 requirements and ASME Code Section XI, Subsection IWL would not enable identification of the reduction of strength and modulus of elasticity due to elevated temperature. Thus, for any portions of concrete pad that exceed specified temperature limits, further evaluations are warranted. Subsection CC-3400 of ASME Code Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	Site-specific AMP Further evaluation, if temperature limits are exceeded
V.4.A-24	Coatings (if applied) (C)	SS Not ITS	Coating	Air – inside the module, uncontrolled, or Air – outdoor	Loss of coating integrity due to blistering, cracking, flaking, peeling, or physical damage	Chapter IV.S2, "Protective Coating Monitoring and Maintenance Program"	Generic program

Table V.4.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.4.A-25	Moisture Barriers (caulking, sealants, and expansion joint fillers) (C)	SS Not ITS	Elastomers, rubber and other similar materials	Air – outdoor	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter IV.S1, “Structures Monitoring Program”	Generic program
V.4.A-26	Lightning Protection System (C)	SS Not ITS	Various materials	Air – outdoor	Loss of lightning protection due to wear, tear, damage, surface cracks, or other defects	Chapter IV.S1, “Structures Monitoring Program”	Generic program
V.4.A-27	Electrical Equipment subject to 10 CFR 50.49 EQ requirements (B)	Monitoring system	Various metallic and polymeric materials	Adverse localized environment caused by heat, radiation, oxygen, moisture, or voltage	Various degradation/ various mechanisms	EQ is a TLAA to be evaluated for the period of extended operation. See Section III.6, “Environmental Qualification of Electrical Equipment,” for acceptable methods for meeting acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.4.A-28	Overpack Neutron Shielding (A)	RS	Boron carbide in various matrices	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of radiation-shielding materials is a TLAA to be evaluated for the period of extended operation. See Section III.5, “Time-Dependent Degradation of Radiation-Shielding Materials,” for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.4.A-29	Cathodic Protection Systems (B)	Cathodic protection of reinforcing steel	Various materials	Embedded in concrete	Reduction of cathodic protection effect on bond strength due to degradation of cathodic protection current	Chapter IV.S1, “Structures Monitoring Program”	Generic program

Table V.4.B NAC International S/T Storage Casks: Multipurpose Canister (MPC)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.4.B-1	MPC: Baseplate, shell, shield lid, port cover, closure ring, bottom plate, and associated welds (A)	CB, CC, HT, SS, FR	Stainless steel	Air – inside the storage overpack, uncontrolled (external), Helium (internal)	Cumulative fatigue damage due to cyclic loading	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, “Fatigue of Metal and Concrete Structures and Components,” for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.4.B-2	MPC: Baseplate, shell, shield lid, port cover, closure ring, bottom plate, and associated welds. (A)	CB, CC, HT, SS, FR	Stainless steel	Air – inside the storage overpack, uncontrolled (external)	Cracking and leakage due to stress corrosion cracking when exposed to moisture and aggressive chemicals in the environment	Chapter IV.M1, “External Surfaces Monitoring of Mechanical Components” Chapter IV.M3, “Welded Canister Seal and Leakage Monitoring Program”	Generic program
V.4.B-3	MPC Internals: Fuel basket, fuel spacer, basket support; heat conduction elements; drain pipe, vent port; neutron absorber panels (A)	CC, CB, HT, SS, FR	Stainless steel, aluminum alloy, borated aluminum or boron carbide/-aluminum alloy plate or BORAL composite	Helium	Degradation of heat transfer, radiation shield, criticality control, confinement boundary, or structural support functions of the MPC internals due to extended exposure to high temperature and radiation.	Chapter IV.M5, “Canister Structural and Functional Integrity Monitoring Program”	Generic program
V.4.B-4	Poured-in gamma radiation shield (NAC I28) (A)	RS	Lead	Long-term exposure to elevated temperatures	Slumping	Further evaluation is required to determine if a site-specific AMP is needed to manage this potential aging effect.	Further evaluation to determine whether a site-specific AMP is needed

Table V.4.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.4.B-5	Shell and lid neutron shielding (NAC I28) (A)	CC	Bisco NS4-FR	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of radiation-shielding materials is a TLAA to be evaluated for the period of extended operation. See Section III.5, "Time-Dependent Degradation of Radiation-Shielding Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.4.B-6	Lid bolting (NAC I28) (A)	CB, SS, RS, HT	Carbon steel, low-alloy steel	Air, leaking rainwater –under the protective cover (external), Helium (internal)	Cumulative fatigue damage due to cyclic loading	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.4.B-7	Lid bolting (NAC I28) (A)	CB, SS, RS, HT	Carbon steel, low-alloy steel	Air, leaking rainwater –under the protective cover (external), Helium (internal)	Loss of material due to corrosion	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components"	Generic program
V.4.B-8	Fuel basket neutron absorber panels (A)	CC	BORAL	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of neutron-absorbing materials is a TLAA to be evaluated for the period of extended operation. See Section III.4, "Time-Dependent Degradation of Neutron-Absorbing Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.4.B-9	Confinement Vessel Inner Components: Fuel Basket, top and bottom fittings, poison plates, basket rails, drain pipe (A)	CC, SS, HT	Stainless Steel, Aluminum, BORAL	Radiation and elevated temperature in helium	Degradation of heat transfer or structural support function due to extended exposure to high temperature and radiation	Chapter IV.M5, "Canister Structural and Functional Integrity Monitoring Program"	Generic program

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V.5 Ventilated Storage Cask System VSC-24

V.5.1 System Description

The Ventilated Storage Cask (VSC) System is a DCSS using a concrete storage cask (i.e., overpack) and a steel, seal-welded basket to store irradiated nuclear fuel. The VSC System can be sized to hold from 4 to 24 PWR assemblies. A VSC-24 system holds 24 PWR assemblies. The VSC-24 System has been designed and analyzed for a lifetime of 50 years. Figure V.5-1 shows the major system components of the VSC-24 System **(EPRI 1021048)**.

The major VSC-24 system components consist of

- Multi-Assembly Sealed Basket (MSB)
- Ventilated Concrete Cask (VCC)
- Concrete Pad

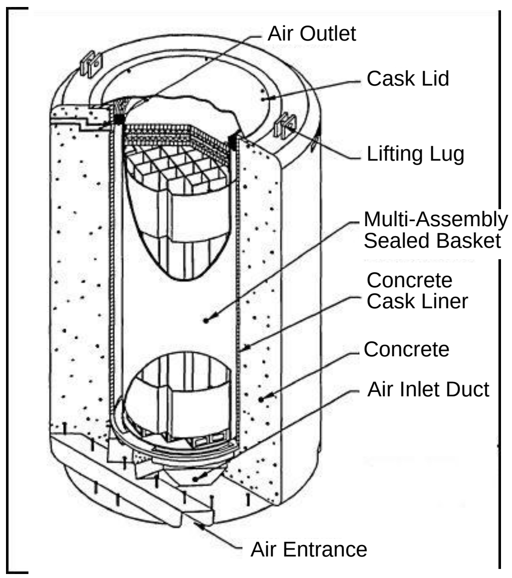

Figure V.5-1: VSC-24 system components.

The MSB is a sealed cylindrical canister containing a basket structure used to support fuel assemblies. The MSB is stored in the central cavity of the VCC. The VCC employs carbon steel-lined air ducts to facilitate natural air circulation, which removes decay heat from the MSB exterior surface. The metal surfaces of VSC-24 cask system components are coated with industry standard coatings, such as Carbo-Zinc, Dimetcote 6, or the equivalent for preventing corrosion of the metal components. The major system components are described below.

V.5.1.1 Multi-Assembly Sealed Basket

The MSB, located in the VCC internal cavity, consists of an outer MSB shell assembly, a shielding lid, a structural lid, and the fuel basket assembly, as shown in Figure V.5-1. The MSB is designed to be free to undergo thermal expansion or contraction relative to the VCC.

