Designation: C 1562 – 03

Standard Guide for Evaluation of Materials Used in Extended Service of Interim Spent Nuclear Fuel Dry Storage Systems[1]

This standard is issued under the fixed designation C 1562; the number immediately following the designation indicates the year of original adoption or, in the case of revision, the year of last revision. A number in parentheses indicates the year of last reapproval. A superscript epsilon (ε) indicates an editorial change since the last revision or reapproval.

1. Scope

1.1 Part of the total inventory of commercial spent nuclear fuel (SNF) is stored in dry cask storage systems (DCSS) under licenses granted by the U.S. Nuclear Regulatory Commission (NRC). The purpose of this guide is to provide information to assist in supporting the renewal of these licenses, safely and without removal of the SNF from its licensed confinement, for periods beyond those governed by the term of the original license. This guide provides information on materials behavior under conditions that may be important to safety evaluations for the extended service of the renewal period. This guide is written for DCSS containing light water reactor (LWR) fuel that is clad in zirconium alloy material and stored in accordance with the Code of Federal Regulations (CFR), at an independent spent-fuel storage installation (ISFSI). The components of an ISFSI, addressed in this document, include the commercial SNF, canister, cask, and all parts of the storage installation including the ISFSI pad. The language of this guide is based, in part, on the requirements for a dry SNF storage license that is granted, by the U.S. Nuclear Regulatory Commission (NRC), for up to 20 years. Although government regulations may differ for various nations, the guidance on materials properties and behavior given here is expected to have broad applicability.

1.2 This guide addresses many of the factors affecting the time-dependent behavior of materials under ISFSI service [10 CFR Part 72.42]. These factors are those regarded to be important to performance, in license extension, beyond the currently licensed 20-year period. Examples of these factors are given in this guide and they include materials alterations or environmental conditions for components of an ISFSI system that, over time, could have significance related to safety. For purposes of this guide, a license period of an additional 20 to 80 years is assumed.

1.3 This guide addresses the determination of the conditions of the spent fuel and storage cask materials at the end of the initial 20-year license period as the result of normal events and conditions. However, the guide also addresses the analysis of potential spent fuel and cask materials degradation as the result of off-normal, and accident-level events and conditions that may occur during any period.

1.4 This guide provides information on materials behavior to support continuing compliance with the safety criteria, which are part of the regulatory basis, for licensed storage of SNF at an ISFSI. The safety functions addressed and discussed in this standard guide include thermal performance, radiological protection, confinement, sub-criticality, and retrievability. The regulatory basis includes 10 CFR Part 72 and supporting regulatory guides of the U.S. Nuclear Regulatory Commission. The requirements set forth in these documents indicate that the following items were considered in the original licensing decisions: properties of materials, design considerations for normal and off-normal service, operational and natural events, and the bases for the original calculations. These items may require reconsideration of the safety-related arguments that demonstrate how the systems continue to satisfy the regulatory requirements. Further, to ensure continued safe operation, the performance of materials must be justified in relation to the effects of time, temperature, radiation field, and environmental conditions of normal and off-normal service. Arguments for long-term performance must account for materials alterations (especially degradations) that are expected during the service periods, which include the periods of the initial license and of the license renewal. This guide pertains only to structures, systems, and components important to safety during extended storage period and during retrieval functions, including transport and transfer operations. Materials information that pertains to safety functions, including retrieval functions, is pertinent to current regulations and to license renewal process, and this information is the focus of the guide. This guide is not intended to supplant the existing regulatory process.

2. Referenced Documents

2.1 *ASTM Standards:*

C 33 Specification for Concrete Aggregates[2]

C 227 Test Method for Potential Alkali Reactivity of Cement-Aggregate Combinations (Mortar-Bar Method)[2]

C 295 Practice for Petrographic Examination of Aggregates for Concrete[2]

[1] This guide is under the jurisdiction of ASTM Committee C26 on Nuclear Fuel Cycle and is the direct responsibility of Subcommittee C26.13 on Repository Waste.

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[2] *Annual Book of ASTM Standards*, Vol 04.02.

C 859 Terminology Relating to Nuclear Materials[3]
C 1174 Practice for Prediction of the Long-Term Behavior of Materials, Including Waste Forms, Used in Engineered Barrier Systems (EBS) for Geological Disposal of High-Level Radioactive Waste[3]

2.2 *Government Documents:*[4]
10 CFR Part 50 Domestic Licensing of Production and Utilization Facilities
10 CFR Part 60 Disposal of High Level Radioactive Wastes in Geologic Repositories
10 CFR Part 63 Disposal of High Level Radioactive Wastes in a Proposed Geologic Repository in Yucca Mountain
10 CFR Part 71 Packaging and Transport of Radioactive Materials
10 CFR Part 72 Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste

2.3 *NUREG Standards:*[5]
NUREG-1536 Standard Review Plan for Dry Storage Cask Systems, January 1997
NUREG-1567 Standard Review Plan for Spent Fuel Dry Storage Facilities, Report, January 1998
NUREG-1571 Information Handbook on Independent Spent Fuel Storage Installations, M. G, Raddatz and M. D. Waters, December, 1996
NUREG/CR-6407 Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, February, 1996, INEL Report 95/0551

