Rationale for the Performance Specification for Standardized Transportation, Aging, and Disposal Canister Systems

Fuel Cycle Research & Development

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July 20, 2015: Rev. 1. Corrects minor errors. Clarifies the versions of ASME-NQA-1. Permits but does not require design of the first generation of STAD canisters to accept damaged fuel. Clarifies the controlling repository design cases for the requirement to design canister internals to meet potential disposal thermal considerations. Revisions are indicated by bars in the margins.

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EXECUTIVE SUMMARY

This document provides the rationale for the specifications for the standardized transportation, aging, and disposal (STAD) canister systems provided in *Performance Specification for Standardized Transportation, Aging, and Disposal Canister Systems, FCRD-NFST-2014-000579*, Rev. 2, referred to as *STAD Spec* herein. The *STAD Spec* was developed to support STAD canister system studies and potential research, development, and demonstration activities. Requirements in the specification may evolve with time based on results from analyses, experiments, design studies, system evaluations and demonstrations, as well as other factors.

The STAD canister system consists of a canister, together with a storage or aging overpack/module/vault, a shielded transfer cask (STC), a site transporter, a transportation overpack, and a transportation skid. Three sizes of circular cross section STAD canister variants are specified. The small STAD canister capacity is four pressurized water reactor (PWR) or nine boiling water reactor (BWR) spent nuclear fuel (SNF) assemblies. The medium STAD canister capacity is 12 PWR or 32 BWR SNF assemblies. The large STAD canister capacity is 21 PWR or 44 BWR SNF assemblies.

There are 60 specifications in the *STAD Spec* that cover the STAD canister general design attributes, canister safety functional requirements, operational considerations, materials, storage and aging system requirements, and transportation system requirements. The 60 specifications are derived from regulatory requirements, regulatory guidance, STAD canister disposability considerations, and interim storage facility (ISF) and geologic repository handling facility considerations.

The specifications and their respective rationales are guided by the following two questions:

- If a specification were included, would it influence the design and analyses to meet a desired objective?
- If a specification were omitted, could its omission influence the design and analyses in a way that would compromise a desired objective?

The rationales in this document provide the basis for the specifications included in the *STAD Spec,* the specifications that were considered and omitted from the *STAD Spec* for reasons that may not be obvious, and potential refinements to specifications that may be considered in a future evolution of the *STAD Spec*.

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FIGURES

TABLES

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ACRONYMS

AAR	Association of American Railroads
AASHTO	American Association of State Highway and Transportation Officials
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
AO	aging overpack
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
BWR	boiling water reactor
CISF	centralized interim storage facility
CFR	Code of Federal Regulations
CISCC	chloride-induced stress corrosion cracking
CNWRA	Center for Nuclear Waste Regulatory Analyses
CoC	NRC Certificate of Compliance
DCRA	disposal control rod assembly
DE	design earthquake
DOE	US Department of Energy
DPC	dual-purpose canister
EBS	engineered barrier system
EF	early failure
FCRD	Fuel Cycle Research and Development
GI	generic issue (NRC)
GTCC	greater than class C
HAC	hypothetical accident conditions
HLW	high-level radioactive waste
IE	inspection and enforcement (NRC)
IEEE	Institute of Electrical and Electronics Engineers
ISF	interim storage facility (for the purposes of this document, an ISF can be a DOE-owned MRS or a privately owned ISFSI not co-located with a nuclear power plant)
ISFSI	independent spent fuel storage installation
ISG	interim staff guidance
ISO	International Organization for Standardization
ITS	important to safety
LPR	linear polar resistance
MPC	multipurpose canister
MRS	monitored retrievable storage installation (DOE-owned, this term is used in the Nuclear Waste Policy Act and in 10 CFR Part 72)
NAC	NAC International, a company that provides spent fuel management technologies
NCT	normal conditions of transport
NDE	nondestructive examination
NFST	Nuclear Fuels Storage and Transportation
NPP	nuclear power plant
NQA	Nuclear Quality Assurance
NRC	US Nuclear Regulatory Commission

PGA	peak ground acceleration
PWR	pressurized water reactor
RCRA	Resource Conservation and Recovery Act
RG	regulatory guide (NRC)
RIS	regulatory issue summary (NRC)
SAR	safety analysis report
SCC	stress corrosion cracking
SFST	Spent Fuel Storage and Transportation
SKB	Swedish Nuclear Fuel and Waste Management Company
SNF	spent nuclear fuel
SNL	Sandia National Laboratories
SS	stainless steel
SSC	structures, systems, and components
STAD	standardized transportation, aging, and disposal
STC	shielded transfer cask
TAD	transportation, aging, and disposal
TDS	total dissolved solids
THCMBR	thermal-hydrologic-chemical-mechanical-biological-radiological (coupled processes)
TN	Transnuclear Inc., now a division of AREVA Inc. (AREVA TN)
TPBAR	tritium-producing burnable absorber rods
TSAR	topical safety analysis report
UFD	Used Fuel Disposition
UNS	unified numbering system (for metal alloys)
WP	waste package

ABBREVIATIONS AND MEASUREMENT UNITS

°C	degrees Centigrade
°F	degrees Fahrenheit
cm	centimeter
cm^2	square centimeter
dpm	disintegrations per minute
ft	foot, also abbreviated as an apostrophe, ’
GWd	gigawatt-day
h or hr	hour
in.	inch, also abbreviated as a quotation mark, ”
kg	kilogram
L	liter
lb	pound(s) (weight; unless otherwise specified)
m	meter
μm	micrometer
m^2	square meter(s)
MTU	metric tons of uranium
nm	nanometer
ppm	part per million
rev.	revision
s or sec	second
t or T	ton (2000 lb)
W	watt
wt%	weight percent
y	year

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RATIONALE FOR THE PERFORMANCE SPECIFICATION FOR STANDARDIZED TRANSPORTATION, AGING, AND DISPOSAL CANISTER SYSTEMS

1. INTRODUCTION

1.1 Purpose

This document provides the rationale for the specifications for the standardized transportation, aging, and disposal (STAD) canister systems provided in *Performance Specification for Standardized Transportation, Aging, and Disposal Canister Systems, FCRD-NFST-2014-000579*, Rev. 2, referred to as *STAD Spec* herein. The *STAD Spec* was developed to support STAD canister system studies and potential research, development, and demonstration activities. Requirements in this specification may evolve with time based on results from analyses, experiments, design studies, system evaluations and demonstrations, as well as other factors.

The rationale in this document provides the basis for the following:

- the specifications included in the *STAD Spec*,
- specifications that were considered and omitted from the *STAD Spec* for reasons that may not be obvious, and
- potential refinements to specifications that may be considered in a future evolution of the *STAD Spec*.

1.2 Background

The concept of a canister system capable of storage, transportation, and disposal without repackaging has been considered for many years. Past standardization efforts include the transportation, aging, and disposal (TAD) canister system (DOE 2008b) and the multipurpose canister (MPC). The most recent iteration of standardization is the STAD canister system, which includes three size variants derived from potential disposal geologies (Hardin and Kalinina 2015). The STAD canister system consists of a canister, together with a storage or aging overpack/module/vault, a shielded transfer cask (STC), a site transporter, a transportation overpack, and a transportation skid. There are 60 specifications in the *STAD Spec* that cover the following aspects of the STAD canister systems:

- STAD canister general design attributes and limitations, such as lifetime, shape, mass, capacity SNF payload characteristics, and handling orientation,
- STAD canister safety functional requirements including structural, thermal, radiation protection criticality safety, and confinement/containment,
- STAD canister operational considerations,
- STAD canister shell and internal component materials, welding, and stress relieving requirements,
- storage and aging system requirements, and
- transportation system requirements.

1.3 Approach

The 60 specifications in the *STAD Spec* are based on the following five requirement drivers:

1. regulatory requirements for storage as documented in 10 Code of Federal Regulations (CFR) Part 72 and for transportation as documented in 10 CFR Part 71, as well as other CFRs (e.g., 10 CFR Part 20 for radiological protection, 10 CFR Part 73 for physical protection) invoked by these regulations,
2. guidance in regulatory guides (RGs), standard review plans (SRPs), and interim staff guidance documents (ISGs) for storage and transportation,
3. consideration of CFRs, RGs, SRPs, and ISGs for a previously submitted geologic repository license application (DOE 2008a),
4. STAD canister disposability considerations in various geologic media, and
5. consideration of potential concepts for interim storage facility (ISF) and geologic repository handling facility design, operations, and licensing.

The specifications provided in the *STAD Spec* and their respective rationales are guided by the following two questions:

- If a specification were included, would it influence the design and analyses to meet a desired objective?
- If a specification were omitted, could its omission influence the design and analyses in a way that would compromise a desired objective?

To develop a complete, transparent, traceable rationale for each specification, the rationales are grouped based on common categories. For each specification or group of related specifications, the following information is provided in the rationale as appropriate:

- the technical or regulatory basis for the specification,
- the reason for the specification (i.e., the desired objective from the specification that otherwise may not be met), and
- alternative forms of the specification that may have been considered or may be considered in the future, and the reason(s) they were not included in the current revision of the *STAD Spec*.

Sect. 2 of this document provides the rationales for the 60 specifications as follows:

- Sect. 2.1 provides the rationales for a grouped set of 13 specifications aimed at ensuring compliance with storage and transportation regulatory requirements and conformance with associated NRC guidance.
- Sect. 2.2 provides the rationales for 15 STAD canister general design attributes.
- Sect. 2.3 provides the rationales for 10 STAD canister safety functional requirements.
- Sect. 2.4 provides the rationales for 4 STAD canister operational considerations.
- Sect. 2.5 provides the rationales for 7 STAD canister shell and internal component materials.
- Sect. 2.6 provides the rationales for 5 storage and aging system requirements.
- Sect. 2.7 provides the rationales for 6 transportation system requirements.

1.4 Quality Assurance

This document was prepared under the ORNL QA Program, which is based on ASME-NQA-1-2008 with the NQA-1a-2009 Addenda. The NRC has endorsed versions of ASME NQA-1 as meeting many of the regulatory requirements for QA programs. In RG 7.10 Rev 2 (Transportation QA, issued in 2005), the NRC endorsed ASME-NQA-1-1983 in its entirety as meeting the requirements of 10 CFR Part 71, Subpart H. In RG 1.28 Rev 4 (NPP QA, issued in 2010), the NRC endorsed use of ASME-NQA-1-2008 including the NQA-1a-2009 Addenda, subject to a set of additions and modifications listed in the RG. The NRC has as yet made no endorsement of the 2012 or 2015 versions of NQA-1.

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2. RATIONALE FOR PERFORMANCE SPECIFICATIONS

There are 60 specifications in the *STAD Spec*. The rationale is quite simple and short for many of the specifications and is more elaborate for a few specifications. The following sections are organized to discern between the types of specifications.

Note that the *STAD Spec* follows as low as (is) reasonably achievable (ALARA) principles in design and operations. However, unlike the *Transportation, Aging, and Disposal (TAD) Specification* (DOE 2008b), no specific ALARA-related design and operational requirements are provided in the *STAD Spec*. The rationale for not providing such specifications includes the following:

1. ALARA principles must be considered to meet the radiological protection objectives in 10 CFR Part 72 and 10 CFR Part 71, and their invocation of 10 CFR Part 20.
2. ALARA principles must be considered to meet the radiological protection objectives during loading operations at utility sites that are regulated under 10 CFR Part 50, which invokes 10 CFR Part 20.
3. There are no current ISF or geologic repository handling facility designs that dictate a specific ALARA objective.
4. The ISF and geologic repository handling facilities can be designed to accommodate any potential STAD system design that would meet the ALARA objectives of 10 CFR Parts 50, 71, and 72.

2.1 Compliance with CFRs and Conformance with Guidance in SRPs and ISGs

The *STAD Spec* includes 13 generic performance specifications (Table 1) aimed at ensuring compliance with storage regulatory requirements specified in 10 CFR Part 72 and transportation regulatory requirements specified in 10 CFR Part 71. These specifications also aim at ensuring conformance with guidance in SRPs (NUREG-1617 for transportation, and NUREG-1536 Rev. 1 and NUREG-1567 for storage) as well as associated ISGs. The SRPs and ISGs provide regulatory compliance approaches previously endorsed by the US Nuclear Regulatory Commission (NRC) staff. Using these approaches in a new application for an NRC Certificate of Compliance (CoC) will streamline the review process compared to the effort and time the NRC staff would need to review new approaches. Therefore, the *STAD Spec* is based on treating guidance from the SRPs and ISGs as requirements unless the STAD system is incompatible with that guidance, necessitating a new approach. Applicants may use alternative approaches if the previously accepted approaches are not applicable to the particular circumstances of the STAD canister.

Table 1. Generic performance specifications

STAD spec	Requirement summary
3.1.1.1	The 10 CFR Part 71 and 10 CFR Part 72 requirements apply to the loaded STAD canister in a transportation or storage overpack, respectively. NUREG-1536 Rev. 1 and NUREG-1567 for storage, and in NUREG-1617 for transportation, provide approaches that have been accepted by the NRC staff in the past. Although these approaches are considered as guidance by the NRC, they are used as requirements in this performance specification document. Applicants may use alternative approaches if the previously accepted approaches are not applicable to the particular circumstances of the STAD canister.
3.1.2.1	There are no structural requirements or elaborations on requirements beyond those necessary to meet 10 CFR Parts 71 and 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1, NUREG-1567, and NUREG-1617) with applicable NRC ISG documents.
3.1.4.1	Other than those listed below, there are no radiation protection or shielding requirements or elaborations on requirements beyond those necessary to meet 10 CFR Parts 20, 71 and 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1, NUREG-1567, and NUREG-1617) with applicable NRC ISG documents.
3.1.5.1	Other than listed below, there are no criticality safety requirements or elaborations on requirements beyond those necessary to meet 10 CFR Parts 71 and 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1, NUREG-1567, and NUREG-1617) with applicable NRC ISG documents.
3.1.6.1	Other than listed below, there are no confinement and containment requirements or elaborations on requirements beyond those necessary to meet 10 CFR Parts 71 and 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1, NUREG-1567, and NUREG-1617) with applicable NRC ISG documents.

Table 1. Generic performance specifications (continued)

STAD spec	Requirement summary
3.1.7.1	Other than those listed below, there are no operations requirements or elaborations on requirements beyond those necessary to meet 10 CFR Parts 71 and 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1, NUREG-1567, and NUREG-1617) with applicable NRC ISG documents.
3.1.8.1	Other than those listed below, there are no materials requirements or elaborations on requirements beyond those necessary to meet 10 CFR Parts 71 and 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1, NUREG-1567, and NUREG-1617) with applicable NRC ISG documents.
3.1.9.1	There are no security requirements or elaborations on requirements beyond those necessary to meet 10 CFR Parts 71 and 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1, NUREG-1567, and NUREG-1617) with applicable NRC ISG documents. Note that 10 CFR Parts 71 and 72 invoke compliance with 10 CFR Part 73, "Physical Protection of Plants and Materials."
3.2.1	Other than those listed below, there are no requirements specific to the storage and aging configuration or elaborations on requirements beyond those necessary to meet 10 CFR Part 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1 and NUREG-1567) with applicable NRC ISG documents.
3.3.1	There are no requirements specific to the STC or elaborations on requirements beyond those necessary to meet 10 CFR Part 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1 and NUREG-1567) with applicable NRC ISG documents.
3.4.1	There are no requirements specific to the site transporter or elaborations on requirements beyond those necessary to meet requirements in 10 CFR Part 72, including applicable acceptance criteria in associated review plans (NUREG-1536 Rev. 1 and NUREG-1567) with applicable NRC ISG documents.
3.5.1	Other than those listed below, there are no requirements specific to the transportation overpack or elaborations on requirements beyond those necessary to meet 10 CFR Part 71, including applicable acceptance criteria in the associated review plan (NUREG-1617) with applicable NRC ISG documents.
3.7.1	Quality assurance program(s) consistent with the requirements of 10 CFR Part 71, Subpart H (packaging and transportation) and 10 CFR Part 72, Subpart G (storage) shall be used for the design, purchase, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair, modifications, and decommissioning of the STAD canister systems.

2.2 STAD Canister Design Specifications

There are 15 STAD canister design specifications. Table 2 lists these specifications and provides their rationales. If a rationale is not brief enough to be included in a table cell, then the table references the subsection in this document that discusses the rationale in detail.

Table 2. STAD canister design specifications and rationales

STAD spec	Requirement summary	Rationale
3.1.1.2	The design lifetime of the STAD canister shall be 150 years from the time the canister is loaded with SNF to the time the canister is loaded into a disposal overpack; that period could include multiple dry storage and transportation cycles. It is acceptable to use aging management protocols and/or engineered measures to ensure continued compliance with applicable requirements.	See Sect. 2.2.1.
3.1.1.3	The capacity of the small STAD canister shall be either four PWR SNF assemblies or nine BWR SNF assemblies. The outside diameters of the small PWR and BWR canisters shall be the same.	The size of the small STAD canister is derived from repository designs with limited heat dissipation capability, such as vertical boreholes in crystalline rock or clay. Examples of such designs are Cases 1 and 2 (Hardin and Kalinina 2015), and the KBS-3 design shown in SKB 2011, Fig. S-5. A small STAD canister is also used in a salt repository design in Case 4 (Hardin and Kalinina 2015). The requirement for identical outside diameters for PWR and BWR canisters reduces the number of design variants necessary for storage, transportation, and disposal overpacks. Given the differences between PWR and BWR fuel assembly designs and the fact that PWR and BWR STAD canisters will not likely be commingled in a single overpack, a future evolution in the *STAD Spec* may consider different PWR and BWR canister diameters if there are substantiated cost savings and operational advantages.

Table 2. STAD canister design specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.1.4	The capacity of the medium STAD canister shall be either 12 PWR SNF assemblies or 32 BWR SNF assemblies. The outside diameters of the medium PWR and BWR canisters shall be the same.	The size of the medium STAD canister is derived from repository designs with good heat dissipation capability and relatively early emplacement, such as in-drift emplacement in salt or hard rock (unsaturated and saturated zone designs). Aging and/or preclosure ventilation can extend the range of media to include clay. Examples of such designs are Cases 3, 5, 8, 11, and 14 (Hardin and Kalinina 2015). The requirement for identical outside diameters for PWR and BWR canisters reduces the number of design variants necessary for transportation overpacks and for disposal overpacks. Given the differences between PWR and BWR fuel assembly designs and the fact that PWR and BWR STAD canisters will likely not be commingled in a single overpack, a future evolution in the *STAD Spec* may consider different PWR and BWR canister diameters if there are substantiated cost savings and operational advantages.
3.1.1.5	The capacity of the large STAD canister shall be either 21 PWR SNF assemblies or 44 BWR SNF assemblies. The outside diameters of the large PWR and BWR canisters shall be the same.	The size of the large STAD canister is derived from repository designs with excellent heat dissipation capability such as in-drift emplacement in salt or hard rock, along with designs that use a combination of aging and preclosure ventilation to limit peak temperatures in the engineered barrier system and the host rock. Examples of such designs are Cases 6, 9, 12, and 15 (Hardin and Kalinina 2015) and a previous repository design (DOE 2008a, Sect. 1.5.1.1.1.2.1.4). The requirement for identical outside diameters for PWR and BWR canisters reduces the number of design variants necessary for transportation overpacks and disposal overpacks. Given the differences between PWR and BWR fuel assembly designs, a future evolution in the *STAD Spec* may consider different PWR and BWR canister diameters if there are substantiated cost savings and operational advantages.

Table 2. STAD canister design specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.1.6	A STAD canister shall be capable of being loaded with SNF from all facilities licensed by the NRC and holding a contract with DOE for disposal of SNF.	This specification will facilitate broad use of STAD canisters. (See Sect. 2.5.2 for a discussion of potential dry loading/unloading options.)
3.1.1.7	A STAD canister shall be capable of accepting undamaged PWR or BWR SNF assemblies with initial enrichment up to 5 wt% ^{235}U and burnup up to 62.5 gigawatt day (GWd)/metric tons of uranium (MTU). Required cooling (decay) time before loading shall be at least one year and depends on enrichment, burnup, and assembly design.	The maximum allowed initial enrichment for commercial fuel is 5 wt% (10 CFR Part 50). The maximum allowed peak rod average burnup is 62.5 GWD/MTU (NRC 2012). Allowance of variable cooling time will afford maximum flexibility for the thermal and shielding attributes of the STAD canister design. 10 CFR 71.4 and 72.3 define SNF as fuel that has been withdrawn from a nuclear reactor following irradiation, has undergone at least one year of decay 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 fuel includes the special nuclear material, byproduct material, source material, and other radioactive materials associated with fuel assemblies. The previous limits in the TAD specification (DOE 2008b) (i.e., 85 and 75 GWD/MTU for PWR and BWR SNF, respectively) were based on the source term used in the preclosure shielding design. The TAD designs that were submitted to the NRC for review were limited to 45 GWd/MTU (NAC 2009). The current specification requires acceptance of undamaged SNF assemblies, which comprise the majority of the SNF inventory. The ability to accommodate damaged fuel cans, consolidated fuel rods, and fuel debris is not precluded by this specification, but is not required at this time in order to facilitate obtaining a CoC for the initial generation of STAD canisters.
3.1.1.8	The STAD canister shall be designed to be stored in either a horizontal or vertical orientation.	This specification offers the maximum storage configuration flexibility and compatibility with existing dry storage practices.