MSB Shell: The 25-mm (1-in.)-thick MSB shell is fabricated from SA-516 Gr. 70 pressure vessel steel with a diameter of 1.59 m (62.5 in.). The length depends on fuel type, with or without control elements, and varies from 4.17 to 4.88 m (164.2 to 192.25 in.). The MSB bottom plate is a 0.75-in.-thick plate that is welded to the shell in the fabrication shop. The MSB sits on ceramic tiles that prevent contact with the VCC bottom plate, to prevent galvanic reactions and potential contamination of the VCC bottom plate.

The MSB shell and the internals are coated to prevent detrimental effects from the fuel-pool water chemistry. The exterior of the MSB shell is also coated to prevent corrosion.

The MSB shield lid and structural lid thicknesses are 241 mm and 76 mm (9.5 in. and 3 in.), respectively. Both lids are welded to the MSB shell after fuel loading. The MSB is lifted from above via six hoist rings that are bolted to the MSB structural lid.

Shield Lid: The 241-mm (9.5-in.)-thick MSB shield lid consists of one 64-mm (2.5-in.) steel plate, one 51-mm (2.0-in.) RX-277 neutron shield layer and one 127-mm (5.0-in.) steel plate. The shield lid is placed in the shielding support ring that is welded to the MSB shell by 12.7-mm (0.5-in.) partial-penetration welds. The shield lid is welded to the MSB shell after the fuel is inserted with a 6.4-mm (0.25-in.) partial-penetration weld. Two penetrations for draining, vacuum drying, and backfilling with helium are also located in the shield lid.

A guide tube is screwed into a threaded hole on the backside of the draining penetration. The tube reaches to within 1.59 mm (1/16 in.) of the MSB bottom to facilitate removal of the water from the MSB after fuel loading. After fuel is loaded into the MSB, the MSB is seal welded, dried, backfilled with helium, and structurally welded.

Structural Lid: The MSB structural lid is a 3-in.-thick steel disk that has a penetration for access to the fittings in the shield lid. This penetration is sealed via multiple welds once the helium backfill process has been completed. The structural lid is welded to the MSB shell after the fuel is inserted and provides a redundant seal for confinement.

MSB Internal Fuel Basket: The MSB fuel basket (sleeve basket) is a welded assembly that consists of 24 welded 9.2-in.-square structural tubes, each with a thickness of 0.20 in.

Structural support in the horizontal direction is provided by the curved horizontal support assemblies located at each end and at the center of the basket assembly. The support assembly consists of outer support bar, outer radial support plate, and outer support wall.

All material is SA-516 Gr. 70 or equivalent. A coating is applied to the interior basket to protect it against the fuel-pool chemistry.

V.5.1.2 Ventilated Concrete Cask

The VCC is a cylindrical annulus of reinforced concrete with an outside diameter of 132 in. and an overall height that varies from 5.00 to 5.72 m (196.7 to 225.1 in.). The VCC has 29 in. of concrete in the radial direction and 18 in. of concrete at the bottom for shielding. The concrete of the VCC is Type II Portland Cement.

The internal cavity of the VCC has a diameter of 1.79 m (70.5 in.), with a 44.5-mm (1.75-in.)-thick steel liner. The MSB is stored in the central cavity of the VCC. The VCC provides structural support, shielding, and natural convection cooling for the MSB. The natural convection is provided through the 102-mm (4-in.)-wide annulus between the VCC steel liner and the MSB.

The VCC is provided with a 19.1-mm (0.75-in.)-thick carbon steel cask cover plate that provides shielding and weather cover to protect the MSB from the environment and postulated tornado missiles. The cask cover plate is bolted in place. The bottom of the VCC has a 6.4-mm (0.25-inch)-thick steel plate that covers the entire bottom and prevents any loss of material during a bottom drop accident.

The concrete VCC has chamfered edges in order to mitigate potential damage due to a cask drop. The chamfered edges eliminate the sharp corners at the cask top and bottom, where chipping, spalling, and loss of material predominately occur in a drop accident. The chamfered edges are reinforced to spread the load throughout a larger section of the cask for minimizing concrete material loss during a drop accident.

The VCC is lifted from below via the hydraulic roller skid inserted in the skid access channels.

Inlet and Outlet Air Ducts: The natural-circulation air flow path is formed by air entrance, air inlet ducts at bottom of the VCC, the gap between the MSB and the VCC steel liner, and the air outlet ducts at the top of the cask. Each air duct is 1.22 m (48 in.) wide and is lined on all sides with 12.7-mm (0.5-in.) carbon steel to facilitate natural air circulation. There are four air inlet ducts placed at 90° apart around the cask circumference. A 152-mm (6.0-in.)-thick steel ring is placed at the top of the duct to provide protection from radiation streaming up the ventilation duct. Screens are provided for each duct for preventing air flow blockage due to blowing debris, snow, or animals.

Confinement: The confinement of the VSC consists of multi-pass seal welds at five locations:

1. MSB shell bottom to end plate,
2. MSB shield lid to shell,
3. MSB structural lid to shell,
4. MSB draining, drying and backfilling penetration port to shield lid, and
5. MSB drain and vent cover plates to structural lid.

The MSB welds are helium leak checked to ensure helium leakage of less than 104 atm-cc/sec and repaired, if necessary, in accordance with the facility technical specification. The MSB pressure boundary shield lid, structural lid, and valve cover plate closure welds are PT-tested.

Shielding: The shielding materials used in the VSC-24 cask system include carbon steel and RX-277 neutron shielding material in the MSB and carbon steel and concrete in the VCC.

MSB radial shielding is provided by the 1-in.-thick carbon steel MSB shell, the 44.5-mm (1.75-in.) thick carbon steel VCC liner, and 0.74 m (29 in.) of VCC concrete.

MSB bottom shielding is provided by the 19.1-mm (0.75-in.) thick carbon steel MSB bottom plate, 0.46 m (18 in.) of VCC concrete, and the 51-mm (2 in.) thick carbon steel VCC bottom plate.

MSB top shielding is provided by the 214-mm (9.5-in.) thick carbon steel shield lid (containing 191 mm [7.5 in.] of carbon steel plate and 51 mm (2.0 in.) of RX-277 neutron shielding material), the 76-mm (3.0-in.)-thick carbon steel structural lid, and the 19.1-mm (0.75-in.) thick carbon steel VCC cover lid.

All of the cask lids are carbon steel. The RX-277 in the MSB shield lid is baked to remove unbound moisture present in the material, which prevents off-gassing within the shield lid neutron shield cavity during fuel storage.

The VSC concrete pad consists of three major sections: the truck/trailer loading area, the cask construction area, and the cask storage area. Casks are placed in the vertical position on the pad in linear arrays as defined by the owner utility. Actual array sizes could range from 20 to more than 200 total casks. Plant technical specifications require a 4.6-m (15-ft) center-to-center distance between two casks.

V.5.2 Codes and Service Life

The MSB is designed by analysis to meet material and stress requirements of ASME Code Section III, Division 1. The VCC is designed by analysis to meet load combinations in ACI-349 and the American Nuclear Society ANS-57.9. The VSC cask is designed to withstand the design basis daily and seasonal temperature fluctuations, and tornado, wind, flood, seismic, snow, and ice loads.

The cask is designed to normal, off-normal, and accident loads. The accident loads include full blockage of air inlets, maximum heat load, MSB drop accident, tornado (wind and missiles), flood, and earthquake. The VSC is designed to withstand a maximum horizontal ground acceleration of 0.25 g and a maximum vertical ground acceleration of 0.17 g, in accordance with 10 CFR Part 72, 72.102 (a) requirements appropriate for the majority of sites east of the Rocky Mountains. Site-specific analyses are necessary for sites whose design basis earthquake is larger than 0.25 g.

V.5.3 Current Insepection and Monitoring Program

The following surveillance activities are required in the facility Technical Specification:

1. Visually inspect the inlet and outlet ducts and screens daily to detect and to prevent any airflow blockages.

2. Inspect the VCC exterior annually for any damage or degradation (chipping, spalling, cracks, etc.), for preventing degradation of the concrete interior and avoiding any adverse impact on shielding performance.

3. Inspect the VCC interior surfaces and MSB exterior surfaces every five years for the first VSC unit placed in service at each site, to identify potential air flow blockage and material degradation mechanisms affecting system performance.