Table 2. STAD canister design specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.1.9	A STAD canister shall be designed for transportation between sites in a horizontal configuration.	The transportation envelope has a height limit of 15’1” and a width limit of 10’8” (Feldman et al. 2014, Fig. 3-3). The length of SNF assemblies is between 13 and 14’ for much of the inventory, with a small fraction of the inventory being as long as 16’7” (DOE 2008a, Table 1.5.1–2; and Sect. 2.2.2 of this document). Given that the cask design includes structural and shielding components, it is clear that fuel cannot be shipped vertically (or horizontally with the fuel axis across the railcar width). The spent fuel length is not an issue compared to railcar length. Hence, horizontal configuration with the fuel axis and railcar axis aligned is specified.
3.1.1.10	The STAD canister is required to have a circular cross section (in the plane perpendicular to the canister’s long axis).	Currently certified storage and transportation canisters have circular cross sections. To reduce the design and application burden for the STAD canisters and to focus resources on the much longer design lifetime, the *STAD Spec* requires a circular cross section for the STAD canister. It is recognized that other shapes (e.g., square cross sections) could allow for increasing the number of STAD canisters and SNF assemblies that could fit in a single storage or transportation overpack and that other shapes could provide for enhanced heat transfer properties. Conversely, canisters with noncircular cross sections may have reduced structural integrity and may require more elaborate aging management requirements. Thus, it is possible that a future revision of the *STAD Spec* could allow other shapes.

Table 2. STAD canister design specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.1.11	The STAD canister design shall accommodate the varying lengths of the current inventory of SNF (including non-fuel components [e.g., rod cluster control assemblies]) by using a flexible design that can be fabricated at more than one length. The design shall be integrated with storage overpack and transportation overpack designs and with dimensional clearances at existing reactor facilities.	See Sect. 2.2.2.
3.1.1.12	The loaded and closed STAD canister shall be capable of being cut open while submerged in a borated or nonborated pool.	If retrieval of the SNF assemblies becomes necessary, the STAD canisters may need to be handled and reopened in a borated or nonborated pool depending on the criticality safety design of the pool. Note that the Wet Handling Facility design for a previously submitted repository license application included a borated pool (DOE 2008a, Sect. 1.2.5.3.2.1). See Sect. 2.5.2 for a discussion on potential dry loading/unloading options.
3.1.1.13	Loading operations for each STAD canister capacity variant shall be compatible with load limits and crane-lifting capacities at all existing reactor sites with a handling capacity of at least 100 tons. Design of multiple STC variants to accommodate a range of crane capacities is acceptable.	There is a range of crane capacities that must be considered in the STAD canister system design for loading operations, as follows: • 53 reactor sites have at least 125-ton crane capacities (up to 150 tons); however, four sites have administrative restrictions to lower capacities (at least two are below 100 tons). • 13 reactor sites have 100–125 ton crane capacity. At Indian Point*, Unit 2 has a capacity of 110 tons, and Unit 3 has a capacity of 40 tons (Gutherman 2015). STC variants can be designed to accommodate site-specific crane capacities. This allows for optimizing the loading operations (e.g., multiple small STAD canisters in a carrier within an STC) for sites with higher crane capacities without excluding sites with lower crane capacities.

* Indian Point licensed a shielded transfer canister to move fuel from Unit 3 to Unit 2 in lieu of upgrading the 40-ton crane to 125 tons. The STC holds 12 fuel assemblies and is moved with borated water in the fuel cavity (Holtec 2009).

Table 2. STAD canister design specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.1.14	The canister-lifting feature shall be incorporated into the canister top lid and shall not protrude beyond the canister sidewalls.	Prohibition of protruding lifting features beyond the canister sidewalls facilitates packing multiple STAD canisters in a single overpack for storage, transportation, and disposal, as well as a single carrier during handling.
3.1.1.15	All external edges of the STAD canister shall have a radius of curvature sufficient to protect against gouging of the internal surfaces of the overpacks.	This specification helps protect the inner surface of the overpacks (storage, transportation, and disposal), as well as the interfaces with the STC or carrier, preventing gouging that could adversely affect their performance. Note that the TAD (DOE 2008b) specified 0.25 in.
3.1.1.16	Projections or protuberances from reasonably smooth adjacent surfaces shall be avoided or smoothly blended into the adjacent smooth surfaces so that loading into a storage or transportation overpack will be facilitated with a low potential for damage to the interior of the overpack.	This specification ensures minimization of stress risers and line-stress loads on the inner surface of the overpacks (storage, transportation, and disposal). It also ensures compatibility with a cylindrical overpack.

2.2.1 Canister Service Lifetime (Performance Specification 3.1.1.2)

STAD Specification 3.1.1.2 states:

> *The design lifetime of the STAD canister shall be 150 years from the time the canister is loaded with SNF to the time the canister is loaded into a disposal overpack; that period could include multiple dry storage and transportation cycles. It is acceptable to use aging management protocols and/or engineered measures to ensure continued compliance with applicable requirements.*

Selection of a 150-year service life for STAD canisters is tied to assumptions used in previous work, regulatory considerations, and documented descriptions of alternative disposal concepts. It is recognized that an absolute assurance for a 150-year service life cannot be made, but rather that projected performance is based on engineering judgment and is qualified by available data on material performance in potential service environments.

A previous study that evaluated technical feasibility of direct disposal of SNF in dual-purpose canisters (DPCs) adopted an assumption that the combined duration of surface storage and repository operation would not be evaluated beyond 150 years. This assumption was made in order to limit any additional assumptions about long-term stability of institutions responsible for waste management (Hardin and Howard 2013). This assumption is comparable to 40 CFR 191.14(a) which states "...performance assessments that assess isolation of the wastes from the accessible environment shall not consider any contributions from active institutional controls for more than 100 years after disposal." Active institutional controls for the STAD canister prior to disposal can occur at utility sites, at an ISF, and during pre-disposal repository operations. None of these phases is expected to exceed 100 years, and 150 years' aggregate is reasonable. 10 CFR 72.42 limits licensing of ISFSIs to a maximum of 40 years. Each license renewal cannot exceed 40 years. Aging management programs are required as part of the license renewal process (NUREG-1927). The STAD canister lifetime of 150 years establishes the goal to be addressed in the aging management plans.

The 150-year service lifetime provides ample time for STAD canisters to cool before packaging and emplacement in a repository. For some disposal concepts, especially those involving use of clay-based buffer or backfill materials, this cooling time would be essential (see emplacement power limits for Cases 1 to 3 and 11 to 16 in Hardin and Kalinina, 2015; those cases corresponding to the three STAD sizes are summarized in Section 2.3.1 of this document). For other concepts such as the salt repository cases and hard rock unsaturated/non-backfilled cases, disposal could occur much sooner even for higher burnup SNF (compared to assembly aging-burnup power curves in Fig. 2 in Hardin et al., 2013; and Cases 4 to 10 in Hardin and Kalinina, 2015, see Section 2.3.1 of this document).

2.2.2 Assembly Lengths (Performance Specification 3.1.1.11)

STAD Specification 3.1.1.11 states:

> *The STAD canister design shall accommodate the varying lengths of the current inventory of SNF (including non-fuel components [e.g., rod cluster control assemblies]) by using a flexible design that can be fabricated at more than one length. The design shall be integrated with storage overpack and transportation overpack designs and with dimensional clearances at existing reactor facilities.*

The dimensions of unirradiated SNF assemblies for PWR reactors and BWR reactors are provided in Tables 3 and 4, respectively (DOE 2008a, Tables 1.5.1-2 and 1.5.1-3). The listed dimensions do not take nonfuel components (e.g., rod cluster control assemblies) and irradiation effects into consideration. The following is a summary of these dimensional characteristics:

- The PWR assembly population ranges between 111.8–199.0" in length and 6.27– 8.54" in width. Many short assemblies (e.g., 111.8" at Yankee Rowe and 137.1" at Haddam Neck) are already in dry storage.
- The BWR assembly population ranges between 84–176.2" in length and 4.28–6.52" in width. Many of the shorter assemblies (e.g., 84" at Big Rock Point, 95" at Humboldt Bay, and 102.5" at LaCrosse) are already in dry storage.

A single specific STAD canister length was not specified because there are currently no established repository length constraints for the STAD canister, there are considerable differences between PWR and BWR fuel assembly dimensions, and there are relatively significant differences in assembly lengths within the populations of assemblies for each fuel type. Rather, the specification requires a flexible design that can be fabricated at multiple lengths to be specified by the STAD canister designer to efficiently accommodate the range of the inventory.

Note that the TAD canister developed for a previously submitted repository license application had a specified external length range between 186 and 212" (DOE 2008b). The upper limit, which was based on the Naval long waste package design, excluded South Texas and Combustion Engineering System 80 PWR SNF assemblies (considering the part of the canister length allotted to the lids and shield plug). A future variation on the TAD canister design (and associated disposal overpack) was planned for these assembly types. It is not clear what drove the lower length limit. The TAD design that NAC submitted to the NRC for review (and later withdrew) included a TAD canister variant with an external length of 183" (NAC, 2009).

Table 3. Physical characteristics of unirradiated PWR assemblies

Assembly class	Array size	Length (in.)	Width (in.)
B&W 15 × 15	15 × 15	165.7	8.54
B&W 17 × 17	17 × 17		
CE 14 × 14	14 × 14	157.0	8.10
CE 16 × 16	16 × 16	176.8	
CE System 80	16 × 16	178.3	
WE 14 × 14	14 × 14	159.8	7.76
WE 15 × 15	15 × 15	159.8	8.44
WE 17 × 17	17 × 17		
South Texas	17 × 17	199.0	8.43
Ft. Calhoun	14 × 14	146.0	8.10
Haddam Neck	15 × 15	137.1	8.42
Indian Point-1	13 × 14	138.8	6.27
Palisades	15 × 15	147.5	8.20
St. Lucie-2	16 × 16	158.2	8.10
San Onofre-1	14 × 14	137.1	7.76
Yankee Rowe	15 × 16, 17 × 18	111.8	7.62

Table 4. Physical characteristics of unirradiated BWR assemblies

Assembly class	Array size	Length (in.)	Width (in.)
GE BWR/ 2,3	7 × 7, 8 × 8, 9 × 9, 10 × 10	171.2	5.44
GE BWR/ 4-6	7 × 7, 8 × 8, 9 × 9, 10 × 10	176.2	5.44
Big Rock Point	7 × 7, 8 × 8, 9 × 9, 11 × 11	84	6.52
Dresden-1	6 × 6, 7 × 7, 8 × 8	134.4	4.28
Humboldt Bay	6 × 6, 7 × 7	95	4.67
LaCrosse	10 × 10	102.5	5.62

2.2.3 STAD Canister Design Specifications Not Included in the *STAD Spec*

2.2.3.1 Consideration of HLRWS-ISG-1

The following STAD canister specification was proposed in the *STAD Spec*:

> *For storage at an ISFSI or ISF with a site-specific license (as opposed to an ISFSI with a general license), the seismic analysis in NUREG-1567 should consider the probabilistic seismic hazard analysis and performance-based safety assessment guidance in HLWRS-ISG-01 that supplemented the guidance in NUREG-1804 Rev. 2.*

HLWRS-ISG-1 provides an example methodology to review seismically initiated event sequences in the context of the preclosure safety analysis for compliance with 10 CFR 63.111(b)(2). The methodology considers the likelihood of seismic initiating events and the structural fragility of structures, systems, and components (SSCs) to estimate their failure probability. This guidance was developed to take advantage of probabilistic seismic hazard analyses and performance-based safety assessments as opposed to the design-based and deterministic hazard criteria previously used for licensing of nuclear facilities. The probability of occurrence of an event sequence affecting an SSC is determined by convolution of the mean seismic hazard curve with the mean conditional failure probabilities (i.e., fragility) of the SSCs. The mean fragility curve for an SSC may be estimated using (1) probability density functions for controlling parameters in a Monte Carlo analysis; (2) simplified methods outlined in Sect. 4 of Electric Power Research Institute, TR-103959 (EPRI 1994a); or (3) other methods that capture appropriate variability and uncertainty in parameters used to estimate the capacity of the SSCs ITS to withstand seismic events.

The guidance in this ISG cannot be applied at this point, so a specification was not included in the *STAD Spec* for the following two reasons:

1. Seismic hazard curves are site specific and cannot be determined generically.
2. 10 CFR Part 72 and associated SRPs use design-based and deterministic hazard criteria. Therefore, unless the regulation is revised or new NRC guidance is issued, such an analysis would not serve a STAD canister CoC objective.

2.2.3.2 STAD Canister Fragility

The following STAD canister specification was proposed:

> *The STAD canister structural design shall take into consideration potential structural loads onto the STAD canister during handling operations; use of impact limiters (e.g., skirts), if required in the structural design, is acceptable.*

Based on the projected SNF inventory, if tens of thousands of STAD canisters are deployed, they will each be handled multiple times during loading, ISFSI storage, transportation, ISF storage, and repository facility aging and loading into disposal overpacks. These multiple handling operations of the STAD canister may sufficiently raise the number of off-normal structural loads on the STAD canister such that potential canister failure (and subsequent release) may need to be considered in the design and licensing of an ISF.

Future ISF and disposal facilities will be designed to accommodate a variety of waste container variants. Therefore, specific structural requirements cannot be established at this time. Prematurely establishing a structural requirement on a large number of canisters may prove to be costly when compared to shifting the burden onto the ISF and disposal facility designs. Conversely, establishing requirements on a handling facility (e.g., canister alignment, confinement, inert environment) that will need to meet a high throughput rate may prove to be more costly than designing structurally robust canisters. Additionally, during the licensing process of an ISF or geologic repository facility, potential canister failures may also present more opportunities for contentions, hearings, and regulatory complexity, as well as negative perception.

Taking the above considerations into account, this specification was not included at this time, but it may be considered in the future.

2.3 STAD Canister System Safety Functions Specifications

There are ten specifications in the *STAD Spec* that are related to the STAD canister system safety functions. Table 5 provides the rationales for these specifications. The general safety functions of storage are listed in the definition of ITS SSCs in 10 CFR Part 72, which states:

> *(1) To maintain the conditions required to store spent fuel … safely; (2) To prevent damage to the spent fuel … [and] waste container during handling and storage; or (3) To provide reasonable assurance that spent fuel … can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.*

To meet these regulatory requirements, Sect. 4.4.3.1 in NUREG-1567 states:

> *The basic design criteria for SSCs which are important to safety shall: maintain subcriticality, maintain confinement, ensure radiation rates and doses for workers and public do not exceed acceptable levels and remain as low as is reasonably achievable (ALARA), maintain retrievability, and provide for heat removal (as necessary to meet the above criteria).*

Subparts E and F of 10 CFR Part 71 include three main safety requirements for the transportation cask and its contents: subcriticality, containment, and radiation protection under normal conditions of transport (NCT) and hypothetical accident conditions (HAC). Additionally, 10 CFR 71.55(d) provides requirements that the geometry of the package contents and packaging effectiveness must be maintained under the tests specified in 10 CFR 71.71, “Normal conditions of transport.” In order to meet these safety requirements, specific performance characteristics are necessary.

Therefore, the storage and transportation safety functional areas are grouped as follows:

- subcriticality,
- confinement for storage and containment for transportation,
- radiation protection,
- retrievability for storage and geometry control for transportation, and
- thermal performance.

Table 5. STAD canister system safety function specifications and rationales

STAD spec	Requirement summary	Rationale
3.1.3.1	SNF cladding temperatures in STAD canisters shall meet applicable limits established in NUREG-1536 Rev. 1 for storage and loading operations. For transportation, the cladding temperatures in STAD canisters during NCT shall not exceed 400°C. For transportation HAC, this specification does not impose cladding temperature limits.	The *STAD Spec* requires that intact cladding be preserved during storage and transportation because the repository designer may choose to take performance credit for cladding during disposal. Refer to NUREG-1536 Rev. 1 (SRP) Sects. 2.4.3.6, 4.4.2, and 8.8.1 for the evaluation of the impacts of elevated temperatures (i.e., > 400°C) on cladding degradation and structural characteristics. The cladding temperature limit of 400°C is specified during NCT to ensure that the potential for cladding degradation is minimized during transportation. Note that there are regulatory options under which a storage or transportation applicant could otherwise design a system that would allow for this temperature limit to be exceeded. Such options could include: 1. not relying on cladding integrity for retrievability during storage by placing the SNF assemblies in damaged fuel cans (DFCs), and 2. demonstrating that transportation safety functional requirements can be met without crediting the geometry of the cladding (e.g., moderator exclusion under HAC).
3.1.3.2	The repository thermal management specification is included in Sect. 2.3.1.	See Sect. 2.3.1 for rationale.

Table 5. STAD canister system safety functions specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.4.2	The STAD canister shall be designed so that removable surface contamination on an accessible external surface shall be less than 1,000 dpm/100 cm^2 beta-gamma and 20 dpm/100 cm^2 alpha.	The STAD canister concept is based on sequential movements of the loaded and sealed canister into a storage overpack, a transportation overpack, and a disposal overpack. Operations in facilities that load or remove the STAD canister from the overpack will be simplified if the outer surface of the STAD canister meets these surface decontamination limits. These limits are taken from the NRC guidance in IE Circular 81-07 (included as Appendix A) and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Only loose contamination is controlled, as fixed contamination will not result from the loading process.
3.1.5.2	SFST-ISG-8 Rev. 3, *Burnup Credit in the Criticality Safety Analyses of PWR Spent Fuel in Transport and Storage Casks*, shall be applied.	Although burnup credit may not be required to meet storage and transportation criticality safety requirements, it is expected that burnup credit will be required to demonstrate subcriticality during disposal. Burnup credit analyses rely on many detailed parameters including assembly design, burnup, enrichment, irradiation history, and consideration of misload. Measures taken to reduce misload likelihood could include detailed documentation of assembly characteristics prior to loading and/or burnup measurements. A burnup credit analysis basis for the STAD canister would ensure that the needed information, already accepted by the regulator, is readily available for disposal burnup credit during repository licensing.

Table 5. STAD canister system safety functions specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.5.3	SFST-ISG-19, *Moderator Exclusion under Hypothetical Accident Conditions and Demonstrating Subcriticality of Spent Fuel under the Requirements of 10 CFR 71.55(e)*, shall be applied to ensure that moderator exclusion under HAC can be used as a design approach and CoC basis.	Although moderator exclusion may not be required to meet transportation HAC criticality safety requirements (based on a combination of crediting cladding integrity, burnup credit, and neutron absorbers), it is a specification in the *STAD Spec* for the following reasons: 1. It will allow for the moderator exclusion option to be exercised for future transportation cycles (e.g., from an ISF to a geologic repository) after an extended storage period at an ISF during which the cladding properties may change, necessitating the consideration of more conservative geometries (e.g., fuel rubble) than were assumed in the initial transportation packaging certification (i.e., CoC). 2. It will ensure a robust structural design of the STAD canister, taking into consideration transportation HAC structural loads. Enhanced structural robustness may prove to be important given the multiple storage and transportation cycles to which the STAD canister may be subjected.
3.1.5.4	Neutron absorber specification text is included in Sect. 2.3.2.	See Sect. 2.3.2 for rationale.
3.1.6.2	The STAD canister shall be designed to be “leak tight” as defined in ANSI N14.5-2014.	NUREG-1536 Rev. 1, Section 5.4.4 states that dose consequence analyses are unnecessary for storage casks, including closure lid, which are designed and tested to be “leak tight” as defined in ANSI N14.5.
3.1.6.3	The STAD canister shall constitute the confinement boundary during storage. 10 CFR 72.236 requires redundant sealing of confinement systems for a storage cask. This performance specification document assigns that requirement to the STAD canister, requiring dual welded closures.	During its lifecycle, the STAD canister may be stored at multiple sites (ISFSI, ISF, and a geologic repository) in potentially different configurations (concrete storage overpack, concrete module, metal storage cask, vault). Assigning the confinement function to the STAD canister will provide for maximum flexibility for the storage configurations and associated safety functional requirements. Additionally, this specification will ensure a robust structural design of the STAD canister. Enhanced structural robustness may prove to be important given the multiple storage and transportation cycles to which the STAD canister may be subjected.
3.1.6.4	Helium shall be the STAD canister fill gas for storage and transportation.	Helium has the most effective heat transfer properties of all the inert gases.

Table 5. STAD canister system safety functions specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.6.5	Because STAD canisters may be stored at sites with a wide range of seismological characteristics, the design of the STAD canister shall assume a standardized design earthquake (DE) ground motion described by an appropriate response spectrum anchored at 3 g in lieu of the regionally- and geologically-based seismological characteristics described in 10 CFR 72.103. Following a seismic event characterized by horizontal and vertical peak ground accelerations of 3 g, the STAD canister in a storage or aging configuration shall maintain confinement consistent with the requirements of 10 CFR Part 72 and the guidance in NUREG-1536, Rev. 1.	See Sect. 2.3.3 for rationale.

2.3.1 Repository Thermal Management (Performance Specification 3.1.3.2)

Specification 3.1.3.2 states:

> *To meet repository thermal management objectives, the maximum cladding temperature after disposal shall not exceed 400°C given emplacement of a disposal overpack containing a single STAD canister with the power and surface temperature boundary conditions provided in Table 1 [of the STAD Spec]. For the purpose of canister internal temperature analysis, the canister outer surface temperature may be assumed to be 2°C hotter than the disposal overpack outer surface temperature.*

Table 1 [from the *STAD Spec*]. Thermal boundary conditions for disposal

Concept	STAD canister thermal power	Disposal overpack surface temperature boundary condition
Small STAD canister	2,200 W	200°C
Medium STAD canister	5,500 W	200°C
Large STAD canister	10,000 W	200°C

The thermal performance of a repository is based on meeting temperature limits on engineered components (outside of the disposal overpack) and natural barriers. The repository designer selects the repository host geologic medium, spacing between disposal overpacks, engineered barrier system design, and limits on thermal power from each disposal overpack (in turn based on canister capacity, burnup, and aging). However, the repository designer will not have control of the internal design of the existing STAD canisters. For the purposes of the *STAD Spec*, the cladding temperature limit for disposal will be met by controlling the STAD canister internal design for a set of boundary conditions determined from existing repository design studies.