The AMPs to manage aging effects for specific structures and components, the materials of construction, and the environment of the VSC-24 spent-fuel storage cask are given in Tables V.5.A and V.5.B. In these tables, the DCSS components listed in the Structure and/or Component column are classified as “A”, “B”, or “C” according to importance to safety, as described in Section I.2.

V.5.4 References

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Safety Evaluation Report, Ventilation Storage Cask (VSC-24) USNRC Docket No. 72-1007 Certificate of Compliance No. 1007, Amendment No. 4, U.S. Nuclear Regulatory Commission, Washington, DC, January 2003.

Table V.5.A VSC-24 Storage Cask: MSB and VCC (Concrete Overpack)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.A-1	MSB: Shell, bottom plate, hoist rings (including bolting), shield lid, shield lid support ring, draining, drying and backfilling penetration ports of the shielding lid, drain and vent cover plates of the structural lid, and all associated welds (A)	CB, RS, SS, HT, FR	Carbon or low-alloy steel	Air – inside the storage overpack, uncontrolled (external), Helium (internal)	Cracking due to stress corrosion cracking; and loss of material due to corrosion, when exposed to moisture and aggressive chemicals in the environment	Chapter IV.M1, "External Surfaces Monitoring of Mechanical Components" Chapter IV.M3, "Welded Canister Seal and Leakage Monitoring Program"	Generic program
V.5.A-2	MSB: Shell, bottom plate, hoist rings (including bolting), shield lid, shield lid support ring, draining, drying and backfilling penetration ports of the shielding lid, drain and vent cover plates of the structural lid, and all associated welds (A)	CB, RS, SS, HT, FR	Carbon or low-alloy steel	Air – inside the storage overpack, uncontrolled (external), Helium (internal)	Cumulative fatigue damage due to cyclic loading	Fatigue is a TLAA to be evaluated for the period of extended operation. See Section III.2, "Fatigue of Metal and Concrete Structures and Components," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.5.A-3	MSB neutron-absorber panels (A)	CC, RS	RX-277	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of neutron-absorbing materials is a TLAA to be evaluated for the period of extended operation. See Section III.4, "Time-Dependent Degradation of Neutron-Absorbing Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA

Table V.5.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.A-4	MSB Internals: Storage sleeves, fuel basket, fuel spacer, basket support assembly, drain pipe, vent port (A)	CC, CB, HT, SS, FR	Carbon or low-alloy steel	Helium	Degradation of heat transfer, radiation shield, criticality control, confinement boundary, or structural support functions of the MSB internals due to extended exposure to high temperature and radiation.	Chapter IV.M5, "Canister Structural and Functional Integrity Monitoring Program"	Generic program
V.5.A-5	VCC: Liner, bottom plate, weather cover cask lid (including bolting), lifting lugs, skid access channels for lifting (A)	SS, FR, FT, RS	Carbon or low-alloy steel	Outside air or marine environment	General corrosion, pitting, crevice corrosion	Chapter IV.S1,"Structures Monitoring Program." Chapter IV.S2, "Protective Coating Monitoring and Maintenance Program."	Generic program
V.5.A-6	VCC air ventilation components: Air inlet and outlet ducts, screens, and gamma shield cross plates (A)	HT, RS	Carbon or low-alloy steel	Air – inside the module, uncontrolled or Air – outdoor	Reduced heat convection capacity due to blockage	Chapter IV.M2, "Ventilation Surveillance Program."	Generic program

Table V.5.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.A-7	VCC Concrete (A)	RS, SS, HT	Concrete and rebars	Air – inside the module, uncontrolled or Air – outdoor	Reduction of strength and modulus of concrete and degradation of shielding performance due to long-term exposure to high temperature and gamma radiation	Site-specific AMP The compressive strength and shielding performance of reinforced concrete is maintained by ensuring that the minimum concrete density is achieved during construction and the allowable concrete temperature and radiation limits are not exceeded. Further evaluations are warranted if these specified limits for temperature and gamma radiation are exceeded during service. Subsection CC-3400 of ASME Code Section III, Division 2, specifies that for normal operation, concrete temperature shall not exceed 66°C (150°F) for a long period and for accident conditions, concrete temperature shall not exceed 93°C (200°F) for short periods. Also, a gamma radiation dose of 10^{10} rads may cause significant reduction of strength. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are incorporated in the design calculations.	Site-specific AMP Further evaluation, if temperature and gamma radiation limits are exceeded
V.5.A-8	VCC Concrete (A)	RS, SS, HT	Concrete and rebars	Air – inside the module, uncontrolled, or Air – outdoor	Cracking due to expansion from reaction with aggregates	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.A-9	VCC Concrete (A)	RS, SS, HT	Concrete and rebars	Air – inside the module, uncontrolled, or Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program"	New Generic

Table V.5.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.10.A-10	VCC Concrete (A)	RS, SS, HT	Concrete and rebars	Air – inside the module, uncontrolled, or Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.A-11	VCC Concrete (A)	RS, SS, HT	Concrete and rebars	Air – inside the module, uncontrolled, or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.A-12	VCC Concrete (A)	RS, SS, HT	Concrete and rebars	Air – inside the module, uncontrolled, or Air – outdoor	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.A-13	Moisture Barriers (caulking, sealants) (if applied) (C)	SS Not ITS	Elastomers, rubber and other similar materials	Air – outdoor	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.A-14	Electrical Equipment subject to 10 CFR 50.49 EQ requirements (if applied) (B)	Monitoring system	Various metallic and polymeric materials	Adverse localized environment caused by heat, radiation, oxygen, moisture, or voltage	Various degradation phenomena/ various mechanisms	EQ is a TLAA to be evaluated for the period of extended operation. See Section III.6, "Environmental Qualification of Electrical Equipment," for acceptable methods for meeting acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA

Table V.5.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.A-15	Cathodic Protection Systems (if applied) (B)	Cathodic protection of rein-forcing steel	Various materials	Embedded in concrete	Reduction of cathodic protection effect on bond strength due to degradation of cathodic protection current	Chapter IV.S1, “Structures Monitoring Program”	Generic program

Table V.5.B VSC-24 Storage Cask: Concrete Pad

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.B-1	Concrete (accessible areas): Above-grade (A or B)	HT, RS, SS	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.B-2	Concrete (inaccessible areas): Below-grade (A or B)	HT, RS, SS	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a site-specific AMP is needed to manage cracking and expansion due to reaction with aggregate of concrete in inaccessible areas. A site-specific AMP is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) for potentially reactive aggregates, aggregate concrete reaction is not significant.	Further evaluation to determine whether a site-specific AMP is needed
V.5.B-3	Concrete (accessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.B-4	Concrete (inaccessible areas): All (A or B)	RS, SS, HT	Reinforced Concrete	Any environment	Loss of strength due to concrete interaction with aluminum	Further evaluation is required to determine if a site-specific AMP is needed to manage loss of strength due to concrete interaction with aluminum in inaccessible areas.	Further evaluation to determine whether a site-specific AMP is needed
V.5.B-5	Concrete (accessible areas): Above-grade (A or B)	HT, RS, SS	Reinforced Concrete	Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.5.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.B-6	Concrete: (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, “Structures Monitoring Program” Inaccessible Concrete Areas: For facilities with non-aggressive groundwater/soil, i.e., pH >5.5, chlorides <500ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilities with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm), and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program
V.5.B-7	Concrete (accessible areas): Above-grade (B)	SS	Reinforced Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter IV.S1, “Structures Monitoring Program”	Generic program

Table V.5.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.B-8	Concrete (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for facilities that are located in moderate to severe weathering areas (weathering index >100 day-in./yr) (NUREG-1557) to determine if a site-specific AMP is needed. A site-specific AMP is not required if documented evidence confirms that the existing concrete had air entrainment content (as per Table CC-2231-2 of the ASME Code Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a site-specific AMP is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental U.S. is shown in ASTM C33-90, Fig. 1.	Further evaluation, for plants located in moderate to severe weathering areas
V.5.B-9	Concrete: Above-grade (A or B)	HT, RS, SS	Reinforced Concrete	Air – inside the module, uncontrolled, or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.5.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.B-10	Concrete (inaccessible areas): Below-grade (A or B)	HT, RS, SS	Reinforced Concrete	Ground-water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program" Inaccessible Concrete Areas: For facilities with non-aggressive groundwater/soil; i.e., pH >5.5, chlorides <500ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilities with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm) , and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program
V.5.B-11	Concrete (accessible areas): Above-grade (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.5.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.B-12	Concrete (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a site-specific AMP is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in inaccessible areas. A site-specific AMP is not required if (1) there is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Further evaluation, if leaching is observed in accessible areas that impact intended function.
V.5.B-13	Concrete: Below-grade (B)	SS	Reinforced Concrete	Soil and water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter IV.S1, "Structures Monitoring Program." If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement
V.5.B-14	Concrete: All (A or B)	HT, RS, SS	Reinforced Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter IV.S1, "Structures Monitoring Program." If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement

Table V.5.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.B-15	Concrete: All (A or B)	HT, RS, SS	Reinforced Concrete	Air – inside the module, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Site-specific AMP The implementation of 10 CFR 72 requirements and ASME Code Section XI, Subsection IWL would not enable identification of the reduction of strength and modulus of elasticity due to elevated temperature. Thus, for any portions of concrete pad that exceed specified temperature limits, further evaluations are warranted. Subsection CC-3400 of ASME Code Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	Site-specific AMP Further evaluation, if temperature limits are exceeded
V.5.B-16	Storm Drainage System (drain pipes and other components) (if applied) (C)	SS Not ITS	PVC and other materials	Air – outdoor	Loss of drainage function due to blockage, wear, damage, erosion, tears, cracks, or other defects	Chapter IV.M1, “External Surfaces Monitoring of Mechanical Components”	Generic program

Table V.5.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.5.B-17	Lightning Protection System (if applied) (C)	SS Not ITS	Various materials	Air – outdoor	Loss of lightning protection due to corrosion, wear, tears, damage, surface cracks, or other defects	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.B-18	Coatings (if applied) (C)	SS Not ITS	Coating	Air – inside the module, uncontrolled, or Air – outdoor	Loss of coating integrity due to blistering, cracking, flaking, peeling, or physical damage	Chapter IV.S2, "Protective Coating Monitoring and Maintenance Program"	Generic program
V.5.B-19	Moisture Barriers (caulking, sealants) (if applied) (C)	SS Not ITS	Elastomers, rubber and other similar materials	Air – outdoor	Loss of sealing due to wear, damage, erosion, tears, surface cracks, or other defects	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.5.B-20	Cathodic Protection Systems (if applied) (B)	Cathodic protection of rein-forcing steel	Various materials	Embedded in concrete	Reduction of cathodic protection effect on bond strength due to degradation of cathodic protection current	Chapter IV.S1, "Structures Monitoring Program"	Generic program

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V.6 Westinghouse MC-10 Metal Dry Storage Cask

V.6.1 System Description

The MC-10 Metal Dry Storage Cask is a totally self-contained vertical bolted metal storage system that provides passive heat removal. A schematic diagram of the MC-10 system is shown in Fig. V.6-1. Each cask stores 24 PWR or 52 BWR fuel assemblies with burn-ups up to 35,000 MWD/MTU and with a heat dissipation up to 15 kW. The casks are provided with pressure monitoring systems for monitoring cask interior helium pressure and leak tightness. The major SSCs with safety functions include the cask body (including the outer shell containing neutron-shielding materials), fuel basket, cask covers and penetrations, instrumentation port, cask seals, and the pad.

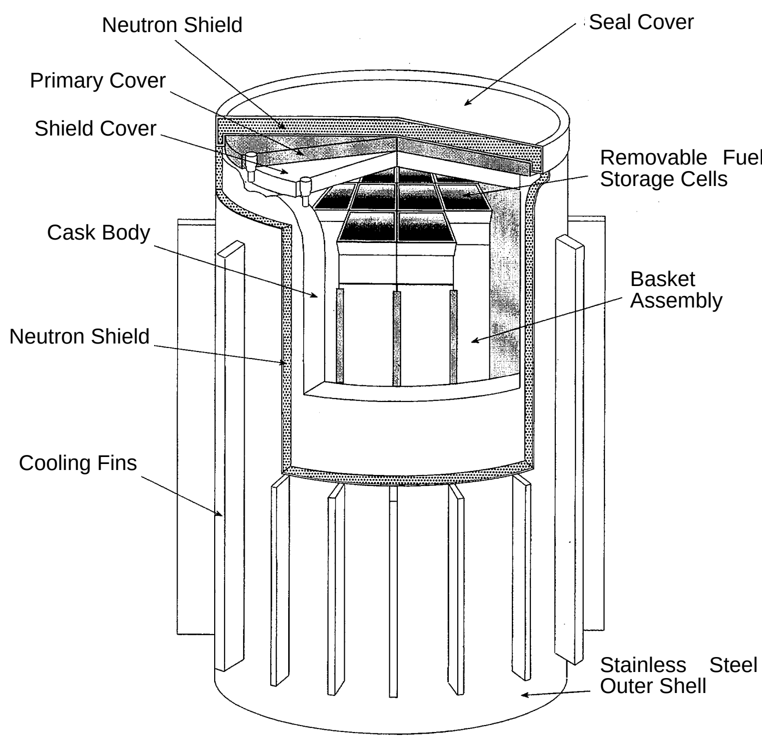

Figure V.6-1: Diagram of MC-10 spent-fuel dry storage cask.

Cask Body (including outer shell): The cask body consists of a forged low-alloy steel container approximately 2.41 m (95 in.) in diameter and 4.95 m (195 in.) in length, and coated internally with thermally sprayed aluminum for corrosion protection. The forged steel cylindrical container with 254-mm (10-in.)-thick wall (for gamma shielding) is welded to a 279-mm (11-in.)-thick low-alloy steel bottom plate. The internal cavity is filled with helium for heat transfer and corrosion protection. The cask body and bottom plate serve as part of the confinement barrier.

Twenty-four (24) 25.4-mm (1-in.)-thick carbon steel cooling fins are welded to the cask body and extend outward to provide a positive conduction path for heat dissipation. The cask body is enclosed by a 6.4-mm (0.25-in.)-thick carbon steel outer shell, which encases a 76.2-mm (3-in.)-thick layer of BISCO NS-3 that provides neutron shielding. The outer shell is coated externally with an epoxy coating for corrosion protection. Four low-alloy steel trunnions are bolted to the cask body for lifting and rotation of the cask.

Fuel Basket Assembly: The fuel basket (Fig. V.6-2) is a one-piece fabricated aluminum grid system that contains 24 removable stainless steel fuel storage cells. Each cell consists of a stainless steel enclosure, borated neutron-absorbing plates (for criticality control), and steel wrappers. Figure V.6-3 shows the MC-10 basket cell details.

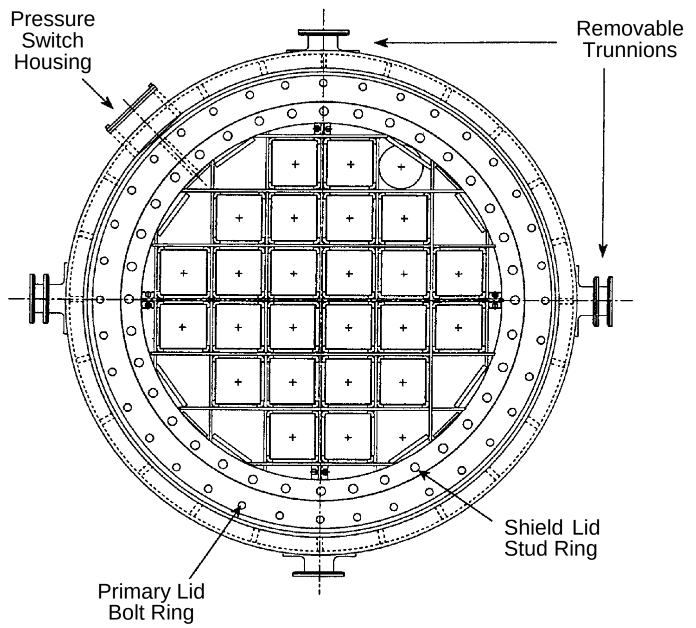

Figure V.6-2: MC-10 fuel basket overview.