A series of parametric calculations have established design points for several repository rock types and engineered barrier system concepts; these calculations determine the appropriate size of the STAD canister for each geologic medium. The repository calculations use STAD canister thermal power as an input and determine the disposal overpack surface temperature history as an output based on heat transfer into and through the host rock. These two parameters (canister thermal power and overpack surface temperature) are suitable inputs for the STAD canister designer to determine peak cladding temperature based on the internal design of the STAD canister. The values in Table 1 of the *STAD Spec* are those that are the most restrictive to the STAD canister designer. In general, both higher canister thermal power and lower temperature difference between the cladding and disposal overpack surface require the canister design to be more effective at transferring heat from the SNF to the canister surface. Table 6 shows these values for twelve repository designs (Hardin and Kalinina 2015). The highlighted cases have the largest thermal power at closure and smallest temperature difference between the STAD interior and disposal overpack surface, and thus they are the controlling cases; the non-highlighted cases do not need to be analyzed by the STAD canister designer.

Because the STAD canister designer does not control the design of the disposal overpack, the *STAD Spec* assumes a 2°C temperature difference between the outer surface of the disposal overpack (the location of repository designer temperature calculations) and the outer surface of the STAD canister (the location of the STAD canister designer's boundary condition). This assumption is based on calculations supporting Hardin and Kalinina, 2015; Table 7 shows three examples.

In the *STAD Spec*, the near-field temperature variation is neglected for "open" disposal concepts for which (1) an air gap is maintained around waste packages after repository closure and (2) in which a drip shield is included (Hardin and Kalinina 2015, cases 8, 9, and 10). None of these cases is used as the basis for controlling postclosure thermal conditions in Table 6, and the *STAD Spec* does not require the STAD canister designer to include the near-field temperature variation for the following reasons. For Cases 8, 9 and 10 (Hardin and Kalinina 2015), the controlling near-field temperature is a 200°C limit for the host rock at the emplacement drift wall. Between the waste package surface and the drift wall is an air gap of approximately 1 to 2 m and a metal drip shield that acts as a thermal radiation shield. The response of both has been shown to result in a temperature drop from the waste package to the drift wall of 10 to 20°C at the time of the waste package peak temperature (BSC 2008, Sect. 6.1.4). The combination of overpack temperature drop plus thermal radiation effects for "open" concepts tends to increase the STAD canister surface temperature. However, the increase is small compared to the overall magnitude of temperature increases in a repository. If the then-existing STAD canisters would not meet cladding temperature limits for the conditions highlighted in Table 6 with the addition of ΔT from Table 7 plus the thermal radiation effect, it is assumed that the repository designer will be able to accommodate that situation by increasing disposal overpack spacing and/or by using waste aging or ventilation before repository closure.

For preclosure handling at a repository, thermal power will be higher than for postclosure, but it will be less than for storage and transportation due to radioactive decay during the sequence of these stages of waste management. The repository surface facility designer can take advantage of handling equipment designs in upstream facilities to ensure that the NUREG-1536 Rev. 1 and SFST-ISG-11 limits will also be met during preclosure repository operations.

Table 6. Limiting repository thermal power and overpack temperatures

Case number[a]	Description	Maximum emplacement disposal overpack power (kW)[b]	Maximum repository closure disposal overpack power (kW)[b]	Maximum disposal overpack surface temperature (°C)
Small STAD canister, 4 PWR SNF assemblies				
1	Crystalline rock, enclosed, bentonite buffer, backfill	1.7		100
2	Clay, enclosed, bentonite buffer, backfill	1.7		100
4[c]	Salt, enclosed, backfill	2.2		200
Medium STAD canister, 12 PWR SNF assemblies				
3	Clay, enclosed, backfill	1.7		200
5[c]	Salt, enclosed, backfill	5.5		200
8	UZ[d] hard rock, open, drip shield	10	4	200
11	SZ[e] hard rock, open, backfill	10	2	200
14	Clay, open, backfill	10	2	200
Large STAD canister, 21 PWR SNF assemblies				
6[c]	Salt, enclosed, backfill	10		200
9	UZ hard rock, open, drip shield	18	7	200
12	SZ hard rock, open, backfill	18	3	200
15	Clay, open, backfill	18	3	200

[a]Hardin and Kalinina 2015.
[b]For "open" designs (ventilation between the disposal overpack and the rock wall until closure), the closure power is used because ventilation removes ~85% of the thermal power before closure.
[c]Highlighted lines require the most effective heat transfer within the canister; non-highlighted lines do not need to be analyzed by the STAD canister designer
[d]UZ = unsaturated zone.
[e]SZ = saturated zone.

Table 7. Disposal overpack temperature drop for postclosure thermal conditions

Canister capacity, PWR assemblies	Case #	Post-closure power, kW	Disposal overpack OD, m	Disposal overpack length, m	Overpack wall thickness, m	Overpack K_{th}, W/m-K	Overpack heat flow area, m^2	ΔT between STAD and overpack outer surfaces, °C
4	4	2.2	0.82	5	0.15	30	11.2	1.0
12	5	5.5	1.02	5	0.15	30	14.8	1.9
21	6	10	1.57	5	0.07	15	27.1	1.7

For comparison purposes, the safety analysis report (SAR) for a previous repository design (DOE 2008a, Sect. 1.5.1.1.1.2.6) requires that the peak cladding temperature not exceed 350°C for three combinations of thermal power and TAD canister surface temperature: 11.8 kW and 274°C, 18 kW and 232°C, and 25 kW and 181°C. Sect. 1.5.1.1.1.1 of DOE 2008a states that no postclosure performance credit is taken for commercial SNF cladding capability but that fuel failure is considered in preclosure safety analyses. The lower temperature limit for disposal than for storage and transportation was due to conservatism in the licensing approach for the previous repository design.

2.3.2 Criticality Control (Performance Specification 3.1.5.4)

Specification 3.1.5.4 states:

To meet repository objectives, the following six requirements are prescribed as a group:

- *a) Neutron absorber plates or tubes shall be made from borated stainless steel produced by powder metallurgy and meeting ASTM A887-89 (2014), "Standard Specification for Borated Stainless Steel Plate, Sheet, and Strip for Nuclear Application", Grade A alloys.*
- *b) Minimum thickness of neutron absorber plates between SNF assemblies shall be 11 mm (0.4375 in.) assuming single plates. Use of multiple plates between the SNF assemblies (i.e., flux traps) is prohibited.*
- *c) The neutron absorber plate shall have boron content of 1.1–1.2 wt %, a range that falls within the specification range for 304B4 (Unified Numbering System [UNS] S30464) as described in ASTM A887-89 (2014).*
- *d) Neutron-absorbing material shall extend the full length of the fuel basket.*
- *e) Neutron-absorbing plates shall either surround each assembly or extend the full cross section of the STAD canister (in the plane perpendicular to the canister's long axis).*
- *f) The borated stainless steel plates shall be incorporated into the basket without the use of welding.*

ASTM-A887-89 describes eight borated stainless steel alloy types with varying boron content (0.20% to 2.35% boron), with two grades specified for each. Grade A must have smaller and more spherical boride (Cr_2B) particles in comparison with Grade B, and therefore Grade A alloys have better mechanical properties. Only Grade A alloys are allowed for the STAD canister design. This group of six requirements was developed for a previously submitted license application for geologic disposal in an unsaturated tuff geology (DOE 2008b). Based on the generic analyses that have been performed thus far for a range of host geologies, there is confidence that this material as specified will perform favorably. Nonetheless, once a site is selected, site-specific analyses coupled with total system performance models will have to be performed to confirm acceptable performance.

Borated stainless steel with 1.1–1.2 wt.% of natural boron and a thickness of at least 6 mm was determined to have the necessary neutron absorption capacity to maintain subcriticality for the TAD canisters based on burnup credit, moderation, and geometry assumptions developed for a previous repository design (DOE 2008b). The disposal criticality analysis for the STAD canister is assumed to have similar bases and therefore similar neutron absorber loading requirements. Borated stainless steel offers improved durability and corrosion resistance compared to the commonly used aluminum-based neutron absorber materials. Use of 304B4 plates that are initially 11 mm thick provides a corrosion allowance of 5 mm. The corrosion of borated stainless steel components is expected if both the disposal overpack and the stainless steel STAD canister are breached after repository closure. The influx of water vapor or liquid water will promote corrosion of the borated steel under both oxic and anoxic conditions. A literature review of the available experimental data on the modes and rates of corrosion for borated stainless steel and their comparison to corrosion of nonborated stainless steel is presented in Appendix B.

Based on this literature review, borated stainless steel corrodes more rapidly than nonborated stainless steel. A corrosion rate of 250 nm/year is assumed for borated stainless steel. This value falls within the available rate values for the simulated in-package conditions and is higher than the borated stainless steel corrosion rates measured in nonaggressive aqueous media. A corrosion rate of 250 nm/yr corresponds to a loss in plate thickness of 5 mm in 10,000 years assuming two-sided corrosion. Immediate failure of both the overpack and the stainless steel canister following permanent disposal is unlikely; therefore, the time period during which borated stainless steel components would corrode under either inundated or humid conditions is expected to be shorter than an assumed regulatory performance period of 10,000 years.

Borated stainless steel has had limited use as a neutron absorber in dry storage canisters and casks (Greene et al. 2013). Two certified canister designs, the FuelSolutions™ W74 and AREVA TN NUHOMS® 52B (a total of 34 canisters), use borated SS304 plates as neutron absorber plates. Three dry storage or dual-purpose casks for bare fuel have been designed to use borated stainless steel as a neutron absorber, but only the Castor V/21 is in commercial use, with 25 casks at the Surry Power Station. Current and proposed storage-related uses of borated stainless steel are discussed in detail in Appendix C.

Limited experimental data are available on the corrosion of borated stainless steel; therefore, critical areas for future research have been outlined. In particular, the modes and rates of corrosion of both borated and nonborated stainless steel under anoxic conditions are not well defined. The *STAD Spec* may be revised in the future based on results obtained from potential future experimental work for borated stainless steel as discussed in Appendix D.

To ensure neutronic decoupling in case of assembly or fuel pin displacement after horizontal emplacement in a repository, specification item (e) requires the boron absorber plates to either surround the assemblies or extend to the inner wall of the STAD canister. Similarly, specification item (d) requires the boron absorber plates to extend axially to prevent neutronic coupling in the case of axial displacement of the fuel assemblies or fuel pins after emplacement in a repository.

Borated stainless steel plates are prohibited from being joined with basket components using fusion or other hot welding techniques because the residual stress from the welds would need to be mitigated. Preliminary studies have shown that high temperature annealing at 1200ºC may be sufficient for mitigating accumulated stresses within the heat-affected zones. Protocols need to be developed for the appropriate weld mitigation. A summary of the welding-induced alterations in borated stainless steel and the proposed mitigation techniques is presented in Appendix E.

Borated stainless steel was the design specification neutron absorber in the TAD canister (DOE 2008a), and was accepted by the NRC in their evaluation of a previously submitted repository license application (NUREG-1949); however, TAD canisters were never certified.

Alternative criticality control specifications that may be considered in a future evolution of the *STAD Spec* include:

1. use of ^{10}B enriched boron in the borated stainless steel plates which would allow for the use of thinner plates and potentially enhanced corrosion properties due to lower overall boron content. Note that there are currently insufficient data to quantify the corrosion rate as a function of boron loading in the stainless steel for the range of conditions in the various potential repository geologies. Additionally, the minimum borated stainless steel plate thickness remaining after the corrosion allowance needs to be substantial enough to stay between the SNF assemblies without breaking or cracking, which would allow for neutron streaming and reduced neutron absorber efficacy,

2. use of other alloys such as Ni-Gd that may have improved corrosion characteristics. Note that gadolinium isotopes are more effective thermal neutron absorbers than boron isotopes,

3. use of moderator displacers and fillers,
4. use of soluble neutron absorbers that would dissolve into the incoming water and act as a soluble neutron absorber. Establishing the type and amount of the soluble neutron absorber material would need to take into consideration the range of water flow rates through the disposal overpack such that the required neutron absorber concentration in the solution would be maintained. This may have significant impacts on the waste package chemistry,
5. use of a robust barrier that would preclude water for the disposal regulatory period,
6. use of two separate neutron absorber materials with contrasting corrosion characteristics to offer redundancy for defense-in-depth, and
7. removal of the specification and reliance on repository geology to introduce the necessary neutron absorbers dissolved in the incoming water. For example, groundwater chloride concentration of 2 molal, which is well below potential concentrations in a salt geology, would essentially preclude criticality, assuming burnup credit. Appendix G discusses in detail the potential solutes and their impacts.

2.3.3 Confinement during Seismic Events (Performance Specification 3.1.6.5)

Specification 3.1.6.5 states:

> *Because STAD canisters may be stored at sites with a wide range of seismological characteristics, the design of the STAD canister shall assume a standardized design earthquake (DE) ground motion described by an appropriate response spectrum anchored at 3 g in lieu of the regionally- and geologically-based seismological characteristics described in 10 CFR 72.103. Following a seismic event characterized by horizontal and vertical peak ground accelerations of 3 g, the STAD canister in a storage or aging configuration shall maintain confinement consistent with the requirements of 10 CFR Part 72 and the guidance in NUREG-1536, Rev. 1.*

10 CFR 72.103 provides the geological and seismological considerations for various regions within the US. For example, for sites east of the Rocky Mountain Front where geological investigations conclude no unstable characteristics, a standardized design earthquake (DE) ground motion described by an appropriate response spectrum anchored at 0.25 g may be used. For sites west of the Rocky Mountain Front and in areas of known potential seismic activity east of the Rocky Mountain Front, seismicity must be evaluated by the techniques presented in 10 CFR 72.103(f). Note that regardless of the results of the seismological evaluation, the minimum DE horizontal ground motion assumed in the design must be no less than 0.10 g (10 CFR 72.103(f)(3)). If an ISFSI is located on a nuclear power plant (NPP) site, the existing geological and seismological design criteria for the site may be used. Note that if the existing design criteria for the NPP are used and the site has multiple NPPs, then the criteria for the most recent NPP must be used.

There is a relatively wide range of ground motion design bases for the various ISFSI sites throughout the US. Therefore, storage systems with a general license are often designed taking into consideration a higher DE ground acceleration than 0.25 g. For example, the MAGNSTOR storage system (NAC 2004), which is used at Zion, is designed for a maximum ground acceleration of 0.37 g (NAC 2004, Section 3.7.3.4). The NUHOMS storage system (Transnuclear 2003), which is used at several ISFSIs including San Onofre, is designed based on an earthquake that produces accelerations in two horizontal directions of 1.5 g and a vertical acceleration of 1.0 g acting simultaneously (Transnuclear 2003, Section 3.1.2.1.7).

DOE assumed a DE with a peak horizontal acceleration of 0.75 in its centralized interim storage facility (CISF) topical safety analysis report (TSAR) (DOE 1997). The CISF TSAR indicates that 0.75 g peak horizontal acceleration will bound the peak horizontal acceleration values at most sites in the US, evaluated at a mean annual probability of 1 x 10^{-4} or higher, except some sites in the western US. The

NRC notes in their final assessment report (NRC 2001) that "The DOE, however, needs to demonstrate that a peak horizontal acceleration of 0.75 g will bound the estimated peak horizontal acceleration at the selected site following the procedures outlined in 10 CFR Part 100, Appendix A, if the selected site is west of the Rocky Mountain Front…"

The TAD Specification developed for a previously submitted repository license application included the following specification (DOE 2008b, Section 3.3.2(1)c):

> *Following a seismic event characterized by horizontal and vertical peak ground accelerations of 96.52 ft/s^2 (3 g):*
> - *TAD canister in an aging overpack, shall maintain a maximum leakage rate of 1.5 x 10^{-12} fraction of canister free volume per second (normal)*
> - *Canister design codes may be exceeded (i.e., vendor may rely on capacity in excess of code allowances).*
> - *The aging overpack shall remain upright and free standing during and following the event.*

In its review of a previously submitted repository license application, the NRC states (NUREG-1949):

> *The NRC staff finds the applicant's design criterion that the vertical [aging overpack] AO must remain upright and free standing without exceeding the allowable leakage rate of the canister during and post seismic event with horizontal and vertical peak ground accelerations (PGAs) of 96.52 ft/s^2 (3 g) acceptable because it corresponds to a probability of exceedance of 10$^{−6}$ per year, which was reviewed and found to be acceptable by the NRC staff in SER Section 2.1.1.1.3.5.2.*

Therefore, in order to envelope the seismological characteristics of potential sites where the STAD canisters could be stored without imposing undue seismic design criteria, two storage overpack/module variants shall be designed. The variant designed for 0.25 g ground acceleration would be usable at most locations, whereas the variant designed for 0.75 g ground acceleration would be usable at the few sites with higher seismicity.

The STAD canister is required to be designed to maintain confinement in a storage configuration taking into consideration ground accelerations of 3 g to ensure that the STAD canisters could be stored at sites with higher ground accelerations than 0.75 g (up to 3 g) once an appropriate overpack/module is designed taking into consideration the site appropriate DE.

2.4 STAD Canister Operational Specifications

There are four STAD canister system operational specifications. Table 8 provides the rationales for these specifications.

Table 8. STAD canister operational specifications and rationales

STAD spec	Requirement summary	Rationale
3.1.7.2	The STAD canister lid shall be designed for handling under water with the STAD canister in a vertical orientation.	STAD canisters will be loaded vertically in SNF pools and are expected to be handled dry, either vertically or horizontally, at an ISF or repository facility. However, in the case of needing to retrieve the fuel assemblies from the STAD canister, retrieval operations will likely be conducted under water. See Sect. 2.5.2 for a discussion on potential dry loading/unloading options.
3.1.7.3	A feature for lifting a vertically oriented loaded STAD canister, with the lifting feature mating with the lid, shall be provided. The lifting feature may be integral with the lid or mechanically attached.	A common integral lifting feature will facilitate handling of the STAD canisters in a manner that will meet potentially high throughput requirements while maintaining ALARA principles.
3.1.7.4	An open, empty, and vertically oriented STAD canister shall have integral lifting feature(s) provided to allow lifting by an overhead handling system.	This specification facilitates handling of an empty STAD canister body (without lid), using an integral lifting feature.
3.1.7.5	It is acceptable to use a carrier approach to load, close, and move STAD canisters in groups.	This specification allows for operational flexibility in handling the STAD canisters individually or in groups.

2.5 STAD Canister Material Specifications

There are seven STAD canister material specifications. Table 9 provides the rationales for these specifications.

Table 9. STAD canister material specifications and rationales

STAD spec	Requirement summary	Rationale
3.1.8.2	The STAD canister shell and lid shall be designed and fabricated in accordance with ASME (2013) *Boiler and Pressure Vessel Code*, Section. III, Division 1, Subsection NB or NC to the extent practicable. The vendor shall identify applicable code exceptions, clarifications, interpretations, and code cases.	NUREG-1536, Rev. 1, Sect. 3.4.1.1 states that the NRC has accepted the use of these subsections. Subsection NB applies to Class 1 components (that are part of a pressure boundary whose failure would violate containment). Subsection NC applies to Class 2 components (that are not Class 1 but are important for post-accident situations). It is noted that the ASME *Boiler and Pressure Vessel Code* language is specific to the reactor coolant pressure boundary, and the wording here uses the analogy of the storage confinement boundary; this is similar to the treatment in NUREG-1536, Rev. 1, Sect. 3.5.1.3. Section 8.4.2.2 provides guidance on the use of code cases, citing Regulatory Guide 1.193.
3.1.8.3	Required materials – The STAD canister and structural internals (i.e., basket, but not thermal shunts and criticality control materials) shall be Type 300-series stainless steel as listed in ASTM A-276-13a, *Standard Specification for Stainless Steel Bars and Shapes*.	This specification addresses the need to consider the potential for canister failure by corrosion during storage. Stainless steels of the 300 series offer many benefits as a canister shell material, including strength, ductility, and weldability. Moreover, they are sufficiently corrosion-resistant so that failure by general corrosion within the 150-year design lifetime of the canister is not possible.
3.1.8.4	Potential problems from uniform corrosion, pitting, stress corrosion cracking, or other types of corrosion shall be evaluated for the environmental conditions and dynamic loading effects that are specific to the component. Because it is assumed that a separately evaluated disposal overpack will be used for disposal, this requirement refers to environmental conditions during storage or aging as well as during transport.	See Section. 2.5.1.

Table 9. STAD canister material specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.8.5	All external welds except the closure welds shall be treated (e.g., stress relieved) prior to loading to mitigate the potential for stress corrosion cracking. The final closure welds shall be capable of being treated after loading. The triggers and timing for treating the final closure welds shall be determined as part of the aging management plan developed in support of CoCs and licenses.	This specification eliminates most high tensile stresses present as weld residual stresses and greatly reduces the risk of canister penetration by stress corrosion cracking (SCC).
3.1.8.6	The STAD canister and its basket materials shall be designed to be compatible with both borated and nonborated pool water.	The STAD canister will be loaded at both borated PWR and nonborated BWR pools.
3.1.8.7	The following is a list of prohibited or restricted materials. a) The STAD canister shall not use organic, hydrocarbon-based materials of construction. b) The STAD canister shall not be constructed of pyrophoric materials. c) The STAD canister (including the basket, thermal shunts, criticality control materials, gaskets, seals, adhesives, and solder) shall not be constructed with materials that would be regulated as hazardous wastes under the Resource Conservation and Recovery Act (RCRA) and prohibited from land disposal under RCRA if declared to be waste. Specific sections of the Environmental Protection Agency (EPA) regulations defining hazardous wastes are listed in Section 4 of this performance specification document (Glossary).	Organic, hydrocarbon-based materials are subject to decomposition within a radiation field, producing reactive gasses. The extremely low organic carbon supply in the repository will limit heterotrophic microbial activity that could otherwise accentuate corrosion or radionuclide transport (BSC 2004b). Pyrophoric materials are prohibited in repository waste packages by 10 CFR 60.135. A pyrophoric event is defined as ignition followed by rapid chemical oxidation or self-sustained burning. A constraint on availability of pyrophoric materials for exothermic reaction was used to exclude exothermic reactions in the engineered barrier system of a previous repository design (BSC 2000). Use of RCRA-regulated materials could complicate licensing a geologic repository, involving approval by the host state and the EPA, as well as the NRC.