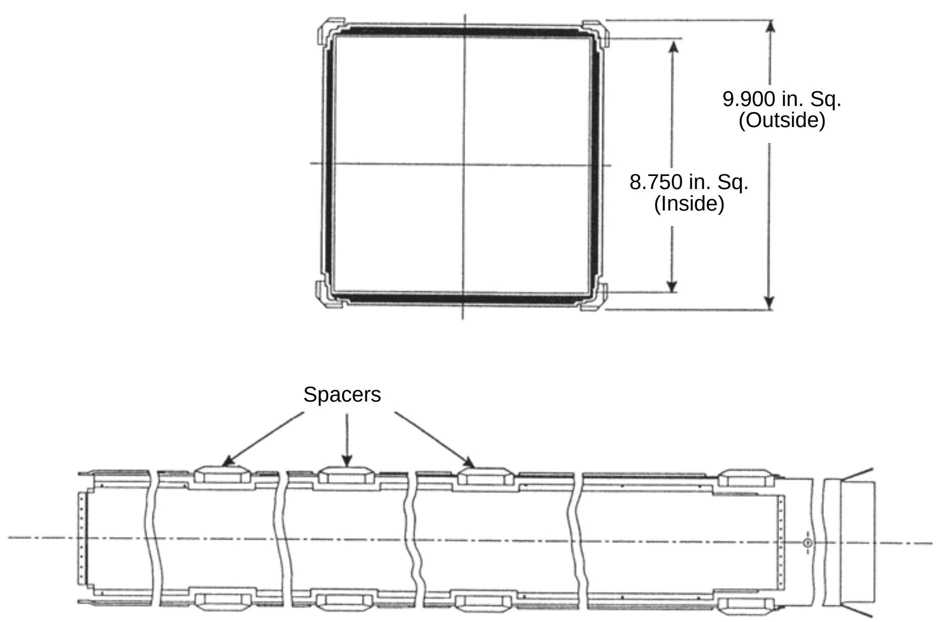

Figure V.6-3: MC-10 fuel cell detail.

Cask Covers and Penetrations: The top end of the cask is sealed by four separated lids (shield lid, primary lid, seal lid, and neutron shield lid) to provide a multiple-barrier redundant-seal system to ensure leak tightness, as shown in Fig. V.6-4. Penetrations are provided for monitoring cask internal helium pressure and for seal leakage testing to monitor the leak-tightness of O-ring assemblies. These four lids and the seal systems are described below.

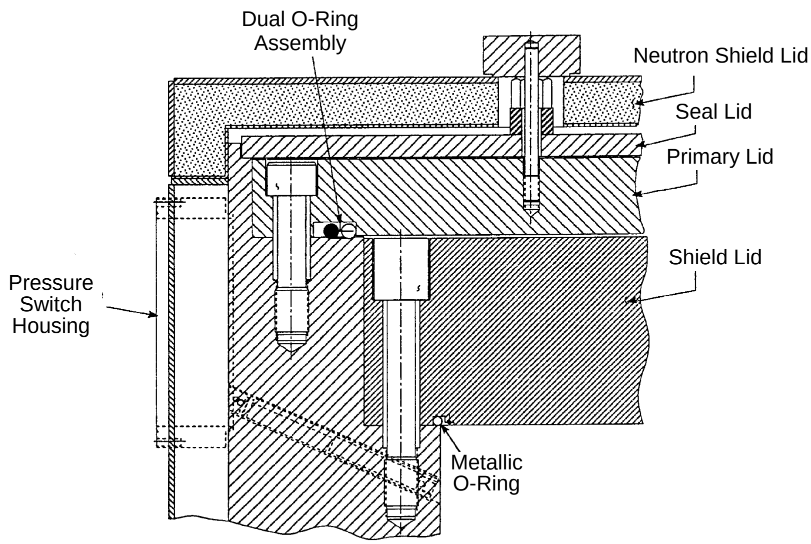

Figure V.6-4: MC-10 cask closure details.

Shield Lid: The shield lid is a 5-in.-thick low-alloy steel plate directly above the fuel basket. The shield lid is bolted to the cask wall by thirty-six (36) 1.5-in.-diameter studs and nuts as shown in Fig. V.6-5. A metallic O-ring provides the seal between studs and cask wall. The shield lid is coated with thermally sprayed aluminum for corrosion protection. The lid includes four tapped holes to interface with lifting equipment.

Two penetrations are provided in the shield lid: a vent penetration for drying and backfilling operations and a siphon penetration and drain on the cask bottom for draining water. Each penetration is provided with a stainless steel cover, which is secured by stainless steel bolts. The covers are sealed with metallic seals. The drain port is also equipped with a shield plug.

Primary Lid: The primary lid is a 3.5-in.-thick carbon steel plate located above the shield lid (Fig. V.6-6). The primary lid is bolted to the cask wall by thirty-six (36) 1.375-in.-diameter low-alloy steel bolts. A metallic and elastomer dual O-ring assembly provides the seal for the bolts. Two penetrations with metallic O-rings are provided for leak testing of the O-ring assembly. Like the shield lid, the primary lid includes four tapped holes to interface with the lifting equipment.

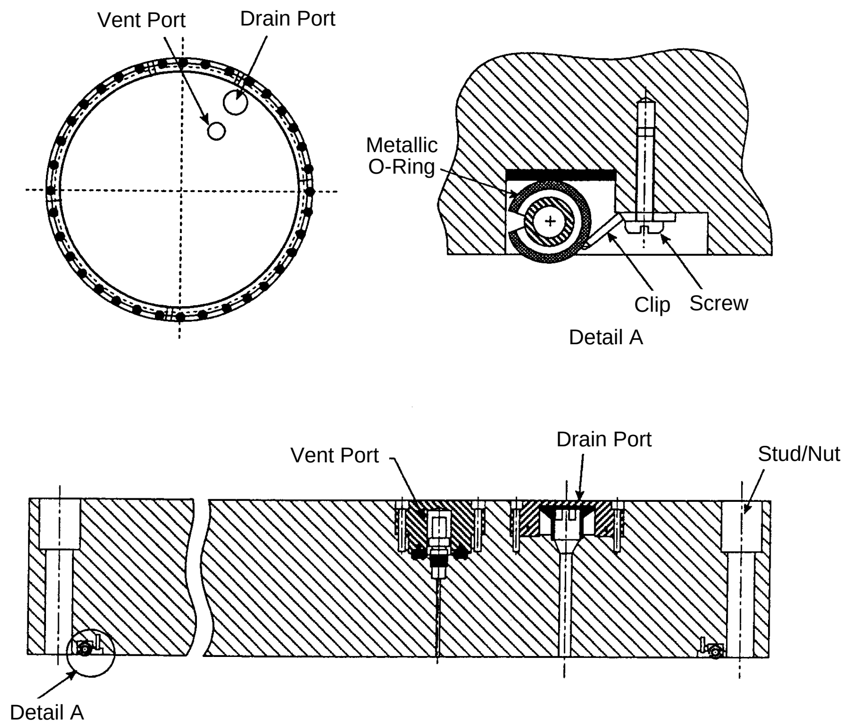

Figure V.6-5: MC-10 shield lid.