Table 9. STAD canister material specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.1.8.8	The following is a list of marking requirements. a) The STAD canister shall be capable of being marked on the lid and body with an identical unique (vendor independent) identifier prior to delivery for loading. b) The markings shall remain legible for the 150-year service life of the STAD canister without intervention or maintenance during normal operations and off-normal conditions associated with loading, closure, storage, transportation, aging, and placement in a disposal overpack.	This specification provides for the capability to implement the requirements for material control and accountability. See 10 CFR Part 74, *Material Control and Accounting of Special Nuclear Material,* for additional information.

2.5.1 STAD Canister Materials Corrosion (Performance Specification 3.1.8.4)

Specification 3.1.8.4 states:

> *Potential problems from uniform corrosion, pitting, stress corrosion cracking, or other types of corrosion shall be evaluated for the environmental conditions and dynamic loading effects that are specific to the component. Because it is assumed that a separately evaluated disposal overpack will be used for disposal, this requirement refers to environmental conditions during storage or aging as well as during transport.*

This specification addresses the need to consider the potential for canister failure by corrosion during storage. Stainless steels of the 300 series offer many benefits as a canister shell material, including strength, ductility, and weldability. Moreover, they are sufficiently corrosion-resistant that failure by general corrosion within the 150-year design lifetime of the canister is unlikely. Failure by pitting or crevice corrosion is also unlikely to occur. However, recent analyses by the Nuclear Waste Technical Review Board (NWTRB 2010), the Electric Power Research Institute (EPRI 2011), the DOE Used Fuel Disposition Program (Hanson et al. 2012), and the Nuclear Regulatory Commission (NRC 2012) have identified the potential for canister penetration by stress corrosion cracking (SCC) as a major concern with respect to the safety performance of long-term interim storage.

The current understanding of canister SCC is discussed in Appendix F. Three criteria must be met for SCC to occur: (1) the metal must be susceptible to SCC, (2) a corrosive (e.g., chloride-rich) environment must be present, and (3) tensile stresses exceeding a threshold value must be present in the metal. Stainless steels of the 300 series are susceptible to SCC. Moreover, a chloride-rich environment can form on canisters by deliquescence of salt aerosols that are deposited on the canisters from the air flowing through the storage overpack. Recent canister surface inspections (Bryan and Enos 2014, EPRI 2014) have confirmed that chloride salts are present on the surface of in-service storage canisters in near-marine settings; canisters at inland sites have not been evaluated. Finally, residual stress modeling by the NRC has indicated that high tensile stresses are likely to be present in weld and weld heat-affected-zones (HAZ) of canisters currently in service (NRC 2013). A literature search of measured SCC growth rates indicates that should SCC initiate, penetration of a dry storage canister wall within 150 years is possible (Appendix F).

2.5.2 Dry Loading/Unloading Options for STAD Canisters (not included in the *STAD Spec*)

Performance specifications 3.1.1.6, 3.1.1.12, and 3.1.7.2 require that the STAD canister shall have the capability of being loaded and unloaded into and out of a pool. The *STAD Spec* does not specifically preclude the design of the STAD canister from being loaded or unloaded in a dry environment. Loading operations of most of the STAD canisters are expected to occur at reactor sites; therefore, the ability to load the STAD canister in a pool is specified in the *STAD Spec* and must be maintained in all future evolutions of the *STAD Spec*.

STAD canisters may also be loaded at an ISF or a geologic repository if the SNF arrives in bare fuel transportation casks or if the need arises to repackage the SNF from DPCs or from larger STAD canisters than will be used in disposal. The option of transferring the fuel in a dry environment may be viable and may offer operational advantages. Dry SNF handling is not uncommon (e.g., the reprocessing facility at La Hague) and was initially considered for waste package loading operations (dry transfer facility) in a previously submitted US repository license application (DOE 2008a), but it was later abandoned in favor of transferring SNF assemblies in a pool (wet handling facility).

Dry transfer operations will require a facility with temperature, confinement, and environment (e.g., inert) controls, as well as the capability to remotely perform closure welds and nondestructive examination (NDE). When the functional and operational requirements of a dry transfer facility have been established, dry loading/unloading specifications for the STAD canister can be developed, and the *STAD Spec* can be revised to include the interface specifications for the dry transfer facility.

2.6 Storage and Aging System Specifications

There are five STAD canister storage and aging system specifications. Table 10 provides the rationales for these specifications.

Table 10. STAD canister storage system specifications and rationales

STAD spec	Requirement summary	Rationale
3.2.2	Storage and aging configurations for STAD canisters include the use of overpacks, modules, or vault systems.	This specification provides the flexibility that storage configurations may use either of the commonly used storage configurations (vertical overpack or horizontal modules) as well as vaults, which are currently in limited use.
3.2.3	Overpacks/modules may be designed to accommodate single or multiple STAD canisters. The use of a multicanister fixture to facilitate handling and/or storage is acceptable.	Storing multiple STAD canisters in a single overpack may have significant cost and operational advantages.
3.2.4	Vault systems will provide similar functions to storage/aging overpacks or modules, but they may contain active components for cooling.	The specification clarifies the fact that vault storage may rely on active cooling, as opposed to storage casks or modules, which must rely on passive cooling.
3.2.5	Storage and aging configurations shall have features that permit periodic monitoring and maintenance of the STAD canister for its 150-year service life. Maintenance or replacement of storage and aging systems during the STAD canister service life is acceptable.	This specification is consistent with the STAD canister design specification provided in Specification 3.1.1.2.
3.2.6	Because STAD canisters may be stored at sites with a wide range of seismological characteristics, two storage/aging overpack/module variants shall be designed assuming two standardized design earthquake (DE) ground motions described by appropriate response spectra anchored at 0.75 g and 0.25 g in lieu of the regionally- and geologically-based seismological characteristics described in 10 CFR 72.103.	See Sect. 2.3.3 for rationale.

2.7 Transportation System Specifications

There are six STAD canister transportation system specifications. Table 11 provides the rationales for these specifications.

Table 11. STAD canister transportation system specifications and rationales

STAD spec	Requirement summary	Rationale
3.5.2	The transportation overpack cavity may be designed to accommodate single or multiple STAD canisters. The use of a multicanister fixture to facilitate handling and/or transportation is acceptable.	Transporting multiple STAD canisters in a single overpack may have significant cost and operational advantages.
3.5.3	The loaded transportation overpack (without impact limiters) shall be capable of being lifted in a vertical orientation by an overhead crane.	This specification is consistent with common industry cask and overpack designs and handling practices, and uses trunnions incorporated into the overpack design. A previous repository design (DOE 2008a) was also based on this approach.
3.5.4	The loaded transportation overpack (without impact limiters) shall be able to stand upright when set down upon a flat horizontal surface without requiring the use of auxiliary supports.	Requiring the transportation overpack to stand upright in a vertical orientation simplifies operations and preparations for STAD canister loading and unloading. It is consistent with common industry cask and overpack designs and handling practices.
3.5.5	The transportation overpack shall be designed such that removable surface contamination on an accessible external surface shall be less than 1,000 dpm/100 cm^2 beta-gamma and 20 dpm/100 cm^2 alpha.	These limits are taken from the guidance in IE Circular 81-07 (Appendix A) and are based on the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. Only loose contamination is controlled, as fixed contamination will not result from the loading process.
3.6.1	The transportation skid shall be designed to permit the loaded transportation overpack, without impact limiters, to be upended and lifted vertically from the transportation skid via overhead crane.	Handling a vertically oriented loaded overpack using an overhead crane will facilitate interface of the STAD canister system with existing ISFSIs and future ISF and geologic repository handling facilities.

Table 11. STAD canister transportation system specifications and rationales (continued)

STAD spec	Requirement summary	Rationale
3.6.2	The transportation skid shall facilitate lifting of the loaded transportation overpack (including the impact limiters), in a horizontal orientation, and transfer of the transportation overpack from one conveyance to another. The attachment of the transportation skid to the railcar shall be in accordance with the requirements of AAR Interchange Rule 88, A.16.c(3) (AAR 2008).	The transportation skid is the interface between the transportation overpack and the railcar. The skid permits transfer of the loaded package in its transportation configuration, including the impact limiters, from one conveyance to another, which may be needed if the transportation cask is transported via multiple methods (heavy-haul truck, barge, and rail) en route to its destination. The skid may also be used if conveyance repair is needed during transport.

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3. GLOSSARY

This section provides definitions and descriptions of major terms of art used throughout this document.

Aging - Safely placing commercial SNF in a storage overpack/module/vault on an aging pad to allow the SNF to cool via radioactive decay. Safe aging of SNF is a prerequisite for transportation and geologic disposal to ensure that the SNF has sufficiently decayed (cooled) to meet licensed thermal limits for transportation and for repository emplacement.

Burnup - A measure of nuclear reactor fuel consumption expressed either as the percentage of fuel atoms that have undergone fission or as the amount of energy produced per initial unit weight of fuel.

Canister - A structure enclosing one or more SNF assemblies that facilitates handling, storage, aging, transportation, and disposal.

Design bases - Information that identifies the specific functions to be performed by a structure, system, or component of a facility and the specific values or ranges of values chosen for controlling parameters as reference bounds for design. These values may be constraints derived from generally accepted state-of-the-art practices for achieving functional goals or requirements derived from analysis (based on calculation or experiments) of the effects of a postulated event under which a structure, system, or component must meet its functional goals. The values for controlling parameters for external events include:

- estimates of severe natural events to be used for deriving design bases that are based on consideration of physical data, historical data on the associated parameters, or analysis of upper limits of the physical processes involved, and
- estimates of severe external human-induced events to be used for deriving design bases that are based on analysis of human activity in the region, taking into account the site characteristics and the risks associated with the event.

Dual Purpose Canister (DPC) – A canister designed to contain and confine bare SNF, and to be placed in storage and transportation overpacks.

Event sequence (repository) - A series of actions and/or occurrences within the natural and engineered components of a geologic repository that could lead to exposure of individuals to radiation. An event sequence includes one or more initiating events and associated combinations of repository system component failures, including those produced by the action or inaction of operating personnel.

Fuel assembly - A number of fuel rods held together by plates and separated by spacers, to allow coolant to flow over the rods in a reactor. This assembly is sometimes called a fuel bundle or fuel element.

Hazardous wastes (under the Resource Conservation and Recovery Act, RCRA) - According to the EPA website, hazardous waste has properties that make it dangerous or potentially harmful to human health or the environment. In regulatory terms, RCRA hazardous wastes fall into two categories:

- Listed wastes appearing on one of the four EPA hazardous wastes lists:
  - The F-list (nonspecific source wastes), 40 CFR 261.31.
  - The K-list (source-specific wastes), 40 CFR 261.32.
  - The P-list and the U-list (discarded commercial chemical products), 40 CFR 261.33.
- Characteristic wastes exhibiting one or more of four characteristics defined in 40 CFR Part 261 Subpart C:
  - Ignitability, 40 CFR 261.21
  - Corrosivity, 40 CFR 261.22
  - Reactivity, 40 CFR 261.23
  - Toxicity, 40 CFR 261.24

High-level radioactive waste (HLW) - (1) The highly radioactive material resulting from reprocessing of SNF, including liquid waste produced directly in reprocessing and any solid material derived from such liquid waste that contains fission products in sufficient concentrations; and (2) other highly radioactive material that the NRC, consistent with existing law, determines by rule requires permanent isolation.

Hypothetical accident conditions (HAC) - The sequential conditions and tests defined in 10 CFR Part 71, Subpart E, "Package Approval Standards," and Subpart F, "Package, Special Form and LSA-III Tests," against which a package or array of packages must be evaluated.

Important to safety - The phrase "structures, systems, and components important to safety" refers to those features of the ISFSI, ISF, loaded SNF storage overpack, loaded SNF transportation cask, and repository waste package, whose functions are to

- maintain the conditions required to store, transport, and dispose of SNF or reactor-related greater than class C (GTCC) low-level radioactive waste safely,
- prevent damage to the SNF or reactor-related GTCC waste container during handling, storage, transportation, and disposal, and
- provide reasonable assurance that SNF or reactor-related GTCC waste can be received, handled, packaged, stored, retrieved, transported, and disposed without undue risk to the health and safety of the public.

Important to waste isolation - Regarding design of the engineered barrier system and characterization of natural barriers, the phrase "important to waste isolation" refers to the engineered and natural barriers providing a reasonable expectation that HLW can be disposed of without exceeding post-closure performance requirements.

Neutron absorber - A material (e.g., boron) that absorbs neutrons and is used for criticality control.

Normal conditions of transport (NCT) - The conditions and tests defined in 10 CFR Part 71, Subpart E, "Package Approval Standards," and Subpart F, "Package, Special Form and LSA-III Tests," that all packages must be evaluated against during normal situations.

Overpack - The outer container component for storage, transportation, or disposal. The overpack can contain a single STAD canister or multiple STAD canisters in a carrier or basket.

Postclosure - The period of time after closure of the geologic repository.

Preclosure - The period of time before and during closure of the geologic repository.

Shielded transfer cask (STC) - A cask that meets applicable 10 CFR Part 72 requirements for safe transfer of a STAD canister and its contents within various surface facilities.

Site transporter - A system to transport loaded or empty STCs within a storage site (ISFSI or ISF), licensed in accordance with 10 CFR Part 72. The site transporter may also be capable of moving the loaded transportation overpack (with or without the transportation skid). Alternatively, the applicant may design separate transporters for the STC and transportation overpack.

Spent nuclear fuel (SNF) - Fuel that has been withdrawn from a nuclear reactor following irradiation, the constituent elements of which have not been separated by reprocessing. 10 CFR 71.4 and 72.2 further define "spent nuclear fuel" as fuel that has been discharged from the reactor for at least one year.

Storage - For the purposes of this specification, the placement of SNF in an ISFSI licensed in accordance with 10 CFR Part 72.

Standard transportation, aging, and disposal (STAD) system - the set of components consisting of one or more STAD canisters, storage/aging overpacks/modules/vaults, shielded transfer casks, site transporters, transportation overpacks, transportation skids, and ancillary equipment used to facilitate handling of SNF.

Trunnion - Cylindrical protuberance for supporting and/or lifting located on the outside of a container or cask (e.g., waste package, aging overpack, or transportation cask).

Undamaged SNF - SNF that can meet all fuel-specific and system-related functions. Undamaged fuel may be breached and may have assembly defects (per SFST-ISG-1, Revision 2).

Waste package - The waste form and any containers (disposal overpack), shielding, packing, and other materials immediately surrounding an individual waste container.

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4. REFERENCES

The citation sequences in this section are by author except where the reference designator is a more logical locator (e.g., CFR, NUREG, and regulatory guide references). Many of these references are not cited in this document but are cited within references that this document cites. Some reference citations are annotated to indicate their status or usage.

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Appendix A. IE Circular No. 81-07, Control of Radioactively Contaminated Material

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IEC 81-07

UNITED STATES
NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

May 14, 1981

IE Circular No. 81-07: CONTROL OF RADIOACTIVELY CONTAMINATED MATERIAL

Description of Circumstances:

Information Notice No. 80-22 described events at nuclear power reactor facilities regarding the release of radioactive contamination to unrestricted areas by trash disposal and sale of scrap material. These releases to unrestricted areas were caused in each case by a breakdown of the contamination control program including inadequate survey techniques, untrained personnel performing surveys, and inappropriate material release limits.

The problems that were described in IE Information Notice No. 80-22 can be corrected by implementing an effective contamination control program through appropriate administrative controls and survey techniques. However, the recurring problems associated with minute levels of contamination have indicated that specific guidance is needed by NRC nuclear power reactor licensees for evaluating potential radioactive contamination and determining appropriate methods of control. This circular provides guidance on the control of radioactive contamination. Because of the limitations of the technical analysis supporting this guidance, this circular is applicable only to nuclear power reactor facilities.

Discussion:

During routine operations, items (e.g., tools and equipment) and materials (e.g., scrap material, paper products, and trash) have the potential of becoming slightly contaminated. Analytical capabilities are available to distinguish very low levels of radioactive contamination from the natural background levels of radioactivity. However, these capabilities are often very elaborate, costly, and time consuming making their use impractical (and unnecessary) for routine operations. Therefore, guidance is needed to establish operational detection levels below which the probability of any remaining, undetected contamination is negligible and can be disregarded when considering the practicality of detecting and controlling such potential contamination and the associated negligible radiation doses to the public. In other words, guidance is needed which will provide reasonable assurance that contaminated materials are properly controlled and disposed of while at the same time providing a practical method for the uncontrolled release of materials from the restricted area. These levels and detection capabilities must be set considering these factors: 1) the practicality of conducting a contamination survey, 2) the potential of leaving minute levels of contamination undetected; and, 3) the potential radiation doses to individuals of the public resulting from potential release of any undetected, uncontrolled contamination.

Studies performed by Sommers[1] have concluded that for discrete particle low-level contamination, about 5000 dpm of beta activity is the minimum level of activity that can be routinely detected under a surface contamination control program using direct survey methods. The indirect method of contamination monitoring (smear survey) provides a method of evaluating removable (loose, surface) contamination at levels below which can be detected by the direct survey method. For smears of a 100cm^2 area (a de facto industry standard), the corresponding detection capability with a thin window detector and a fixed sample geometry is on the order of 1000 dpm (i.e., 1000 dpm/100 cm^2). Therefore, taking into consideration the practicality of conducting surface contamination surveys; contamination control limits should not be set below 5000 dpm/100 cm^2 total and 1000 dpm/ 100 cm^2 removable. The ability to detect minute, discrete particle contamination depends on the activity level, background, instrument time constant, and survey scan speed. A copy of Sommers studies is attached which provides useful guidance on establishing a contamination survey program.

Based on the studies of residual radioactivity limits for decommissioning (NUREG-0613[2] and NUREG-0707[3]), it can be concluded that surfaces uniformly contaminated at levels of 5000 dpm/ 100cm^2 (beta-gamma activity from nuclear power reactors) would result in potential doses that total less than 5 mrem/yr. Therefore, it can be concluded that for the potentially undetected contamination of discrete items and materials at levels below 5000 dpm/100cm^2, the potential dose to any individual will be significantly less than 5mrem/yr even if the accumulation of numerous items contaminated at this level is considered.

Guidance:

Items and material should not be removed from the restricted area until they have been surveyed or evaluated for potential radioactive contamination by a qualified* individual. Personal effects (e.g., notebooks and flash lights) which are hand carried need not be subjected to the qualified individual survey or evaluation, but these items should be subjected to the same survey requirements as the individual possessing the items. Contaminated or radioactive items and materials must be controlled, contained, handled, used, and transferred in accordance with applicable regulations.

The contamination monitoring using portable survey instruments or laboratory measurements should be performed with instrumentation and techniques (survey scanning speed, counting times, background radiation levels) necessary to detect 5000 dpm/100 cm^2 total and 1000 dpm/100 cm^2 removable beta/gamma contamination. Instruments should be calibrated with radiation sources having consistent energy spectrum and instrument response with the radionuclides being measured. If alpha contamination is suspected appropriate surveys and/or laboratory measurements capable of detecting 100 dpm/100 cm^2 fixed and 20 dpm/100 cm^2 removable alpha activity should be performed.

*A qualified individual is defined as a person meeting the radiation protection technician qualifications of Regulatory Guide 1.8, Rev. 1, which endorses ANSI N18.1, 1971.

In evaluating the radioactivity on inaccessible surfaces (e.g., pipes, drain lines, and duct work), measurements at other appropriate access points may be used for evaluating contamination provided the contamination levels at the accessible locations can be demonstrated to be representative of the potential contamination at the inaccessible surfaces. Otherwise, the material should not be released for unrestricted use.

Draft ANSI Standard 13.12[4] provides useful guidance for evaluating radioactive contamination and should be considered when establishing a contamination control and radiation survey program.

No written response to this circular is required. If you have any questions regarding this matter, please contact this office.

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Appendix B. Corrosion Rates and Mechanisms for Borated and Nonborated Stainless Steels

This appendix provides an overview of the available experimental results on the modes and rates of corrosion for borated stainless steel along with a relative comparison to the rates measured for nonborated analogs under conditions representative of geologic disposal.

Applicable Environmental Conditions

Environmental factors such as aqueous chemistry, redox potential, and microbial activity strongly influence the corrosion rates of stainless steel alloys. In particular, the corrosion mechanism, rate, and corrosion products are controlled by the availability of dissolved oxygen. As a rule, general and localized corrosion proceed faster in oxic conditions, where cathodic reactions on metallic surfaces are dominated by the reduction of dissolved oxygen (Rebak 2011). Most geologic repository media are expected to return to anoxic or anaerobic conditions shortly after backfilling and closure. This is particularly true for repositories in a low-permeability water-saturated host medium that contains reducing minerals (e.g., pyrite) or natural organic matter (Bryan et al. 2011). In the absence of oxygen, water acts as an electron acceptor for metallic iron, and the cathodic reaction is controlled by hydrogen evolution (Rebak 2011). The ubiquity of water as a possible electron acceptor is one reason that steels continue to corrode under anoxic aqueous conditions. While steel corrosion is generally anticipated to be slower under anoxic conditions, environmental factors such as sulfide concentration or microbial activity may result in rapid corrosion even under anoxic conditions (Jack and Wilmott 2011).