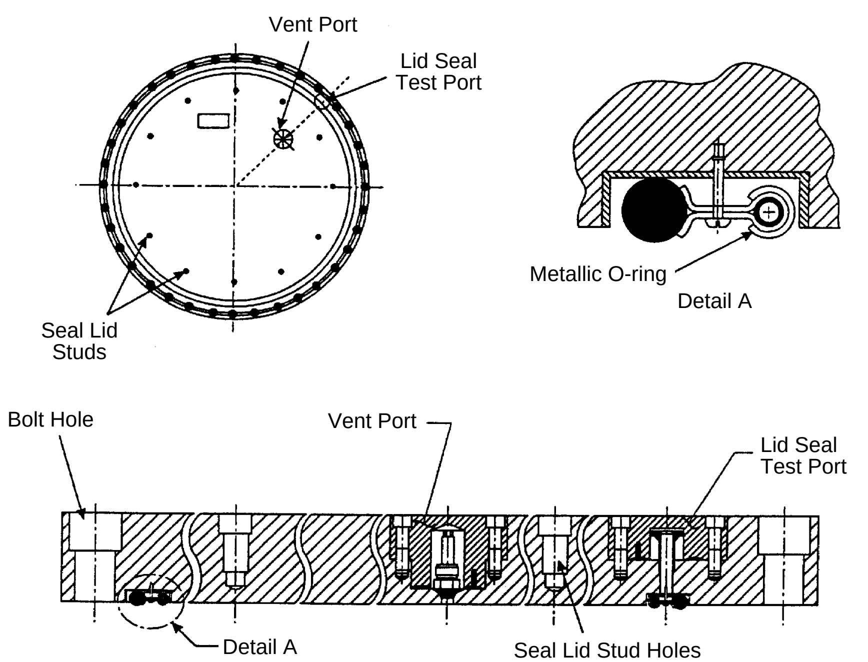

Figure V.6-6: MC-10 cask primary lid and bolting details.

Seal Lid: The seal lid as shown in Fig. V.6-4 is a 1-in.-thick carbon steel plate which is bolted to the primary lid using twelve (12) 1.125 studs. It provides a redundant cover for the cask. No O-ring is provided in the seal lid.

Closure Lid (Neutron Shield Lid): The final protective closure cover, as shown in Fig. V.6-4, is the neutron shield lid, which is a stainless steel closure containing ~5-in.-thick BISCO NS-3 (neutron-absorbing material) for neutron shielding. The closure lid is exposed to the atmosphere and is secured to the seal lid by using the same studs that are used to secure the seal lid to the primary lid. Elastomer O-rings provide a watertight seal for the closure lid.

Instrumentation Port: A pressure sensor port is located on the side of the cask to monitor internal pressure. It consists of an inner and an outer housing. The stainless steel inner housing cover is sealed with metallic seals and stainless steel bolts. The outer housing cover is sealed with elastomer seals and bolts. The outer housing minimizes the introduction of moisture into the housing during cask immersion in the spent-fuel pool. In addition, instead of sealed by elastomer O-rings and studs, the closure lid may be welded over the primary lid to provide seal redundancy.

Cask Seal: For redundancy, at least two metal seals exist at each leak path between the cask cavity and the environment. The metal seals consist of a nickel-based alloy spring with an aluminum jacket and stainless steel sleeve. The metal seals are designed to be leak-tight. Elastomer seals associated with the cask do not perform a function required for license renewal.

The cask is loaded underwater in the spent-fuel pool and the shield cover is placed on the cask. After being lifted out of the spent-fuel pool, the shield lid is bolted in place, and the cask is drained, pressurized with helium, and decontaminated. Next, a pressure-monitoring device is mounted in the primary seal, the primary lid is bolted in place, and the cask is vacuum-dried and repressurized with helium. The seal lid is then bolted to the primary lid, and the neutron shield lid is welded to the cask rim. Following decontamination of the outer surface, the cask is transferred to the ISFSI site and set in place on the concrete pad and normal radiation survey monitoring and daily monitoring of internal helium pressure are performed on the cask.

The design base of the fuel assumes that it has been irradiated to an exposure of 35,000 MWD/MTU and cooled for ten years.

The MC-10 cask contains both gamma- and neutron-shielding materials, which limit surface rate to a maximum 58 mRem per hour. The steel of the cask wall provides gamma shielding, and the BISCO NS-3 neutron-absorbing material, filling the cavities between the cask wall and outer shell, provides neutron shielding.

The MC-10 is designed for passive heat dissipation of up to 15 kW, or 0.625 kW per rod. Decay heat is removed from the cask internals to limit the maximum fuel rod cladding temperature to less than 340°C. Heat is extracted by conduction through the basket grid members and through the grid/cask wall interface. Heat dissipation to ambient atmosphere is through the cooling fins welded to the cask wall.

Pad: The reinforced concrete pad is 230 feet long, 32 feet wide, and approximately 3.0 feet thick. Each pad is designed to accommodate 28 casks. The pad is partially embedded.

V.6.2 Codes and Service Life

Codes and standards representing an acceptable level of design are:

a. American Welding Society (AWS) The Structural Welding Code (AWS Dl.l-1980)
b. American Iron and Steel Institute (AISI) Steel Products Manual
c. American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section II
d. American Society of Testing and Materials (ASTM) Standards

The concrete pads are built in accordance with the BOCA Basic Building Code and applicable American Concrete Institute codes and standards (ACI 318-77, and 1980 Supplement and Commentary and ACI 315-74) with design compressive strength of 3000 psi after 28 days.

V.6.3 Current Insepection and Monitoring Program

The current inspection program for the MC-10 system in the original or extended license period (20- or 40-year extension) involves monitoring of the cask internal helium pressure on a daily basis, in addition to the normal radiation monitoring. The pressure-monitoring device is mounted to the wall of the cask to provide a direct means of detecting a loss of cask integrity or fuel rod integrity with either a low-pressure or a high-pressure alarm. The pressure transducers are calibrated biannually. A cover seal test port is provided to test the leakage of O-ring sealing systems.

The aging management of MC-10 casks in the Surry ISFSI relies on the Dry Storage Cask Inspection Activities Program during the period of extended operation. The scope of program involves 1) the continuous pressure monitoring of the in-service dry storage casks, 2) the quarterly visual inspection of all dry storage casks that are in service at the Surry ISFSI, 3) a visual inspection of the MC-10 dry storage cask seal cover area which is to be performed prior to the end of the original operating license period, and 4) the visual inspection of the normally inaccessible areas of casks in the event they are lifted in preparation for movement or an environmental cover is removed for maintenance.

The AMPs to manage aging effects for specific structures and components, the materials of construction, and environment of the MC-10 spent-fuel storage cask are given in Tables V.6.A and V.6.B. In these tables, the DCSS components listed in the Structure and/or Component column are classified as “A”, “B”, or “C” according to importance to safety, as described in Section I.2.

V.6.4 References

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Table V.6.A Westinghouse MC-10 Metal Dry Storage Cask: Cask Body, Fuel Basket, Cask Lids, and Seals

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.A-1	Closure lid (neutron shield lid) (A)	RS, CB, SS, HT	Carbon Steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M1, “External Surface Monitoring of Metal Components” Chapter IV.S2, “Protective Coating Monitoring and Maintenance Program”	Generic program
V.6.A-2	Closure lid (neutron shield lid) (A)	RS	BISCO NS-3	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of radiation-shielding materials is a TLAA to be evaluated for the period of extended operation. See Section III.5, “Time-Dependent Degradation of Radiation-Shielding Materials,” for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.6.A-3	Cask outer shell and coating (A)	SS, HT	Steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M1, “External Surface Monitoring of Metal Components” Chapter IV.S2, “Protective Coating Monitoring and Maintenance Program”	Generic program
V.6.A-4	Cask outer shell neutron shield (B)	RS	BISCO NS-3	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of radiation-shielding materials is a TLAA to be evaluated for the period of extended operation. See Section III.5, “Time-Dependent Degradation of Radiation-Shielding Materials,” for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.6.A-5	Cask cooling fins, coating, and associated welds (A)	HT	Carbon Steel	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M1, “External Surface Monitoring of Metal Components” Chapter IV.S2, “Protective Coating Monitoring and Maintenance Program”	Generic program

Table V.6.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.A-6	Fuel Basket Assembly: grid, fuel storage cells, borated neutron absorbing plate, wrappers, spacers (A)	CC, CB, HT, SS, FR	Carbon or low-alloy steel, Stainless Steel	Helium	Degradation of heat transfer, radiation shield, criticality control, confinement boundary, or structural support functions of the MSB internals due to extended exposure to high temperature and radiation.	Chapter IV.M5, "Canister Structural and Functional Integrity Monitoring Program"	Generic program
V.6.A-7	Fuel Basket Assembly: neutron absorbing plate (A)	CC	Borated aluminum composite	Radiation and elevated temperature	Degradation of shielding properties due to long-term exposure to high temperature and gamma and neutron radiation	Degradation of neutron-absorbing materials is a TLAA to be evaluated for the period of extended operation. See Section III.4, "Time-Dependent Degradation of Neutron-Absorbing Materials," for acceptable methods for meeting the acceptance criteria in Section 3.5.1 of NUREG-1927.	TLAA
V.6.A-8	Shield lid, primary lid, seal lid, closure lid (neutron shield lid), penetration steel covers, O-ring assemblies, and associated bolts and welds (A)	RS, CB, SS, HT	Carbon or low-alloy steel	Helium and Air – outdoor	Loss of material due to corrosion, loss of sealing forces due to stress relaxation and creep of the metallic O-rings, corrosion and loss of preload of the closure bolts, and stress corrosion cracking of penetration welds.	Chapter IV.M4, "Bolted Canister Seal and Leakage Monitoring Program"	Generic program

Table V.6.A (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.A-9	Pressure Monitoring System: Pressure sensor inner and outer housing and associated elastomer seals and bolts. (C)	Monitoring system	Steel, elastomers, rubber and similar materials	Air – outdoor	Loss of material due to general, pitting, and crevice corrosion	Chapter IV.M4, "Bolted Canister Seal and Leakage Monitoring Program" Chapter IV.M1, "External Surface Monitoring of Metal Components"	Generic programs
V.6.A-10	Moisture Barriers (caulking, sealants) (if applied) (C)	SS Not ITS	Elastomers, rubber and other similar materials	Air – outdoor	Loss of sealing due to wear, damage, erosion, tear, surface cracks, or other defects	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.6.A-11	Cathodic Protection Systems (if applied) (B)	Cathodic protection of reinforcing steel	Various materials	Embedded in concrete	Reduction of cathodic protection effect on bond strength due to degradation of cathodic protection current	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.6.B Westinghouse MC-10 Metal Dry Storage Cask: Concrete Pad

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.B-1	Concrete (accessible areas): Above-grade (A or B)	HT, RS, SS	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.6.B-2	Concrete (inaccessible areas): Below-grade (A or B)	HT, RS, SS	Reinforced Concrete	Any environment	Cracking due to expansion from reaction with aggregates	Further evaluation is required to determine if a site-specific AMP is needed to manage cracking and expansion due to reaction with aggregate of concrete in inaccessible areas. A site-specific AMP is not required if (1) as described in NUREG-1557, investigations, tests, and petrographic examinations of aggregates performed in accordance with ASTM C295 and other ASTM reactivity tests, as required, can demonstrate that those aggregates do not adversely react within concrete, or (2) for potentially reactive aggregates, aggregate concrete reaction is not significant.	Further evaluation to determine whether a site-specific AMP is needed
V.6.B-3	Concrete (accessible areas): Above-grade (A or B)	HT, RS, SS	Reinforced Concrete	Air – outdoor	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.6.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.B-4	Concrete: (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Cracking; loss of bond; and loss of material (spalling, scaling) due to corrosion of embedded steel	Chapter IV.S1, “Structures Monitoring Program” Inaccessible Concrete Areas: For facilities with non-aggressive groundwater/soil; i.e., pH >5.5, chlorides <500ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilities with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm) , and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program
V.6.B-5	Concrete (accessible areas): Above-grade (B)	SS	Reinforced Concrete	Air – outdoor	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Chapter IV.S1, “Structures Monitoring Program”	Generic program

Table V.6.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.B-6	Concrete (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Ground-water/soil	Loss of material (spalling, scaling) and cracking due to freeze-thaw	Further evaluation is required for facilities that are located in moderate to severe weathering areas (weathering index >100 day-inch/yr). Refer to NUREG-1557 to determine if a site-specific AMP is needed. A site-specific AMP is not required if documented evidence confirms that the existing concrete had air-entrainment content (as per Table CC-2231-2 of the ASME Code Section III Division 2), and subsequent inspections of accessible areas did not exhibit degradation related to freeze-thaw. Such inspections should be considered a part of the evaluation. If this condition is not satisfied, then a site-specific AMP is required to manage loss of material (spalling, scaling) and cracking due to freeze-thaw of concrete in inaccessible areas. The weathering index for the continental U.S. is shown in ASTM C33-90, Fig. 1.	Further evaluation, for plants located in moderate to severe weathering areas
V.6.B-7	Concrete: Above-grade (A or B)	HT, RS, SS	Reinforced Concrete	Air – inside the module, uncontrolled, or Air – outdoor	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.6.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.B.8	Concrete (inaccessible areas): Below-grade (A or B)	HT, RS, SS	Reinforced Concrete	Ground-water/soil	Increase in porosity and permeability; cracking; loss of material (spalling, scaling) due to aggressive chemical attack	Chapter IV.S1, "Structures Monitoring Program" Inaccessible Concrete Areas: For facilities with non-aggressive groundwater/soil; i.e., pH >5.5, chlorides <500ppm, or sulfates <1500 ppm, as a minimum, consider (1) examination of the exposed portions of the below-grade concrete, when excavated for any reason, and (2) periodic monitoring of below-grade water chemistry, including consideration of potential seasonal variations. For facilties with aggressive groundwater/soil (i.e., pH <5.5, chlorides >500 ppm, or sulfates >1500 ppm) , and/or where the concrete structural elements have experienced degradation, a site-specific AMP accounting for the extent of the degradation experienced should be implemented to manage the concrete aging during the period of extended operation.	Generic program
V.6.B-9	Concrete (accessible areas): Above-grade (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Chapter IV.S1, "Structures Monitoring Program"	Generic program

Table V.6.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.B-10	Concrete (inaccessible areas): Below-grade (B)	SS	Reinforced Concrete	Water – flowing	Increase in porosity and permeability; loss of strength due to leaching of calcium hydroxide and carbonation	Further evaluation is required to determine if a site-specific AMP is needed to manage increase in porosity, and permeability due to leaching of calcium hydroxide and carbonation of concrete in inaccessible areas. A site-specific AMP is not required if (1) there is evidence in the accessible areas that the flowing water has not caused leaching and carbonation, or (2) evaluation determined that the observed leaching of calcium hydroxide and carbonation in accessible areas has no impact on the intended function of the concrete structure.	Further evaluation, if leaching is observed in accessible areas that impact intended function
V.6.B-11	Concrete: Below-grade (B)	SS	Reinforced Concrete	Soil and water – flowing under foundation	Reduction of foundation strength and cracking due to differential settlement and erosion of porous concrete subfoundation	Chapter IV.S1, "Structures Monitoring Program" If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement
V.6.B-12	Concrete: All (A or B)	HT, RS, SS	Reinforced Concrete	Soil	Cracking and distortion due to increased stress levels from settlement	Chapter IV.S1, "Structures Monitoring Program" If a de-watering system is relied upon for control of settlement, then the licensee is to ensure proper functioning of the de-watering system through the period of extended operation.	Further evaluation, if a de-watering system is relied upon for control of settlement

Table V.6.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.B-13	Concrete: All (A or B)	HT, RS, SS	Reinforced Concrete	Air – inside the module, uncontrolled	Reduction of strength and modulus due to elevated temperature (>150°F general; >200°F local)	Site-specific AMP The implementation of 10 CFR 72 requirements and ASME Code Section XI, Subsection IWL would not enable identification of the reduction of strength and modulus of elasticity due to elevated temperature. Thus, for any portions of concrete pad that exceed specified temperature limits, further evaluations are warranted. Subsection CC-3400 of ASME Code Section III, Division 2, specifies the concrete temperature limits for normal operation or any other long-term period. The temperatures shall not exceed 150°F except for local areas, such as around penetrations, which are not allowed to exceed 200°F. If significant equipment loads are supported by concrete at temperatures exceeding 150°F, an evaluation of the ability to withstand the postulated design loads is to be made. Higher temperatures than given above may be allowed in the concrete if tests and/or calculations are provided to evaluate the reduction in strength and modulus of elasticity and these reductions are applied to the design calculations.	Site-specific AMP Further evaluation, if temperature limits are exceeded
V.6.B-14	Storm Drainage System (drain pipes and other components) (if applied) (C)	SS Not ITS	PVC and other materials	Air – outdoor	Loss of drainage function due to blockage, wear, damage, erosion, tears, cracks, or other defects	Chapter IV.M1, “External Surfaces Monitoring of Mechanical Components”	Generic program
V.6.B-15	Lightning Protection System (if applied) (C)	SS Not ITS	Various materials	Air – outdoor	Loss of lightning protection due to corrosion, wear, tears, damage, surface cracks, or other defects	Chapter IV.S1, “Structures Monitoring Program”	Generic program

Table V.6.B (Cont.)