Corrosion Rates of Nonborated Stainless Steels 304/304L and 316/316L under Oxic Conditions

Uniform Corrosion – Uniform corrosion of stainless steel under oxic alkaline conditions and the water chemistry typical of a clay repository (no added chloride) vary from 0.03 μm yr^{-1} at 30ºC to 0.5 μm yr^{-1} at 80ºC (Kursten et al. 2004). Immersion tests (documented in NUREG/CR-5598) were conducted on artificially creviced samples of 304L in fresh water, J-13 water, simulated J-13 water, J-13 water with crushed tuff, and simulated concentrated waters at 90°C, as well as the vapor above these liquids. Driven corrosion tests in aerated simulated J-13 well water at 90°C resulted in a corrosion rate of 0.02–0.14 μm yr^{-1} in liquid and 0.96–2.95 μm yr^{-1} in vapor (NUREG/CR-5598). In the concentrated J-13 water (Solution No. 20), the corrosion rates were 0-0.150 μm yr^{-1}, and in vapor over Solution No. 20, the corrosion rates were 0.03-1.25 μm yr^{-1} (NUREG/CR-5598). The effect of added hydrogen peroxide (electrolysis product) on 304L stainless steel corrosion was tested, with measured rates of 0.04-6.58 μm yr^{-1} (NUREG/CR-5598). The corrosion rates of 304L reported by McCright et al. (1987) after a yearlong exposure to J-13 tuff water, performed under both irradiated and non-irradiated conditions at room temperature, are 0.151 μm yr^{-1} (irradiated) and 0.285 μm yr^{-1} (non-irradiated). Corrosion rates summarized in BSC 2004a are 0.001–1.57 μm yr^{-1} in freshwater at ambient to boiling temperature, 1.588–39.147 μm yr^{-1} in saltwater at 26.7°C, and 0.660-15.900 μm yr^{-1} in saltwater at 90°C. The uniform corrosion rates for nonborated stainless steels are summarized in Table B-1.

Localized Corrosion – The presence of chloride poses one of the strongest chemical controls on whether pitting will take place. Experimental testing indicated no pitting of 316L in alkaline solutions containing up to 100 g L^{-1} chloride at room temperature of 21ºC (Kursten et al. 2004). When the concentration of chloride is 50 g L^{-1}, a critical pitting temperature of 45ºC was reported (Kursten et al. 2004). Similar chloride concentration threshold behavior was observed for pitting of 304L: pitting was observed at 60ºC with >50 g L^{-1} chloride (Kursten et al. 2004). Crevice corrosion of 304L was observed at 80ºC and background chloride concentrations of 20 g L^{-1} or greater; and no crevice corrosion was observed at 40ºC and chloride concentrations up to 20 g L^{-1} (Kursten et al., 2004). Pit initiation testing for 304/304L and

316/316L indicates that pitting progression varies: in particular, the oxidative history of the sample is associated with a large difference in the number of pits for the 316/316L alloys, but not for 304/304L. NUREG/CR-5598 shows that in relatively aggressive aqueous media at 90°C, creviced samples have corrosion rates of 0.03 μm yr^{-1} in vapor and 0.29-0.43 μm yr^{-1} in liquid, with significant pitting and pit depths of 15–62 μm after 2,855 hours of testing.

Stress corrosion cracking is observed in an unstressed sample of 304L stainless steel after aging in cementitious material containing 100 g L^{-1} chloride for 2 years (Kursten et al. 2004). Additional testing in alkaline solutions indicated that increased chloride (17.7 g L^{-1}) and thiosulfate ($S_2O_3^{2-}$ at 3.4 g L^{-1}) increased both pitting and stress corrosion cracking of the 316L and 304L alloys (Kursten et al. 2004). In aggressive environments (e.g., 45% MgCl or 26% NaCl), stress corrosion cracking is observed to take place within hours to days. Cracking is observed in less than 3 hours in the magnesium chloride solution at 155°C, and after 48 to 72 hours in the sodium chloride tests at 102 and 200°C (Streicher and Grubb 2011).

Corrosion Rates of Nonborated Stainless Steels 304/304L and 316/316L under Anoxic Conditions

Uniform Corrosion – Significantly less experimental data are available on the corrosion of stainless steel under anoxic conditions. Uniform corrosion rates of stainless steel measured under anoxic conditions and the water chemistry typical of a clay repository (no added chloride) are from 0.001 μm yr^{-1} to 0.1 μm yr^{-1} for both tested temperatures of 30ºC and 80ºC (Kursten et al. 2004). The uniform corrosion rates for nonborated stainless steels are summarized in Table B-1.

Localized Corrosion – Localized corrosion under anoxic conditions includes hydrogen embrittlement, sulfide stress cracking (Elboujdaini 2011), and microbially assisted corrosion. Hydrogen gas is produced during anoxic corrosion of stainless steel. Hydrogen embrittlement, induced blistering, and cracking may occur due to the evolution of atomic hydrogen at the surface, followed by the diffusion of atomic hydrogen into steel (Elboujdaini 2011). Sridhar et al. (1991) state that in a typical repository setting, hydrogen embrittlement is most likely minor in comparison to other corrosion modes; however, the rate of hydrogen evolution and diffusion must be determined on a case-by-case basis. Sulfide stress cracking is a variety of hydrogen-induced cracking and is usually localized in weld zones (Elboujdaini 2011). It can occur in mildly corrosive media at temperatures below 90°C. Several studies summarized in Ilgen et al. (2014b) report that microbially assisted corrosion may significantly shorten the lifetime of a stainless steel component.

Corrosion of Borated Stainless Steel under Oxic Conditions

Uniform Corrosion – Lister et al. (2007) performed electrochemical corrosion testing of borated stainless steel alloys and measured a uniform corrosion rate of 0.0176–0.0371 μm yr^{-1} at 60°C in an aerated simulated in-package solution. The authors state that the results are indicative of short-term corrosion rates and that a longer-term test would be required for a comprehensive prediction of borated stainless steel behavior in a waste package.

The uniform corrosion rate of Neutronit (a steel alloy with neutron absorbers) is assumed to be similar to the corrosion rate of stainless steel Type 321 (BSC 2004a), which varies as a function of water composition (freshwater vs. saltwater) and temperature. The measured values are 0.001–0.011 μm yr^{-1} in freshwater at 29.5°C, 0.025-0.33 μm yr^{-1} in freshwater at 50°C to 100°C, and 1.81–29.22 μm yr^{-1} in saltwater at 26.7°C (BSC, 2004a). The uniform corrosion rates for borated stainless steels are summarized in Table B-2.

Localized Corrosion – Fix et al. (2004) measured weight loss in simulated concentrated groundwater at 90°C where samples were immersed for more than 5 years. The weight loss was mostly caused by localized corrosion, so general corrosion rates were not calculated. He (2008) reports that in chloride solutions, borated stainless steel is susceptible to localized corrosion, and general corrosion rates vary from tens of nm yr^{-1} to μm yr^{-1} depending on the test environment and duration. He et al. (2012) performed tests in water vapor and under immersed (simulated groundwater) conditions for Types 304B4 and 304B5 borated stainless steel at 60, 75, and 90°C. Pitting corrosion was observed when specimens were exposed to humid air at 75 and 90°C, and no pitting occurred at a temperature of 60°C (He et al. 2012).

Borated alloy (UNS S30466, same as borated 304) is susceptible to pitting in the presence of sulfide and chloride ions (Moreno et al. 2004). Another electrochemical study also reports pitting corrosion of borated stainless steel (with 1.2% B) in 0.5 M sodium chloride solution; pitting decreased with increasing degree of solution-annealing (Upadhyay et al. 2014).

Corrosion of Borated Stainless Steel under Anoxic Conditions

As with the nonborated stainless steels, there are significantly less data available on the corrosion of borated stainless steel under anoxic conditions. Lister et al. (2008) measured the corrosion rate of three borated stainless steel 304B alloys with varying boron content at 60°C. Samples were creviced, and potentiostatic tests were performed in anoxic conditions and in an acidic environment. The tests were performed over the course of 7 days under a nitrogen gas purge. The measured corrosion rate increased with increasing boron content. The average corrosion rates were 0.221±0.070 μm yr^{-1} for 304B4 (1.17 % B), 0.427±0.132 μm yr^{-1} for 304B5 (1.32 % B), and 0.464±0.100 μm yr^{-1} for 304B6 (1.69 % B) (Lister et al. 2008). The uniform corrosion rates for borated stainless steels are summarized in Table B-2.

Comparison of Borated vs. Nonborated Stainless Steel Corrosion Rates

A summary of the literature values presented above for general corrosion rates of nonborated and borated stainless steels is shown in **Error! Reference source not found.** and Table B-2. There is a wide range of corrosion rates over a differing range of temperatures and chemistries for the borated and nonborated stainless steels. One comparable environment is the anoxic environment, which shows higher corrosion rates for borated stainless steel at 60°C (0.221-0.464 μm yr^{-1}) than for nonborated stainless steel at 80°C (0.100 μm yr^{-1}).

Fix et al. (2004), He (2008), SNL (2007b), and NUREG-1949 indicate that the corrosion rates of borated stainless steels are higher than for nonborated Types 304 and 316 stainless steels. Fix et al. (2004) measured weight loss in simulated concentrated groundwater at 90°C where samples were immersed for more than 5 years. The results indicate that borated alloys are less resistant to general corrosion or to localized attack. The borated stainless steel had weight loss 3 to 10 times higher than the nonborated materials, and the weight loss was mostly caused by localized corrosion. The SNL (2007b) report summarizes linear polarization resistance (LPR) analyses on borated stainless steel Types 304B4 and 304B5, with average reported corrosion rates of 0.0073–0.253 μm yr^{-1} (LPR) and 0.0423–0.0956 μm yr^{-1} (gravimetric tests). These rates were measured under relatively nonaggressive conditions (starting pH 5.5 to 7, low ionic strength, 60°C) and are 3.5 to 5.5 times greater than the rates measured in analogous experiments for nonborated Type 304L stainless steel.

A variety of tests summarized in the BSC report (2004a) indicate that corrosion of borated Type 304 stainless steel is faster than in the nonborated counterpart. The rate depends on the amount of boron in the alloy (0.3 and 1.5 % alloys were tested), temperature, and aqueous matrix composition. The corrosion rates for nonborated Types 302/304/304L stainless steel vary from 0.001-1.570 μm yr^{-1} in freshwater at temperatures from room temperature up to 100°C, to 1.588-39.174 μm yr^{-1} for saltwater at 26.7°C. The values for borated stainless steel under similar testing conditions ranged from 3.05-12.19 μm yr^{-1} (for 0.3% B) in freshwater at ambient temperature, to 38.22-147.0 μm yr^{-1} at 50°C in "harsh" water. With increasing boron content (1.5% B), the corrosion rates were 161.54-252.98 μm yr^{-1} in freshwater at ambient to boiling temperature and 164.64–1,058.4 μm yr^{-1} in "harsh" water at ambient to boiling temperature.

Table B-1. Uniform corrosion rates of nonborated stainless steels

Minimum µm/year	Maximum µm/year	Temp °C	Medium	Oxic / anoxic	Reference
n/a	0.030	30	Alkaline media	Oxic	Kursten et al. (2004), Table 8
n/a	0.500	80	Alkaline media	Oxic	Kursten et al. (2004), Table 8
0.001[a]	1.570[a]	25-100	Freshwater	Oxic	BSC (2004a), Table 7-1
1.588[a]	39.147[a]	26.7	Saltwater	Oxic	BSC (2004a), Table 7-1
0.660[a]	15,900[a]	90	Saltwater	Oxic	BSC (2004a), Table 7-1
0.020	0.140	90	Aerated simulated J-13 water	Oxic	NUREG/CR-5598, Table 4.2
0.960	2.950	90	Vapor over aerated simulated J-13 water	Oxic	NUREG/CR-5598, Table 4.2
0	0.150	90	Aerated simulated concentrated J-13 water (Solution No. 20),	Oxic	NUREG/CR-5598, Table 5.4
0.030	1.250	90	Vapor over aerated simulated concentrated J-13 water (Solution No. 20),	Oxic	NUREG/CR-5598, Table 5.4
0.040	6.580	90	Aerated simulated concentrated J-13 water (Solution No. 20), with added hydrogen peroxide (radiolysis product)	Oxic	NUREG/CR-5598,Table 5.5
0.151 (irradiated)	0.285 (non-irradiated)	Room temp.	J-13 water, both irradiated and non-irradiated	Oxic	McCright et al. (1987)
0.001	0.100	30	Alkaline media	Anoxic	Kursten et al. (2004), Table 8
0.001	0.100	80	Alkaline media	Anoxic	Kursten et al. (2004), Table 8

[a]SS Types 302/304/304L

Table B-2. Uniform corrosion rates of borated stainless steels

Minimum µm/year	Maximum µm/year	Temp °C	Medium	Oxic / anoxic	Reference
0.0176	0.0371	60	Aerated simulated in-package water	Oxic	Lister et al. (2007), Table 5
0.001[a]	0.011[a]	29.5	Freshwater	Oxic	BSC (2004a), Table 7-1
0.025[a]	0.330[a]	50–100	Freshwater	Oxic	BSC (2004a), Table 7-1
1.810[a]	29.220[a]	26.7	Saltwater	Oxic	BSC (2004a), Table 7-1
3.050 (0.3% B)	12.190 (0.3% B)	Ambient	"Fresh" water (as defined by the reference author)	Oxic	BSC (2004a), Table 7-1
38.220 (0.3% B)	147.000 (0.3% B)	50	"Harsh" water (as defined by the reference author)	Oxic	BSC (2004a), Table 7-1
161.540 (1.5% B)	252.980 (1.5% B)	25–100	"Fresh" water	Oxic	BSC (2004a), Table 7-1
164.64 (1.5% B)	1,058.400 (1.5% B)	25–100	"Harsh" water	Oxic	BSC (2004a), Table 7-1
0.073	0.253	60	Nonaggressive	Oxic	SNL (2007b), Table 4-20
0.221 (1.17% B)	0.427 (1.32% B), 0.464 (1.69% B)	60	Simulated in-package water (acidic, under a nitrogen gas purge)	Anoxic	Lister et al. (2008), p. 6

[a]Aqueous corrosion rates of Neutronit (using SS Type 321 surrogate).

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Appendix C. Review of the Use of Borated Stainless Steel in Existing Designs in the United States

This appendix reviews current uses of borated stainless steel as a neutron absorber in the nuclear industry. Neutron absorber materials currently in use consist mostly of aluminum-based materials that fall into two groups:

1. aluminum alloys or metal matrix composites containing boron as a neutron absorber, usually in the form of a metal boride, and
2. ceramic-metal material (cermet), produced by mixing powdered Al and carbon boride together, placing it between sheets of aluminum, and rolling at high temperatures to produce a sandwich containing the sintered Al-metal-carbon boride mixture.

Borated stainless steel has had limited use as a neutron absorber in dry storage canisters and casks (Greene et al. 2013). Borated stainless steel has both advantages and disadvantages relative to Al-based materials. Because possible boron loadings are relatively low (<2.5 wt%), thicker borated stainless plates are required to achieve the required mass loadings per unit area of boron, adding to the volume and weight of the absorber plates. However, borated stainless steel has a much higher strength and lower corrodibility than Al materials. A list of storage systems using borated SS304 as of August 2013 is given in Greene et al. (2013); a summary is provided below.

Canister Systems

FuelSolutions™ W74M and W74T Canisters **–** The FuelSolutions™ W74M and W74T canisters are 64- BWR-SNF-assembly multipurpose canisters certified for storage and transportation (Greene et al. 2013). Borated SS304 absorber plates line each cell in the basket and are held in position by welded stainless steel retainers that insert into holes in the sheets to hold them into position (Energy*Solutions* Spent Fuel Division Inc. 2007). The neutron absorber plates are nonstructural members. Seven W74 canisters are currently in use, all at the Big Rock Point Nuclear Plant.

AREVA TN NUHOMS® 52B Canister **–** The NUHOMS® 52B is a 52-BWR-SNF-assembly canister certified for storage only. It has a carbon steel basket with borated SS304 absorber plates. Twenty-seven canisters are currently in use at the Susquehanna Nuclear Power Plant (Greene et al. 2013).

Cask Systems (Bare SNF)

CASTOR V/21 Cask **–** The CASTOR V/21 cask is a 24-PWR-SNF-assembly storage cask for bare SNF. The fuel basket is constructed of welded SS304 with integral plates of borated stainless steel (Greene et al. 2013). Twenty-five CASTOR V/21 casks are currently in use at the Surry Nuclear Power Station, and one is at the Idaho National Laboratory.

TN-24 Cask **–** The TN-24 and TN-24P storage casks were designed for the storage and transportation of spent nuclear fuel (NRC, 1993), but they are only certified in the US for storage (Greene et al. 2013). The TN-24 had a capacity of 24 PWR SNF assemblies and was designed to have a basket made of copper-plated borated SS304. There are no TN-24 casks in service, and it is not clear how the basket was to be assembled. The TN-24P differed from the TN-24 in several ways, including the use of aluminum-based neutron absorber plates. A single cask of the TN-24P design is in use at Idaho National Laboratory.

TN-BRP Storage and Transport Cask **–** The TN-BRP cask is a storage and transport cask designed to accommodate 85 BWR SNF assemblies. It was designed for one-time use, to transport 85 BWR SNF assemblies that were used at the Consumers Power Big Rock Point Plant from the DOE West Valley Demonstration Project to the Idaho National Laboratory. Only one cask was built, and the transport

Certificate of Compliance (71-9202) expired immediately upon completion of transport. The fuel basket of the cask was constructed of borated 304 stainless steel; manufacturing details are not available.

***TAD* –** The TAD canister, developed as part of a previously submitted US repository license application, was designed for delivery of SNF to the repository site, onsite aging, and eventual disposal. The TAD was designed to use borated stainless steel plates as the neutron absorber materials for criticality control (DOE 2008a). As part of their safety evaluation of the previously submitted repository license application, the NRC reviewed existing data on the properties and corrosion behavior of borated SS304 and concluded that the data are appropriate for use in the TAD canister (NUREG-1949).

Appendix D. Outline of a Conceptual Experimental Testing Plan for Additional Corrosion Studies on Borated Stainless Steel

Limited experimental data exist on the corrosion of borated stainless steel under anoxic conditions. This appendix presents a conceptual plan for additional experimental testing which would be necessary to better predict the lifetime of a borated stainless steel component in a repository. The experimental plan is based on an experimental plan that has been proposed for austenitic stainless steels 304/304L and 316/316L used for the canister shell (Ilgen et al. 2014a).

Experimental work is needed to identify the modes of corrosion and to measure uniform corrosion rates for the borated stainless steel components in case of an early breach (penetration and flooding with groundwater) of both the disposal overpack and the stainless steel canister. The experimental testing program should account for significant differences in geochemistry between different disposal environments (e.g., salt, crystalline, and argillaceous), and it should address the evolution of corrosion damage in these different geochemical settings.

The tests should be designed to evaluate corrosion behavior for the container materials (borated stainless steel neutron absorbers) during the postclosure period. Following closure and backfill, the conditions are expected to be oxic for several years. Once all residual oxygen is consumed, the conditions will be anoxic in a low-permeability host media with backfill/buffer materials for the remainder of the repository performance period. Therefore, understanding the anaerobic corrosion rates and mechanisms for the materials of interest is important.

Materials Selected for Testing

The test plan includes all eight borated stainless steel alloy types specified under ASTM-A887-89. These alloys should be Grade A alloys. Specification 3.1.5.4 excludes Grade B alloys, which have larger and less evenly dispersed boride grains than Grade A alloys. The exact geometry and thermal history of these samples (e.g., thicknesses, presence of welds, and mechanically stressed zones) should be selected to represent the components within the standardized canister. Welding has been shown to cause significant impact on ductility, which can be restored to some degree if the welds are mitigated. However, there are no experimental results testing the corrosion of both as-welded and annealed welded materials. Tests of welded materials should be given a priority because they are likely to be subject to localized corrosion.

Phenomena to be Examined by the Experimental Testing Program

The experimental testing program should be designed to address the following:

- general corrosion rates under anoxic conditions for borated stainless steels;
- hydrogen embrittlement of the borated stainless steel components under diffusion-controlled conditions;
- localized corrosion of the borated stainless steel under geochemical conditions representative of the salt, shale and granite repositories, and the extent to which localized corrosion may affect structural integrity of the component; and
- geochemical controls on the evolution of the passive film on the borated stainless steel under anoxic conditions.

Geochemical Systems and Geochemical Variables Selected for Testing

The geochemical conditions should be selected to include several water compositions likely for each disposal concept (salt, granite, and clay/shale). These conditions are expected to depend on the failure scenario. For example, the composition of water entering the canister depends on whether the bentonite buffer is breached and whether the groundwater has equilibrated with the bentonite buffer material. Groundwater compositions for laboratory testing are shown in Table D-1 (from Ilgen et al. 2014a).

The mode and rate of corrosion is controlled by moisture, pH, temperature, the presence of oxidizing species, and the concentrations of chloride and sulfide ions. Therefore, these variables should be tested. Because borated stainless steel is a passive metal, special attention should be given to the performance of the passive Cr_2O_3 and NiO oxide layers under very reducing conditions in the presence of typical groundwater ions. Hydrogen embrittlement may be of concern because corrosion of iron-based alloys under anoxic conditions produces hydrogen. If the diffusion of hydrogen from the corroding surface is slow, hydrogen embrittlement of the surface may result and further enhance corrosion.

Table D-1. Proposed representative groundwater chemical compositions

	Constituent	Shale[a]		Granite[b]	
		Shale-1	Shale-2	Granite-1	Granite-2
(mg L^{-1})	TDS[c]	50,990	249,150	53,480	250,360
	Ca^{2+}	2,044	12,983	5,450	63,800
	Na^{+}	16,635	80,430	10,100	18,500
	Mg^{2+}	624.66	2,689	5,260	24
	K^{+}	215.11		57.6	371
	Cl^{-}	30,349	152,817	32,143	166,200
	Br^{-}			244	1,200
	SO_4^{2-}	996.97	207.97	<1	265
	HCO_3^{-}	340.25	24.13	54	0

[a]Blondes et al. (2014).
[b]Frape et al. (2003).
[c]TDS = total dissolved solids.

Corrosion rates of borated stainless steel should be evaluated under both saturated (activity of water ~1) and unsaturated (only water vapor present) conditions to represent different failure scenarios. Therefore, corrosion of borated stainless steel can be evaluated at a constant temperature, while varying the activity of water, the aqueous matrix composition (in particular, chloride and sulfide [or sulfate, if appropriate] concentrations), and also testing how the presence of water-saturated bentonite buffer affects the corrosion rates and corrosion products of borated stainless steel. Both general and localized corrosion should be assessed.

Appendix E. Welding-Induced Alteration of Borated Stainless Steel and Methods for Weld Mitigation

The CASTOR V/21 cask at Idaho National Laboratory was loaded in 1985 and opened in 1999 to assess the effects of long-term interim storage on the SNF and cask internals (NUREG/CR-6745). During the 1999 inspection, it was determined that 15 of 16 examined stitch welds—welds that attached the borated SS304 neutron absorber plates to the SS304 basket structure—had cracked. However, these weld locations had not been examined prior to loading in 1985, and it was determined that the cracking probably occurred due to differential thermal expansion during testing prior to cask loading. Other more accessible basket welds that were examined in 1985 had shown cracking at that time.

The weld cracks illustrate one flaw of borated SS304: welding decreases the ductility of the material. This change in material performance is attributed to redissolution of the borides in the weld zone and formation of a dendritic austenite/boride eutectic (Robino and Cieslak 1995). Some researchers (Martin 1989) have suggested that the eutectic may also form in the heat-affected zone and that boride grains will agglomerate there, but that has been disputed (Robino and Cieslak 1995). The decrease in ductility is greatest at the HAZ-weld fusion line, and it is here that cracking tends to occur. This is consistent with experimental work showing that welding causes reduced ductility in borated SS304 (Robino and Cieslak 1997). This effect is less important for Grade A borated SS304 than for Grade B, which has larger and less evenly dispersed boride grains than Grade A. Note that because of the reduced ductility in borated stainless steel weld zones, the NRC has objected to the use of ASTM 887 borated stainless steel as a structural component for spent fuel storage racks (EPRI 2005).

The weldability of borated stainless steel is similar to the traditional austenitic steels (Robino and Cieslak 1997) and can be achieved using a variety of welding techniques. Fusion welding (e.g., tungsten arc [GTA] and electron beam welding) damage can be mitigated to some degree by the post-weld heat treatment. Several studies have shown that the ASME code's minimum required impact toughness can be achieved after annealing of the welded zone (EPRI 1994b, Park et al. 1997, Robino and Cieslak 1997).

Robino and Cieslak (1997) determined that for fusion-welded 304B4 Grade A alloy (1.16 wt.% B), post-weld heat treatment requires 28,500 hours (at 700°C), 170 hours (at 900°C), 24 hours (at 1000°C), or 1.05 hours (at 1200°C) for the weld damage to be mitigated to near code-acceptable levels. Park et al. (1997) investigated GTA welds of AISI 304-B3 stainless steel plates for a range of post-weld heat treatments (700°C, 800°C, 900°C, 1000°C, or 1100°C for 1 hr; or 1100°C or 1200°C for times between 1 and 7 hr). A variety of tests (bending tests, elongation tests, and Charpy impact tests) were in agreement that the higher temperature of 1200°C is necessary to recover the mechanical properties of the heat-affected zones.

These studies indicate that post-weld heat treatment at a temperature as high as 1200°C is necessary for the welds to have ductility matching that of the base metal. Additional research is needed to optimize the post-weld heat treatment procedures.

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Appendix F. Stress Corrosion Cracking of Spent Nuclear Fuel Interim Storage Canisters

Following initial cooling in pools, SNF is transferred to dry storage casks for longer-term storage at the reactor sites. The storage cask systems are commonly welded stainless steel (Hanson et al. 2012) containers enclosed in ventilated concrete or steel overpacks. These cask systems are intended as interim storage until a permanent disposal site is developed, and until recently, they were licensed for up to 20 years with renewals also up to 20 years. In 2011, 10 CFR 72.42(a) was modified to allow for initial license periods of up to 40 years and license extensions of up to 40 years. However, the United States does not currently have a disposal pathway for SNF, and these containers may be required to perform their waste isolation function for many decades beyond their original design criteria. Recent studies by the Nuclear Waste Technical Review Board (NWTRB 2010), the Electric Power Research Institute (EPRI 2011), the DOE Used Fuel Disposition Program (Hanson et al. 2012), and the Nuclear Regulatory Commission (NRC 2012) have identified and prioritized potential concerns with respect to the safety performance of long-term interim storage. In each of these studies, the potential for canister failure by chloride-induced stress corrosion cracking (CISCC) was identified as the major concern with respect to canister performance.

Criteria for Stress Corrosion Cracking

Stress corrosion cracking (SCC) is a localized corrosion phenomenon by which a through-wall crack could potentially form in a canister outer wall over time intervals shorter than the dry storage service lifetime. In order for SCC to occur, three criteria must be met (Fig. F-1): the metal must be susceptible to SCC, an aggressive environment must exist, and sufficient tensile stress must be present to support SCC. In general, these criteria could be met during the period of interim storage, at least at some ISFSI sites, especially if the development of a repository for final disposal is delayed. SCC of interim storage canisters has never been observed; however, that may be largely because detailed canister surface inspections for SCC have not been performed. Access to the canister surfaces through vents in the overpacks is extremely limited, and high surface radiation fields make removal of the canisters from the overpacks undesirable. Efforts are currently in progress to develop the technologies to reliably detect SCC on in-service canisters.

Aggressive Environment

The environment at any given location on the storage canister surface will be aggressive if (1) aqueous conditions exist and (2) a corrosive chemical species is present. The canister overpack protects the canister from direct rainfall. Water may enter through the ventilation openings and be blown or dripped onto the package. Evidence of this was seen during the recent canister surface inspections (Bryan and Enos 2014, EPRI 2014). Any advective flow of water onto the packages is likely to be transient, and because the storage canisters are hot relative to outside temperatures, water will rapidly evaporate. Hence, persistent aqueous conditions are only anticipated to occur by deliquescence of salts in dust on the surface. For most dry cask storage systems, passive ventilation is utilized to cool the casks within the overpacks, and large volumes of outside air are drawn through the system. Dust and aerosols within the air are deposited on the steel canisters, and as the canisters cool over time, salts in the dust will deliquesce to form brine on the storage container surface. Deliquescence will occur when the relative humidity (RH) at the canister surface reaches a limiting value (RH_L) for corrosion; this value is generally somewhat lower than the deliquescence RH (RH_D) for the deposited salt assemblage.

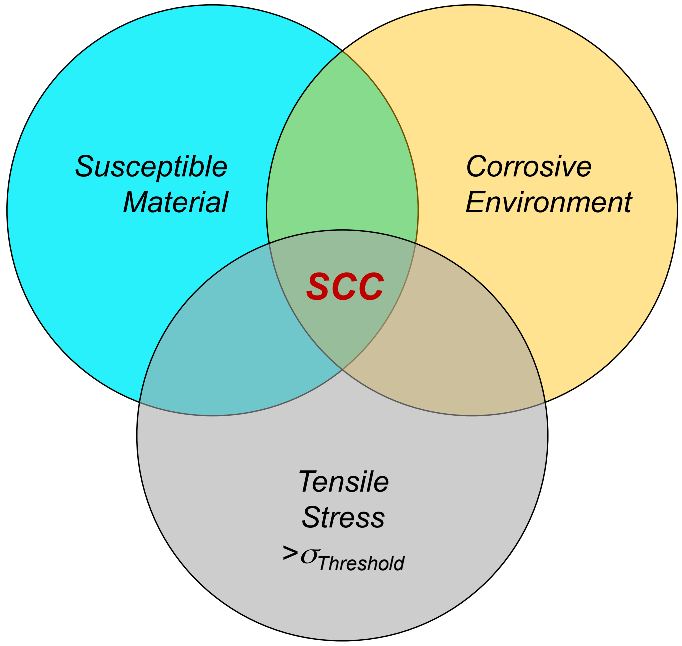


Fig. F-1. Criteria for SCC initiation and growth.

The RH at the canister surface is controlled by the surface temperature at any given location, and the water content, or absolute humidity (AH), of the ambient air entering the overpack (Fig. F-2). For typical conditions, it is anticipated that corrosion will not be possible until local canister surface temperatures drop below 60–70ºC. This does not indicate that newly loaded canisters are safe from corrosion, however. Passive cooling by air advection through the overpacks is extremely effective and creates large temperature gradients on canister surfaces. For instance, surface temperatures measured on Diablo Canyon canisters containing high-burnup fuel only 2–4 years into storage were as low as 50ºC near the inlets at the base of the canisters, although the temperature at the canister top was 150ºC (Bryan and Enos 2014). Hence, even for hot, recently loaded canisters, parts of the canister surface rapidly cooled sufficiently to undergo deliquescence, and potentially SCC.

Although other aggressive species may be present (e.g., high atmospheric concentrations of SO_2), the species considered to be most aggressive for SCC is chloride. Many ISFSIs are located in coastal areas where chloride-rich sea-salt aerosols may be deposited on the canisters. These deliquesce to form chloride-rich brines, and SCC is a well-documented mode of attack for austenitic stainless steels (including SS304 and SS316) in marine environments (Kain 1990). Recent canister surface inspections (Bryan and Enos 2014, EPRI 2014) have confirmed that chloride salts are present on the surface of in-service SNF storage canisters in near-marine settings (Fig. F-3).

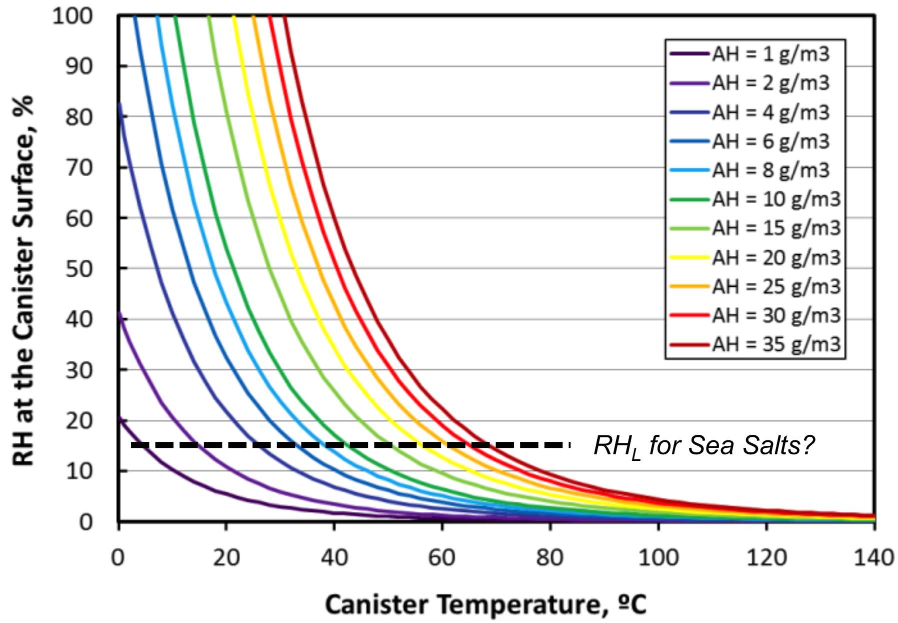


Fig. F-2. Relationship between canister surface temperature, relative humidity, and RH at the canister surface.

A third potential criterion for a corrosive environment may be the amount of chloride present. Some studies have shown that there may be a lower limit on the amount of chloride on the package that can support SCC initiation. For instance, Shirai et al. (2011) determined experimentally that SCC could not initiate on SS304 under conditions nominally representing atmospheric corrosion at chloride loads <0.3 g/m^2. However, other work suggests that if there is a lower limit of chloride loading for CISCC, it is less than that value. Albores-Silva et al. (2011) observed SCC at chloride loadings of 0.1 g/m^2, and experimental studies by the NRC (NUREG/CR-7170) showed that SCC could occur at sea salt loadings as low as 0.1 g/m^2 (0.056 g/m^2 chloride). Other studies (Tokiwai et al. 1985, Taylor 1994, Fairweather et al. 2008) have shown that SCC corrosion may occur at loadings as low as 0.005 to 0.02 g/m^2. The United Kingdom Nuclear Decommissioning Authority has issued cautious operational limits for chloride surface concentrations on 316L waste packages of 0.01 g/m^2 for temperatures between 10 and 30°C and 0.001 g/m^2 for temperatures between 30 and 50°C (NDA, 2012). If there is a threshold chloride limit for SCC initiation, then it is apparently sufficiently low that it cannot be effectively used as a screening criterion for SCC.

It is likely that the rate and/or persistence of SCC growth is a function of the surface salt load, which affects the current carrying capacity of the brine layer and the ability of the cathode (outside of the crack) to support corrosion at the anode (within the crack). This approach has been proposed for estimating maximum pitting penetration depths in several recent papers (e.g., Chen and Kelly 2010, Krouse et al. 2014, Woldemedhin and Kelly 2014), but it has not been rigorously applied to SCC.

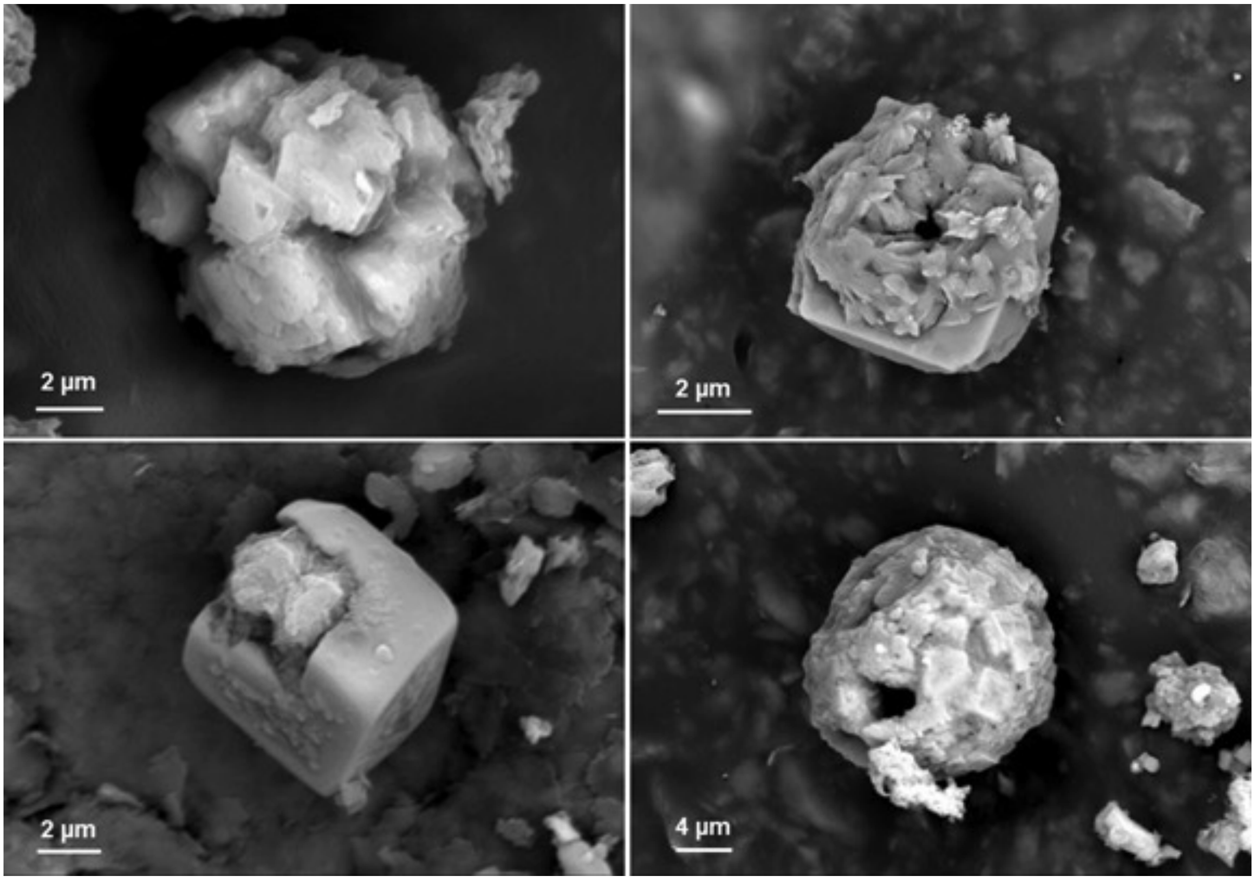


Fig. F-3. Aggregates of sea-salts ($NaCl + MgSO_4$) collected from the surface of an in-service SNF storage canister at Diablo Canyon.

Although the potential for SCC is considered to be highest in near-marine environments, the very low threshold for SCC initiation indicates that inland sites cannot be considered immune. Chloride-rich salt aerosols may be generated by use of brackish water in cooling towers or by salting of nearby roads during bad weather. The NRC applies a generic interim storage environmental impact statement (EIS) for all ISFSI sites and does not accept that the risk of canister failure by SCC may be lower at inland sites. In defense of their generic interim storage EIS, the NRC has concluded that "…the impacts of continued storage will not vary significantly across sites; the impacts of continued storage at reactor sites, or at away-from-reactor sites, can be analyzed generically" (NRC 2014).

Material Susceptibility

The welded interim storage canisters are made of austenitic stainless steels, including 304/304L and 316/316L. These alloys are known to be susceptible to SCC in aggressive environments if sufficient tensile stresses are present. SCC of 304/316 stainless steel has been observed in near-marine ambient temperature field tests and industrial sites (Kain 1990, Hayashibara et al. 2008, Kosaki 2008, Cook et al. 2011, Nakayama and Sakakibara 2013, Cook et al. 2014). In elevated-temperature experimental tests with deliquesced sea-salts that were meant to replicate conditions on the surface of an SNF interim dry storage canister, SCC has been shown to occur readily in both base metal and weld specimens (e.g., Nakayama 2006, Prosek et al. 2009, Tani et al. 2009, Mintz et al. 2012, Prosek et al. 2014).

Although even base metal can undergo SCC in the presence of sufficient stress, there are several factors that can increase material susceptibility. These include the degree of sensitization, the degree of cold working, the presence of iron contamination on the metal surface, and the surface finish (Parrott and Pitts 2011).

Degree of Sensitization – When austenitic stainless steel is welded, the weld metal is melted and homogenized. However, in the heat-affected zone (HAZ) near the weld, the steel becomes sensitized. When the metal is heated during the welding process, Cr diffuses from the metal grains into the grain boundaries, where it combines with carbon to form chromium carbides. Sensitization results in the formation of chromium-depleted zones at grain boundaries that facilitate the nucleation and propagation of localized corrosion such as pitting (often a precursor for SCC) and SCC. In general, the degree of sensitization induced by welding increases with the thickness of the welded material because multiple weld passes are required and the heat input is greater. Increasing degrees of sensitization correspond to shorter incubation times prior to pitting and SCC initiation, formation of more pits and cracks, and more rapid pit and crack growth. Nakayama and Sakakibara (2013) estimate that the SCC initiation lifetime can decrease by more than an order of magnitude as the degree of sensitization increases from 0 to 20%, and crack growth rates can increase by a factor of 5,for atmospheric SCC conditions. Khatak et al. (1996) also saw increases in crack growth rates for sensitized SS304 and noted that sensitization significantly lowered the threshold tensile stress for SCC, although this was for immersed conditions. The degree of sensitization has not been measured in representative storage canister welds, but it is likely that sensitization occurs because of the metal thickness (⅝") and the multiple passes that are used to make the weld. Sandia National Laboratories (SNL) is currently in the process of procuring a full-diameter mockup of a storage canister made using the materials, weld schedules, and procedures used in a NUHOMS storage canister. The mockup will provide prototypical welds that will be characterized with respect to weld residual stresses and degree of sensitization. This information will not be available until late 2015, however.

It should be noted that 304L stainless steel contains less carbon than 304 stainless steel, so it is much less susceptible to sensitization. Existing in-service storage canisters are made of either 304 or 304L; however, as steel fabrication techniques have improved, almost all modern 304 stainless steel is dual-certified, meaning that it not only meets the carbon content threshold for 304 (0.08 % maximum), but also the lower threshold for 304L (0.03 % maximum). Hence, sensitization is less of a factor for new canisters than for canisters made previously.

Degree of Cold Working – Cold working affects corrosion resistance of stainless steels (Khatak et al. 1996, García et al. 2001, Parrott and Pitts 2011) for two reasons. First, it results in the formation of strain-induced martensite in the metal that is less resistant to corrosion than austenite. Second, it induces local stresses and increases defect density; the dissolution rate of the metal is increased by the increased strain energy. The degree of cold working required to roll flat plates into cylindrical storage canister shells is not expected to significantly affect the corrosion resistance of the metal.

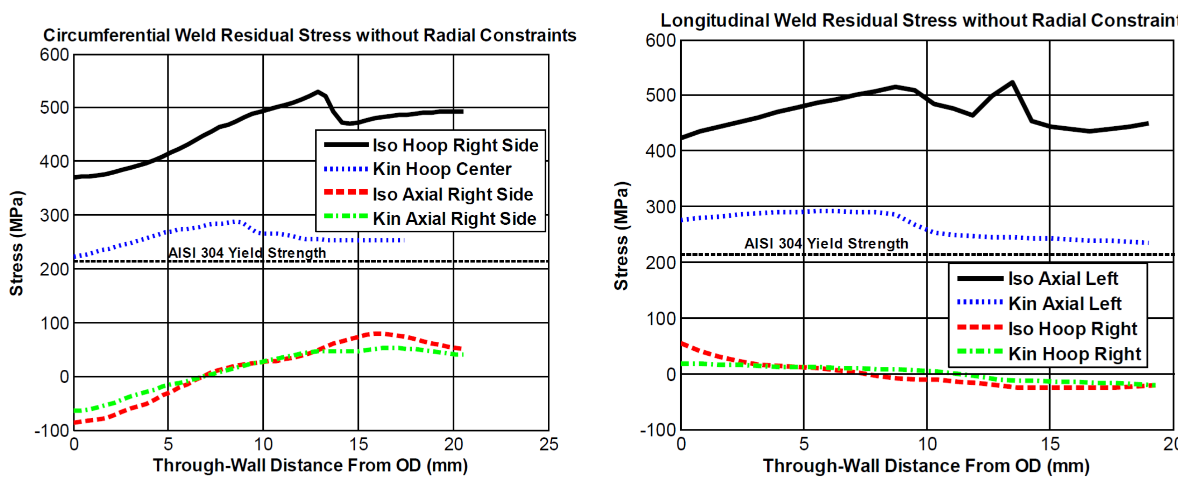
Iron Contamination – Contamination of the stainless steel surface with less corrosion-resistant forms of iron (e.g., tool steel or iron from support rails) will increase the likelihood of SCC because the iron particles corrode more readily, supporting the development and stability of corrosive solutions (increased chloride concentrations and lowered pH) in pits and in SCC. It has been suggested that instances of SCC at temperatures below 60ºC is in many cases due to iron contamination on the stainless steel surface (Parrott and Pitts 2011). This may be very important in some overpack designs; during the canister surface inspection at Calvert Cliffs, corrosion spots were observed on the canister surface which were attributed to scratches and iron contamination from the rails (EPRI 2014).