Item	Structure and/or Component	Intended Function	Material	Environment	Aging Effect/ Mechanism	Aging Management Program (AMP)	Program Type
V.6.B-16	Coatings (if applied) (C)	SS Not ITS	Coating	Air – inside the module, uncontrolled; or Air – outdoor	Loss of coating integrity due to blistering, cracking, flaking, peeling, or physical damage	Chapter IV.S2, "Protective Coating Monitoring and Maintenance Program"	Generic program
V.6.B-17	Moisture Barriers (caulking, sealants) (if applied) (C)	SS Not ITS	Elastomers, rubber and other similar materials	Air – outdoor	Loss of sealing due to wear, damage, erosion, tears, surface cracks, or other defects	Chapter IV.S1, "Structures Monitoring Program"	Generic program
V.6.B-18	Cathodic Protection Systems (if applied) (B)	Cathod-ic pro-tection of rein-forcing steel	Various materials	Embedded in concrete	Reduction of cathodic protection effect on bond strength due to degradation of cathodic protection current	Chapter IV.S1, "Structures Monitoring Program"	Generic program

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APPENDIX A: QUALITY ASSURANCE FOR AGING MANAGEMENT PROGRAMS FOR USED-FUEL DRY STORAGE SYSTEMS

Application for license renewal for an independent spent-fuel storage installation (ISFSI) or dry cask storage system (DCSS) must demonstrate that the effects of aging on structures, systems, and components (SSCs) subject to aging management review (AMR) will be managed in a manner that is consistent with the current licensing basis of the facility for the proposed period of extended operation. Therefore, those aspects of the AMR process that affect the quality of safety-related structures and components are subject to the quality assurance (QA) requirements of 10 CFR Part 72, Subpart G, "Quality Assurance." The aging management program (AMP) elements that are related to QA are Elements 7 (corrective actions), 8 (confirmation process), and 9 (administrative controls). For non-safety-related structures and components subject to an AMR, the existing 10 CFR Part 50, Appendix B, QA program may be used to address the elements of corrective actions, confirmation processes, and administrative controls, provided it meets the recordkeeping requirements of 10 CFR Part 72.174, on the following bases:

- Criterion XVI of 10 CFR Part 50, Appendix B, requires that measures be established to ensure that conditions adverse to quality, such as failures, malfunctions, deviations, defective materials and equipment, and non-conformances, are promptly identified and corrected. In the case of significant conditions adverse to quality, measures must be implemented to ensure that the cause of the condition is determined and that corrective action is taken to preclude repetition. In addition, the cause of the significant condition adverse to quality and the corrective action implemented must be documented and reported to appropriate levels of management.

  The license renewal applicant should ensure that corrective actions include root-cause determinations for SSCs that are important to safety, and that the actions to be taken can prevent recurrence in a timely manner. The operating history, including corrective actions and design modifications, is an important source of information for evaluating the ongoing condition of in-scope SSCs. The applicant should provide detailed discussions of such history. The applicant may consider both site-specific and industry-wide experience, as relevant, as part of the overall condition assessment of in-scope SSCs.

- 10 CFR 72.172 requires that the licensee, applicant for a license, certificate holder, and applicant for a Certificate of Compliance shall establish measures to ensure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and non-conformances, are promptly identified and corrected. In the case of a significant condition identified as adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action is taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action taken must be documented and reported to appropriate levels of management.

  To preclude repetition of significant conditions adverse to quality, the confirmation process element (Element 8) for license-renewal AMPs consists of follow-up actions to verify that the corrective actions implemented are effective in preventing a recurrence. The corrective actions described by the applicant should include root cause determinations for SSCs that are important to safety, and the actions to be taken by the applicant must be sufficient to provide reasonable assurance that recurrence will not occur. The effectiveness of prevention and mitigation programs should be verified periodically, for example, through the use of condition-monitoring activities to verify the effectiveness of the mitigation programs. One-time events should be evaluated for possible mitigating measures during the renewal period.

Administrative controls provide a formal review and approval process, and any AMPs to be relied on for license renewal should have regulatory and administrative controls. Administrative action that must be taken in the event of noncompliance with a limit or condition should be specified.

- 10 CFR Part 72.24(h) requires that the safety analysis report submitted by a DCSS or ISFSI license applicant include a plan for the conduct of operations, including the planned managerial and administrative controls system used by the applicant's organization, and the program for training of personnel pursuant to subpart I, “Training and Certification of Personnel.” Pursuant to 10 CFR 72.44(c)(5), administrative controls include the organization and management procedures, recordkeeping, review and audit, and reporting requirements necessary to ensure that the operations involved in the storage of spent fuel and reactor-related greater-than-Class-C waste in an ISFSI are performed in a safe manner.

Notwithstanding the suitability of its provisions to address quality-related aspects of the AMR process for license renewal, 10 CFR Part 72, Subpart G, covers only safety-related SSCs. Therefore, absent a commitment by the applicant to expand the scope of its 10 CFR Part 72, Subpart G, QA program to include non-safety-related SSCs subject to an AMR for license renewal, the AMPs applicable to non-safety-related SSCs include alternative means to address corrective actions, confirmation processes, and administrative controls. Such alternative means are subject to review on a case-by-case basis.

References

10 CFR Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants, Office of the Federal Register, National Archives and Records Administration, 2009.

10 CFR 72.24, Contents of Application: Technical Information, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR 72.44, License Conditions, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR 72.174, Quality Assurance Records, Office of the Federal Register, National Archives and Records Administration, 2011.

10 CFR Part 72, Subpart G, Quality Assurance, Office of the Federal Register, National Archives and Records Administration, as amended June 9, 2011.

Appendix E
FCT Document Cover Sheet

Name/Title of Deliverable/Milestone	***Managing Aging Effects on Dry Cask Storage Systems for Extended Long-Term Storage and Transportation of Used Fuel Rev. 0/M2FT-12AN0803011/6/30/2012***
Work Package Title and Number	ST R&D Investigations - ANL
Work Package WBS Number	FT-12AN080301
Responsible Work Package Manager	Yung Liu (Name/Signature)

Date Submitted 6/30/2012

Quality Rigor Level for Deliverable/Milestone	☐ QRL-3	☒ QRL-2	☐ QRL-1 ☐ Nuclear Data	☐ N/A*

This deliverable was prepared in accordance with Argonne National Laboratory

(Participant/National Laboratory Name)

QA program which meets the requirements of

☒ DOE Order 414.1 ☐ NQA-1-2000

This Deliverable was subjected to:

☒ Technical Review	☒ Peer Review
Technical Review (TR)	**Peer Review (PR)**
Review Documentation Provided	**Review Documentation Provided**
☐ Signed TR Report or,	☐ Signed PR Report or,
☐ Signed TR Concurrence Sheet or,	☐ Signed PR Concurrence Sheet or,
☒ Signature of TR Reviewer(s) below	☒ Signature of PR Reviewer(s) below

Name and Signature of Reviewers

Yung Y. Liu	Sandra M. Birk (accepted via email)
Yung Liu 2012.06.29 14:26:11 -05'00'	

*Note: In some cases there may be a milestone where an item is being fabricated, maintenance is being performed on a facility, or a document is being issued through a formal document control process where it specifically calls out a formal review of the document. In these cases, documentation (e.g., inspection report, maintenance request, work planning package documentation or the documented review of the issued document through the document control process) of the completion of the activity along with the Document Cover Sheet is sufficient to demonstrate achieving the milestone. QRL for such milestones may be also be marked N/A in the work package provided the work package clearly specifies the requirement to use the Document Cover Sheet and provide supporting documentation.