Surface Finish – A rough surface finish (>1 µm) can promote initiation of corrosion, apparently by trapping water and chloride ions on the surface (Parrott and Pitts 2011). Also, surface grinding can produce large local variations in stress that may contribute by increasing strain energy and the dissolution rate of the metal. All storage canisters have rougher surfaces than 1 µm.

Crevice Corrosion – As with other factors that promote the development of the corrosive low-pH, high-chloride environment, the presence of crevices and crevice corrosion promotes initiation of SCC (Parrott and Pitts 2011). There are many potential crevice locations in storage systems, with perhaps the most important being the contact between the canister and the rail in horizontal storage systems, and contact between the canister and guide rails in vertical systems.

Tensile Stresses

In order for a SCC to form and grow, tensile stress must occur. Residual stresses are imparted into the storage canister during manufacturing (rolling the steel plate to form a cylinder) and by welding. Although the residual stresses from cold working have never been measured, it is anticipated that they will be too low to support stress corrosion cracking. However, residual tensile stresses related to the welding process are likely sufficient to support SCC and may extend through the entire thickness of the shell. There have been no direct measurements of residual stresses associated with typical SNF dry storage canister welds; however, weld residual stress modeling for typical canister welds has been done by the NRC (2013). The through-wall stress profiles that the NRC predicted for circumferential and longitudinal canister welds are shown in Fig. F-4. Results are shown for both isotropic and kinematic hardening laws; the two profiles are expected to bound the actual stresses present in the weld regions. In the direction parallel to each weld, stresses throughout the wall thickness are tensile and greater than or equal to the yield strength of the metal. These tensile stresses are more than sufficient to support SCC, which would form perpendicular to the direction of highest tensile stress, cutting across the weld region at right angles.



Left: circumferential weld Right: longitudinal weld

Fig. F-4. Predicted weld residual stress profiles in canister weld regions (NRC 2013).

SCC Initiation

As discussed above, two of the three criteria required for stress corrosion cracking to occur—a susceptible material, and the necessary tensile stress—are likely to be met by all storage canisters. It is less clear that a corrosive environment will exist at all storage sites, both inland and near-marine. However, the NRC does not currently accept environment as a potential mitigating factor with respect to SCC. Hence, it must be assumed that a corrosive environment will be present and that SCC can and will initiate at all ISFSI sites. The time interval between SNF emplacement and SCC initiation—the SCC incubation time—is not known. As discussed previously, temperatures vary widely over the surfaces of canisters in storage overpacks, and even for canisters with high-burnup fuel, some fraction of the canister surface will be cool enough for deliquescence within several years of placement into storage. Once deliquescent brine develops, localized corrosion (pitting) will initiate. The pits grow over time, and once they reach a sufficient depth (generally around 70–100 µm, but this is a function of the tensile stresses present), they serve as initiation loci for SCC. Experimental studies have shown that SCC commonly originates at corrosion pits (e.g., Kondo 1989, Turnbull and Zhou 2004, Nakayama 2006, Turnbull et al. 2006b, Turnbull et al. 2006a, Kosaki 2008, Prosek et al. 2009, Albores-Silva et al. 2011, Shirai et al. 2011). Pit growth rates are poorly understood; at ambient temperatures, pit growth may be quite slow (e.g., Chen and Kelly 2010), and SCC initiation may take several years. However, experimental testing at even moderately elevated temperatures (35–60ºC), such as those expected on the canister surface for conditions of deliquescence, have resulted in pitting and SCC initiation within days to months (e.g., Prosek et al. 2009, NUREG/CR-7170). Therefore, it must be assumed that SCC initiation times will be short relative to the 150-year design lifetime of the standardized canister.

SCC Growth

In order for SCC to be a concern, crack growth rates must be sufficiently rapid to result in penetration of the canister wall, 0.5 to 0.625 inches thick, within the designated design lifetime of 150 years. Stress corrosion crack growth rate is a function of many parameters, including temperature, magnitude of tensile stress, material properties such as yield strength and degree of sensitization, and environmental parameters such as chloride concentration (a function of RH), chloride surface load, and brine pH. A commonly used form for the crack growth rate is provided below. It includes the effect of tensile stress in the form of the crack tip stress intensity factor (K), and temperature; environmental and material properties parameters are included implicitly by using relevant test data to develop the crack growth amplitude value (α_{crack}). A power law dependence is assumed for K, while an Arrhenius relationship is assumed for the temperature dependence (Wu and Modarres 2012):

$$\frac{\mathrm{dx_{crack}}}{\mathrm{dt}} = \alpha_{\mathrm{crack}} \cdot \exp\left[-\frac{\mathrm{Q}}{\mathrm{R}}\left(\frac{1}{\mathrm{T}} - \frac{1}{\mathrm{T_{ref}}}\right)\right] \cdot (\mathrm{K} - \mathrm{K_{th}})^{\beta_{\mathrm{crack}}}$$

where:

dx_{crack}/dt is the crack growth rate,
α_{crack} is the crack growth amplitude,
Q is the activation energy for crack growth,
R is the universal gas constant (8.314 J mol^{-1} K^{-1}),
T is the temperature (K) of interest,
T_{ref} is a reference temperature (K) at which α was derived,
K is the crack tip stress intensity factor,
K_{th} is the threshold stress intensity factor for SCC, and
β_{crack} is the stress intensity factor exponent.

For a cracked structure under remote or local loads, the stress intensity factor (K) is a measure of the stress field ahead of the crack. In elastic fracture mechanics, when the applied value of the stress intensity factor exceeds the material's critical value, crack advance occurs. For subcritical cracking, the process of crack advance is linked to the applied value of the stress intensity factor through curve fits that are based on extensive experimental data. The stress intensity factor K is defined as (Wu and Modarres 2012):

$$K = \sigma_{applied} Y \sqrt{\pi x_{crack}}$$

where

$\sigma_{applied}$ is the tensile stress,

Y is a shape parameter equal to 1 for an infinite flat plate,* and

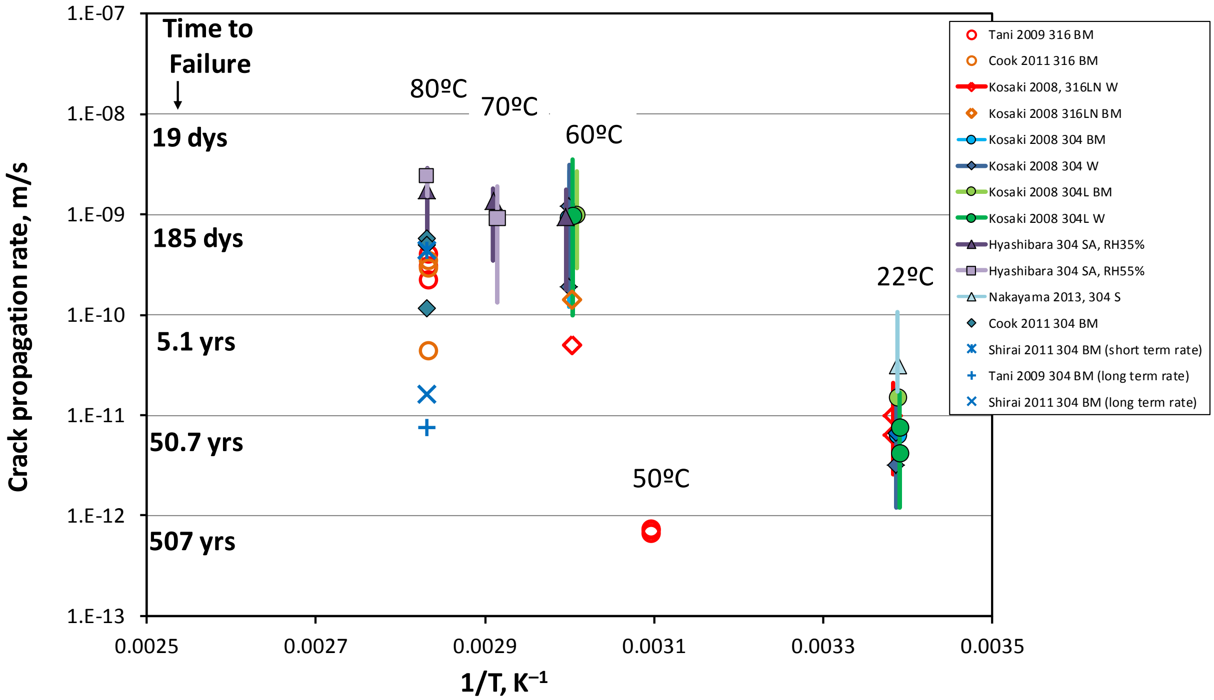
x_{crack} is the crack depth.

*Given that the waste canister circumference and length are much greater than the thickness of the canister wall and the crack depth/length at the time of penetration, this is a reasonable approximation.

Because the crack growth rate is a function of temperature, the elevated temperatures on the canister surface will result in faster crack growth rates. A summary of crack growth rate experimental data, collected for stainless steels exposed to deliquescent sea salts at a range of temperatures, is shown in Fig. F-5. The data in Fig. F-5 can be used to set Q and α_{crack} in the above equation. All of these data were collected specifically to address the issue of SNF dry storage canister corrosion. Data include rates from both 304 and 316 stainless steels, and they include base metal, weld, and sensitized samples (Hayashibara et al. 2008, Kosaki 2008, Tani et al. 2009, Cook et al. 2011, Shirai et al. 2011, Nakayama and Sakakibara 2013). There is a good deal of scatter in the measured rates, and some of the more rapid rates may not be relevant to thick metal samples (Shirai et al. 2011). However, it is apparent that, at elevated temperatures, penetration could occur within 150 years, even at the slowest rates measured. Penetration rates at ambient temperatures are much slower, but penetration is still possible within 150 years.

Summary

On the basis of the available data, the three criteria for SCC are likely to be met on SNF canister surfaces during storage at least at some sites. SCC is likely to initiate within the 150-year design lifetime of the canisters and may penetrate the canister walls. A standardized canister design must address the concerns for SCC and be designed to mitigate this risk. Possible solutions include building the canister out of materials less susceptible to SCC, such as duplex stainless steels. However, 304L and 316L are acceptable materials so long as steps are taken to reduce susceptibility to SCC by performing weld mitigation. Possible mitigation techniques include high temperature thermal annealing of the entire canister prior to loading, which would not only remove highly tensile weld residual stresses but would also eliminate the effects of sensitization by redissolving the Cr-rich carbides back into the metal. Alternatively, mitigation of weld residual stresses could be done using techniques such as shot peening, laser peening, or low plasticity burnishing. These techniques create a thin layer of metal on the surface of the treated region that has high compressive stresses, preventing initiation of SCC.



BM = base metal; W = weld sample; SA = solution annealed; S = sensitized.
Bars represent reported ranges (if more than one), while symbols represent average values.
Time-to-failure corresponds to the time required to penetrate a 0.625-inch thick canister wall.

Fig. F-5. SCC propagation rates for atmospheric corrosion of SS304 and SS316.

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Appendix G. Generic Case for Postclosure Safety of STAD Canisters

Introduction

Disposability of STAD canisters can be demonstrated with a safety case that includes screening of features, events, and processes (FEPs), and a performance assessment for comparison to regulatory postclosure dose standards. This is the basic set of analyses needed for licensing repository postclosure performance, and it has been performed twice in the US using site-specific information (DOE 1996, DOE 2008a). However, without a repository site, the STAD canister postclosure performance analyses must be generic at this time.

Fortunately, the STAD canister will be only one part of a multiple barrier disposal system so that other barriers can be relied on for waste isolation from the biosphere. A previously submitted license application (DOE 2008a) assigned no containment function to the spent fuel canister because such functions were performed by the waste form, waste package, engineered barriers, and natural barriers. The canister contributed to other types of performance such as structural integrity of the waste package.

This appendix presents a brief survey of generic performance assessment analyses for crystalline, clay/shale, and salt host media. It refers to a previous generic safety case study (Freeze et al. 2013) for model description and rationale, and for FEP screening. This survey shows how simple models could be used to establish reasonable assurance that the STAD canister would perform, along with other barriers, in a manner that meets regulatory performance standards.

As noted in Sect. 2.3.2, there is a possibility that the neutron absorber plates in STAD canisters could fail to perform their function for fewer than 10,000 years due to waste package and STAD canister breach, with subsequent exposure of the neutron absorber plates to groundwater. Key challenges for demonstrating STAD canister disposability include understanding the potential for postclosure criticality and the effects of a criticality event if one occurs. This appendix addresses criticality potential for the various host media: salt, unsaturated hard rock, and saturated crystalline or clay/shale media. The discussion includes specific proposals for the types of analysis that could be used for generic demonstration of postclosure criticality control.

Postclosure Waste Isolation

Waste isolation performance was analyzed for generic salt, crystalline, and clay/shale media (Freeze et al. 2013). The basic conceptual model of the disposal system was the same (Fig. G-1). The simplicity of the models illustrates that there are few parameters, so they can be readily adapted to site-specific analysis. For two of the three models (salt and clay/shale), no credit was taken for waste package containment (including the canister) because the natural barriers and slowly dissolving waste form could provide isolation (Figs. G-2 and G-3). For the crystalline rock model, containment was assigned to the disposal overpack and not the fuel canister, and 1% of the overpacks were assumed to fail (Fig. G-4). In each case, the calculated dose meets a 15 mrem yr^{-1} dose standard for the first 10,000 years and 100 mrem yr^{-1} after that. In contrast, for the unsaturated hard rock repository, extensive performance assessment analysis was demonstrated using site-specific information (DOE 2008a).

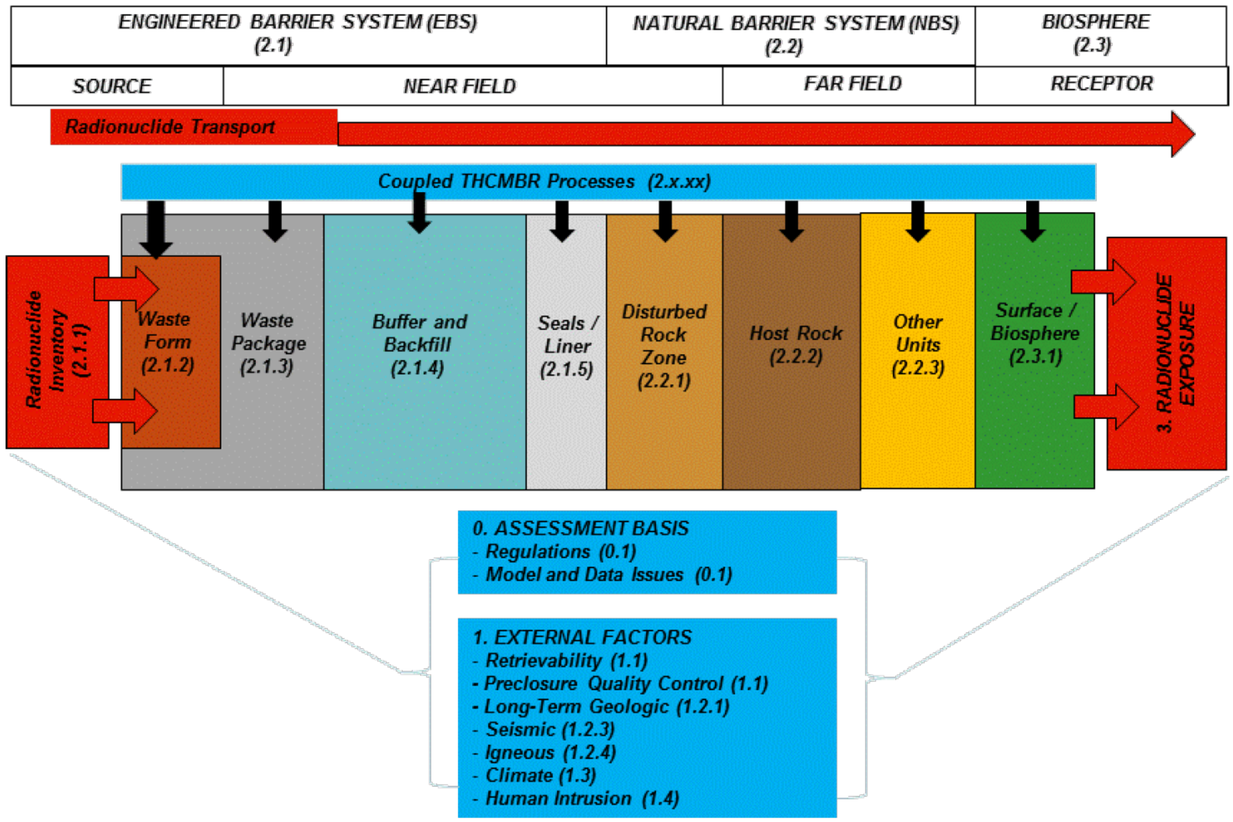

Fig. G-1. Conceptual model for generic repository waste isolation analysis (Freeze et al. 2013).

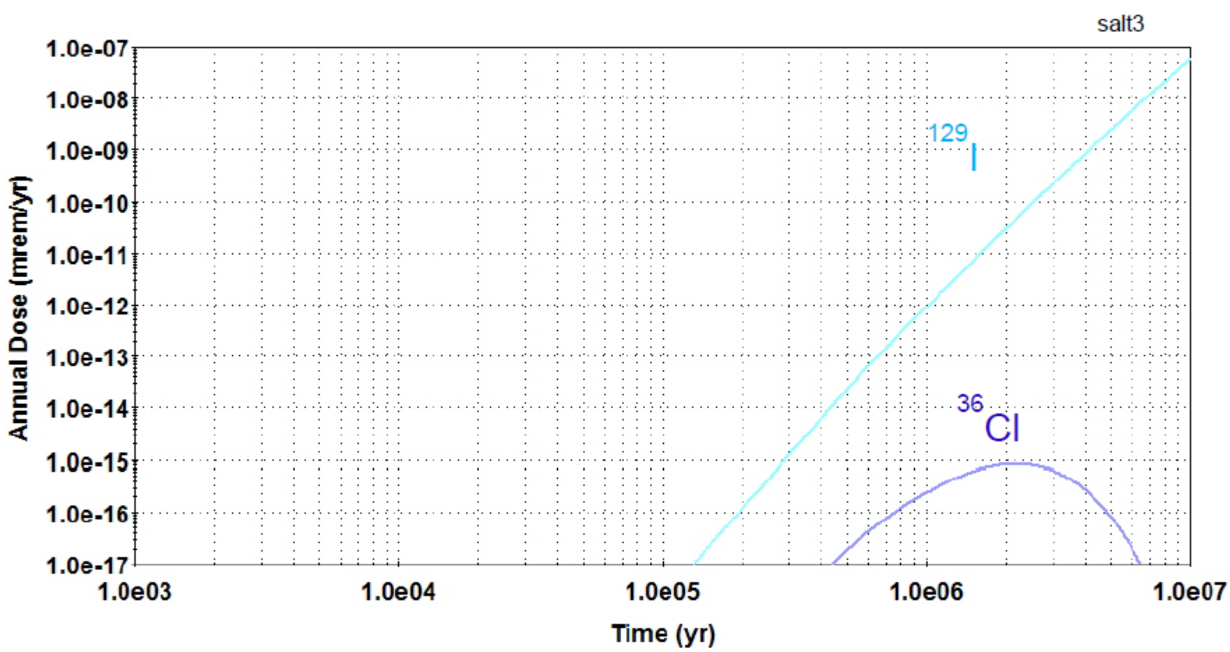

Fig. G-2. Calculated dose for a generic salt repository (Freeze et al. 2013).

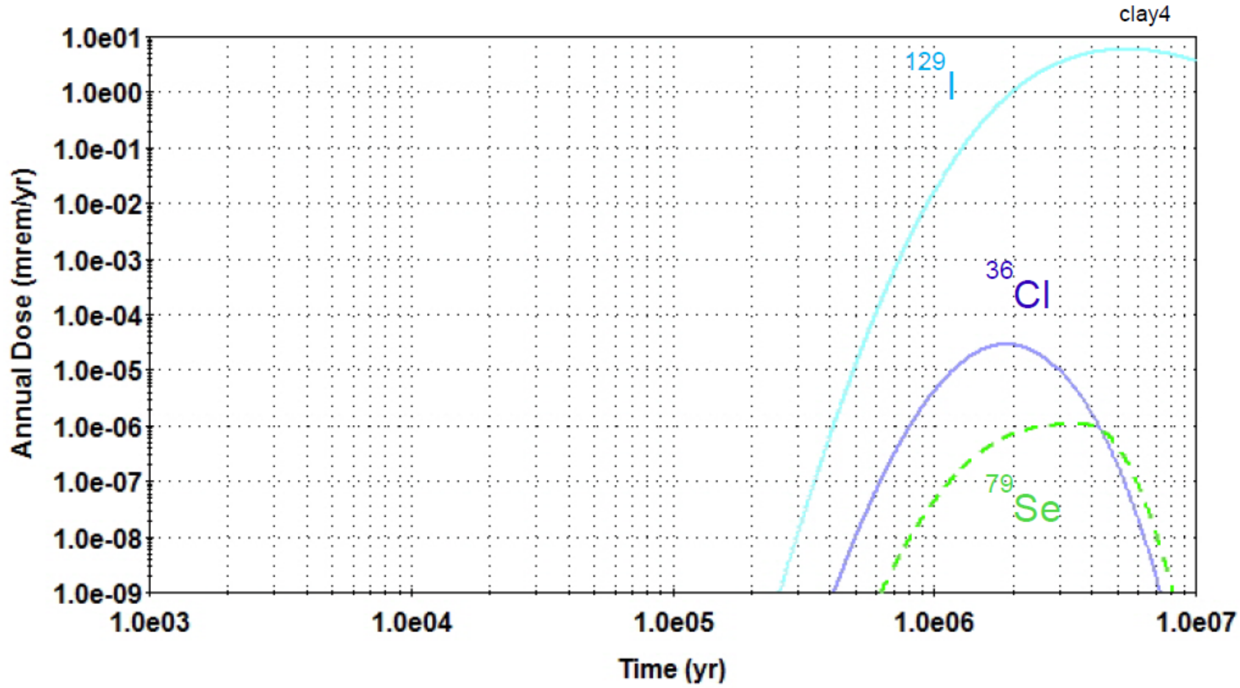

Fig. G-3. Calculated dose for a generic clay/shale repository (Freeze et al. 2013).

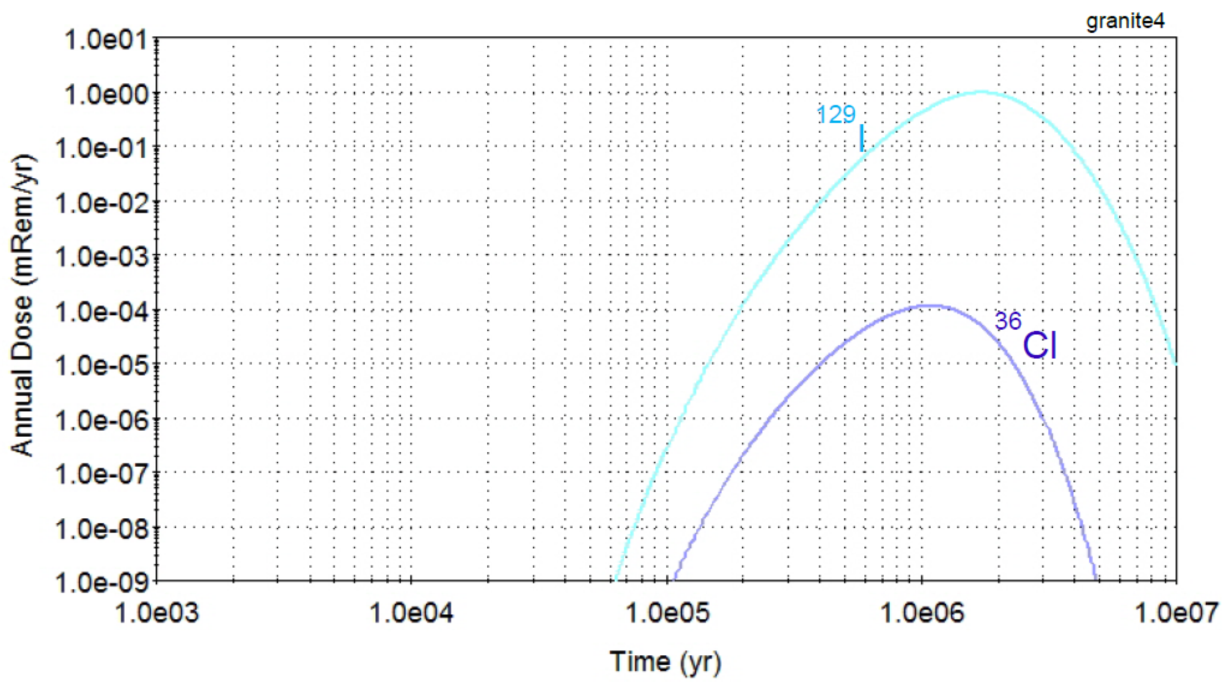

Fig. G-4. Calculated dose for a generic crystalline rock repository (Freeze et al. 2013).

Postclosure Criticality Control

For postclosure criticality to be excluded from performance assessment based on low probability, the aggregate probability must be less than 10^{-4} over 10,000 years (Table G-1). Thus the circumstances leading to waste package breach, flooding, and degradation of neutron absorbers (or degradation of basket structure) must have aggregated probability less than 10^{-4}. For disposal environments with potential to flood a breached package with freshwater (not saline or unsaturated), this analysis assumes (1) the disposal overpack is designed and manufactured to provide reasonably high-reliability containment for at least 10,000 years, and (2) the combined effects from the geologic setting and engineered barriers lower the probability of disruptive events leading to waste package failure, to less than 10^{-4} for 10,000 years. Seismicity and faulting may be the most likely disruptive events for geologic settings, and the latter assumption relies on relatively quiescent tectonics at the selected site, as well as the dampening effect of backfill. This discussion of postclosure criticality is provided because the degradation rate for borated stainless steel (recommended material for neutron absorber plates) is uncertain and could be large enough that complete degradation (leaving less than the minimum thickness to prevent criticality) is possible.

A potentially important event leading to waste package breach is early failure due to manufacturing defects. Previous analysis estimated the mean probability of an early failure condition at 4×10^{-5} for waste packages, and 4×10^{-7} for drip shields (SNL 2007a). The joint probability for early failure of a specific waste package and its drip shield is clearly less than 10^{-8}, but the probability of either type of failure was included in performance assessment (DOE 2008a). The previous analysis was thorough, and the prospect for significant reduction in early failure probability is limited (review of the analysis is currently underway and will be reported in FCRD-UFD-2015-000714 and FCRD-UFD-2015-000129). However, reduction may be possible through improvements in the way manufacturing defects are represented in performance assessment, for example, through impacts on the overpack corrosion rate rather than assigning an initial breach condition. Early failure is therefore likely to be part of any performance assessment, and for robust waste packages, it may be the most probable mode of failure in 10,000 years. In the event of waste package breach due to manufacturing defects, the event sequence possibly leading to criticality will involve other uncertainties that reduce the aggregate probability as discussed below.

Another potentially important event sequence begins with inadvertent human intrusion by drilling into a waste package. The human intrusion standard (defined for a 10,000-year stylized scenario by 10 CFR Part 63 and 40 CFR Part 197) involves larger threshold screening probabilities than individual protection standards (Table G-1); nevertheless, the effects of human intrusion on the potential for criticality must still be considered. A waste package breach caused by human intrusion could have the same long-term effect on neutron absorber materials as a breach due to manufacturing defects. Possible linkage between human intrusion and criticality is also discussed below.

Table G-1. Summary of postclosure dose standards based on 10 CFR Part 63

FEP probability (per year)	Individual protection standard 15 mrem yr^{-1} for 10,000 yr 63.311(a)(1)	Individual protection standard 100 mrem yr^{-1} after 10,000 yr 63.311(a)(2)[a]	Individual protection standard for stylized human intrusion 15 mrem yr^{-1} for 10,000 yr 63.321(b)(1)	Individual protection standard for stylized human intrusion 100 mrem yr^{-1} after 10,000 yr 63.321(b)(2)[a]	Groundwater protection standard limits on combined ^{226}Ra and ^{228}Ra activity; gross α activity; dose from combined β and photon emitting radionuclides; for 10,000 yr 63.331
$< 10^{-8}$ [b]	Not included 63.342(a)	Not included 63.342(c)	Not included 63.342(a)	Not included 63.342(c)(1)	Not included 63.342(a)
$10^{-8} < p < 10^{-5}$	Included	Included	Not included 63.342(b)	Not included 63.342(c)(1)	Not included 63.342(b)
$> 10^{-5}$	Included	Included	Included	Included	Included

[a] For these two standards, 10 CFR 63.342(c) requires the inclusion of seismic and igneous activity subject to probability limits, and it also requires inclusion of the effects of climate change (with prescribed limits on the effects of climate change), as well as inclusion of the effects of general corrosion.
[b] If the probability of a feature, event, or process (FEP) is greater than 1×10^{-8} per year, the FEP can also be excluded if its effect on repository performance (however probable) can be demonstrated to be not significant. (10 CFR 63.342[a]).

Postclosure Criticality in a Salt Repository – Certain neutronic calculations performed evaluating the feasibility of direct disposal of dual-purpose canisters (DPCs) (most of which have readily degraded, aluminum based, absorbers), are applicable to STAD canisters. In particular, a high-reactivity model was formulated to study the effect of flooding ground waters of different composition (Hardin et al. 2014). The model and results for sodium chloride brine over a range of chloride concentrations are shown in Fig. G-5. Whereas saturated NaCl at 20°C has a concentration of approximately 6 molal, or 158,000 ppm chloride, substantial reduction in neutronic reactivity is shown for concentrations half the saturation value, especially for higher burnup SNF. This results because natural chlorine has an isotopic fraction of 75% ^{35}Cl, a neutron absorber.

Under normal conditions in a salt repository, there is very little free water or brine. Disposal concepts for salt typically call for heavy waste packages fabricated from low-alloy steel, which corrodes on contact with water, reacting to form gaseous hydrogen. With little water present, waste package corrosion will be very slow, and there is little possibility for flooding even if waste package breach occurs from corrosion. Should flooding occur, naturally occurring waters in the salt formation will be brines. Waste package breach due to manufacturing defects is insignificant in this environment. Human intrusion may occur, but the drill must penetrate the robust waste package, and the drilling fluid typically used in evaporites is saturated brine (diesel-fuel–based fluids are also used and have not been evaluated). Hence, there may be little potential for criticality to occur in a salt repository.

Postclosure Criticality in an Unsaturated, Hard-Rock Repository – The STAD canister described in the performance specification is based on the transport-aging-disposal (TAD) canister (DOE 2008a). The performance of that canister in an unsaturated, hard-rock repository was analyzed extensively and reviewed by the NRC, resulting in a safety evaluation report (NUREG-1949). The review concurred that borated stainless steel, in combination with other engineered features and the unsaturated natural setting, would function as intended to prevent criticality for at least 10,000 years, in the event of early failure or other waste package breach.

Postclosure Criticality in Other Host Media – The crystalline rock and clay/shale disposal concepts call for packaging in corrosion-resistant overpacks (Hardin and Kalinina 2015). Like the Swedish KBS-3 concept (SKB 2011), clay-based material would surround and condition the corrosion environment at the waste package surface. The result could be a high-reliability containment envelope for which manufacturing defects could be minimized using modern methods of inspection and testing. As an extreme example, early failure could definitely be excluded from consideration if two or more independent, corrosion-resistant containment barriers were used (i.e., joint probability $< 10^{-8}$ per year).

In addition to a high-reliability overpack, realistic representation of other processes provides additional reduction in criticality probability (Fig. G-6). The time to breach may consume a significant portion of the 10,000-year performance period. Flooding of the STAD canister after package breach is not definite because the canister itself must also fail from corrosion, and the source of water must be sufficient to flood the canister (i.e., the water must flow through a hydrated clay backfill, and then through a small breach). Once groundwater enters a STAD canister, the borated stainless steel absorber plates must substantially corrode to allow the possibility of criticality (at corrosion rates discussed in Sect. 2.3.2). Even with substantial or complete degradation of absorber plates, as-loaded burnup analysis of fuel canisters shows that many have available uncredited reactivity margin. Other effects also come into play, such as the salinity of groundwater and the probability of a fuel misload.

The same type of argument can be made for criticality as a consequence of human intrusion (Fig. G-7). Drilling equipment used for clay/shale media is typically not configured for penetrating heavy metal containers (e.g., 5 cm wall thickness). Even if penetration occurs and the STAD canister fills with drilling fluid or groundwater, the absorber plates may corrode slowly, and the as-loaded configuration of the canister may be subcritical.

Note that Figures G-6 and G-7 are illustrative examples of events that may potentially be considered that could reduce the overall probability of criticality.

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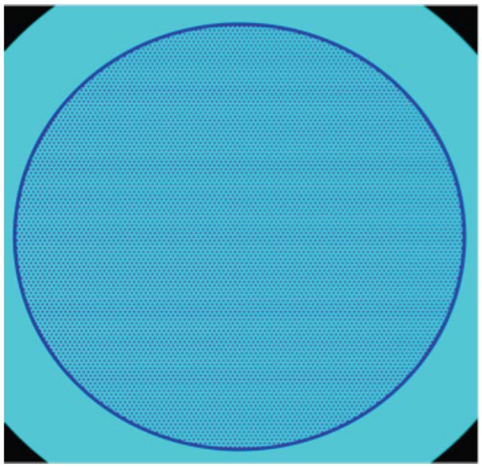
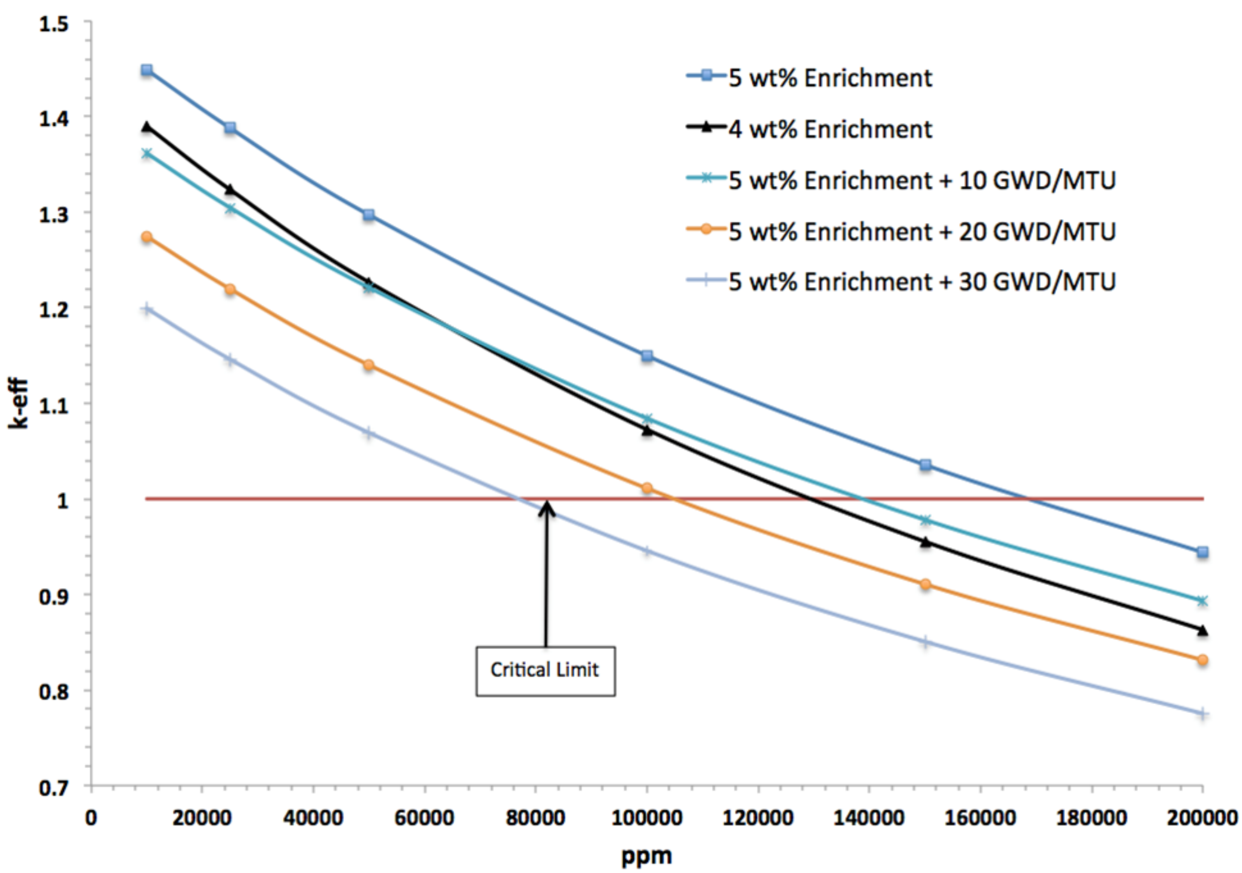


High-reactivity model with PWR fuel rods uniformly distributed, and no basket structure or neutron absorbers.



Note: Concentration of 158,000 ppm chloride corresponds to a saturated salt brine (6 molal) at 20°C.

Fig. G-5. High-reactivity model geometry (upper) and neutron multiplication factor (k_{eff}) as a function of chloride concentration, for different fuel loadings (lower).

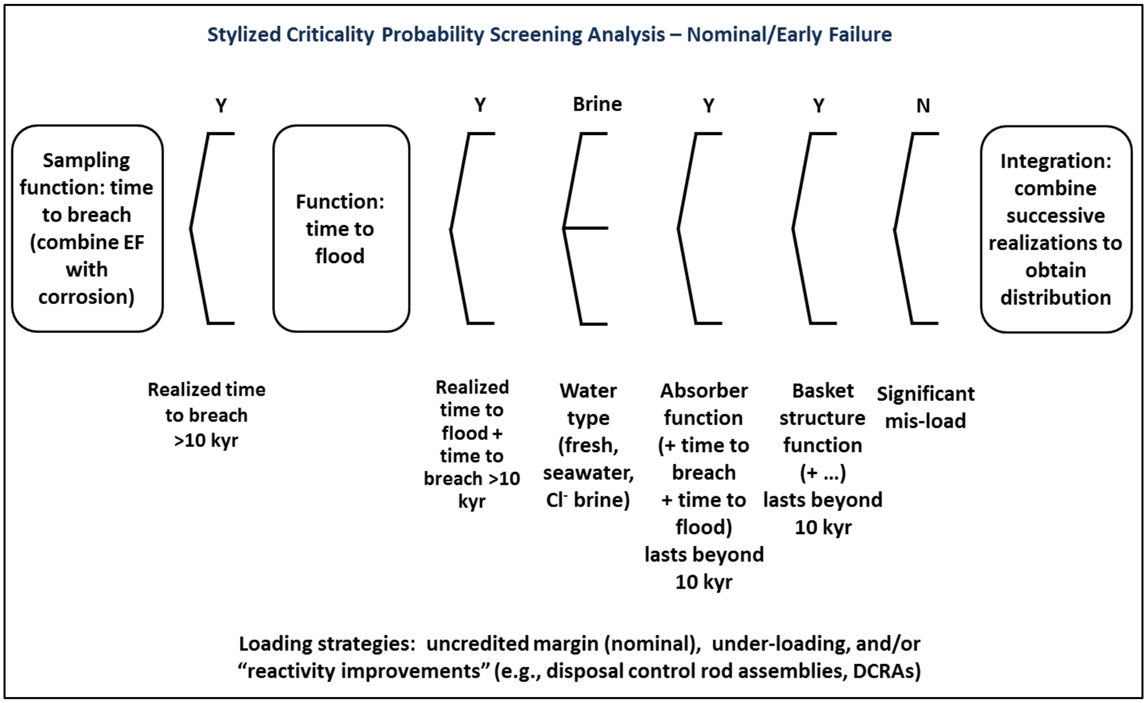


Fig. G-6. Event-tree logic for a stylized criticality screening analysis.

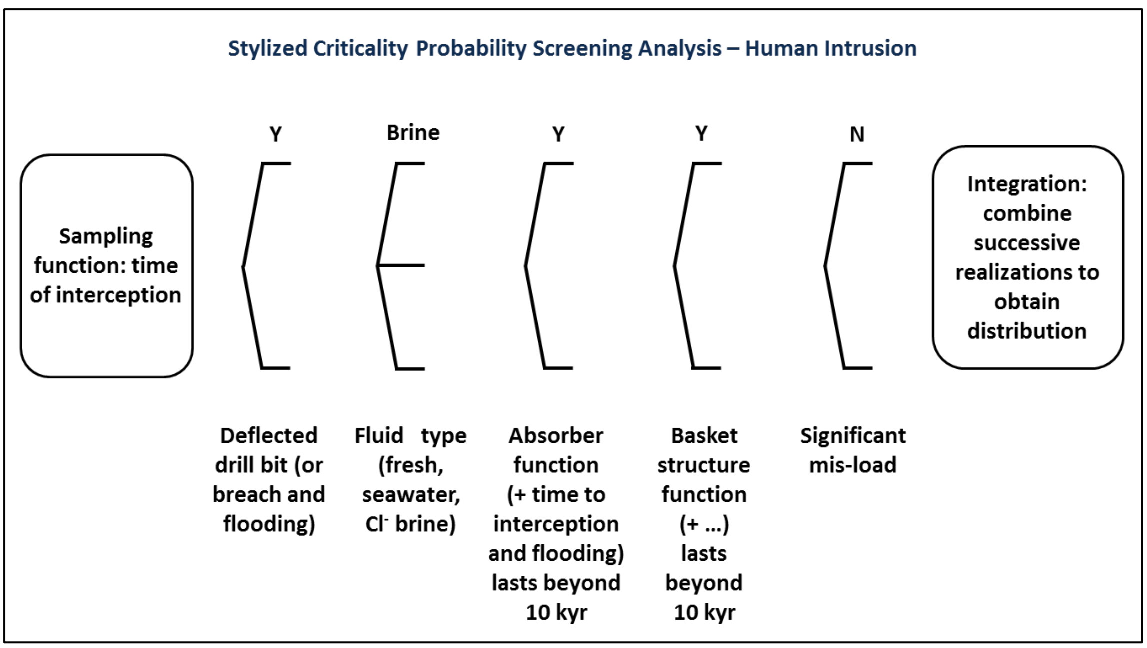


Fig. G-7. Event-tree logic for a stylized criticality screening analysis of the inadvertent human intrusion scenario.