2.4 *American Concrete Institute Standards:*[6]
ACI 201.2R-97 Guide to Durable Concrete
ACI 209R-97 Predication of Creep, Shrinkage and Temperature Effects in Concrete Structures
ACI 301-99 Building Code Requirements for Reinforced Concrete
ACI 318-02 Building Code Requirements for Reinforced Concrete
ACI 349-00 Code Requirements for Nuclear Safety Related Concrete Structures
ACI 359-01 Code for Concrete Reactor Vessels and Containments, also designated as ASME Boiler and Pressure Vessel Code, Section III, Div 2, Code for Concrete Reactor Vessels and Containments

2.5 *ANSI Documents:*[7]
ANSI/ANS-6.4-1985 Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants
ANSI/ANS-57.9 Design Criteria for an Independent Spent Fuel Storage Installation (Dry Storage Type)
ANSI/ANS-57.10 Design Criteria for Consolidation of LWR Spent Fuel

2.6 *Other Documents:*
ASME-B&PV Sect III-Div 2 (2001) Code for Concrete Reactor Vessels and Containments[8]
EPRI-1994 Class I Structures License Renewal Industry Report; Revision 1, TR-103842, July 1994

3. Terminology

3.1 The terminology of Terminology C 859 applies to this document except as given below.

3.2 *Definitions of Terms Specific to This Standard:*

3.2.1 *accident-level events or conditions*—the extreme level of an event or condition for which there is a specified resistance, limit of response, and requirement for a given level of continuing capability, which exceed “off-normal” events or conditions. They include both design basis accidents and design-basis for natural phenomena events and conditions. **NUREG-1536, NUREG-1567**

NOTE 1—Specific accident conditions to be addressed have been evaluated for each dry cask storage system (DCSS) and documented in a Safety Analysis Report.

3.2.2 *alteration mode*—a particular form of alteration, for example, general corrosion, passivation. **C 1174**

3.2.3 *ASTM guide*—a compendium of information or series of options that does not recommend a specific course of action..

3.2.4 *canister—in a dry cask storage system (DCSS) for spent nuclear fuel*, a metal cylinder that is sealed at both ends and is used to perform the function of confinement, while a separate overpack performs the functions of shielding and protection of the canister from the effects of impact loading.

3.2.5 *cask—in a dry cask storage system (DCSS) for spent nuclear fuel*, a stand-alone device that performs the functions of confinement, radiological shielding, and physical protection of spent fuel during normal, off-normal, and accident conditions. **NUREG-1571**

3.2.6 *certificate of compliance—in a dry cask storage system (DCSS) for spent nuclear fuel*, a certificate issued by the NRC to the designer/vendor of a specific cask model that meets the requirements set forth in 10 CFR Part 72.236.

3.2.7 *confinement—in a dry cask storage system (DCSS) for spent nuclear fuel*, the ability to prevent the release of radioactive substances into the environment. **NUREG-1571**

3.2.8 *confinement systems—in a dry cask storage system (DCSS) for spent nuclear fuel*, the assembly of components of the packaging intended to retain the radioactive material during storage. These may include the cladding, storage system shell, bottom and lid, penetration covers, the closure welds or seals and bolts and other components. **NUREG-1536**

3.2.9 *criticality—in a dry cask storage system (DCSS) for spent nuclear fuel*, the condition wherein a system or medium is capable of sustaining a nuclear chain reaction. **C 859**

3.2.10 *degradation*—any change in the properties of a material that adversely affects the behavior of that material; adverse alteration. **C 1174**

3.2.11 *degraded cladding—in spent nuclear fuel*, cladding material that by visual inspection appears to be structurally

[3] *Annual Book of ASTM Standards*, Vol 12.01.
[4] Available from Superintendent of Documents, US Government Printing Office, Washington, DC 20402.
[5] Available from the National Technical Information Service, Springfield, VA 22161.
[6] Available from American Concrete Institute, PO Box 9094, Farmington Hills, MI 48333.
[7] Available from ANSI, 11 W. 42nd Street, 13th Floor, New York, NY 10036.
[8] Available from American Society of Mechanical Engineers, 3 Park Ave., New York, NY 10016.

deformed or damaged to an extent that special handling is expected to be required.

3.2.12 *dry cask storage system (DCSS)—in nuclear waste management*, a set of components that performs the functions of confinement, radiological shielding, and physical protection of spent nuclear fuel during normal, off-normal, and accident conditions. Examples would include canister-based systems with their metal or concrete overpack or vault, or an integrated cask.

3.2.13 *dry storage—in nuclear waste management*, the storage of spent nuclear fuel after removal of the water from the fuel, cladding and all components of a dry cask storage system, and after the atmosphere has been replaced with an inert atmosphere.

3.2.14 *independent spent fuel storage installation (ISFSI)*—any complex designed and constructed for interim dry storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage. It must meet the requirements in 10 CFR Part 72. In this guide a Monitored Retrievable Storage (MRS) site is also considered an ISFSI. **NUREG-1571**

3.2.15 *monitoring—in a dry cask storage system (DCSS) for spent nuclear fuel*, testing and data collection to determine the status of a DCSS and to verify the continued efficacy of the system, on the basis of measurements of specified parameters including temperature, radiation, functionality and/or characteristics of components of the system.

3.2.16 *normal events and conditions*—the maximum level of an event or condition expected to routinely occur.

NOTE 2—Specific normal conditions to be addressed have been evaluated for each licensed DCSS and are documented in a Safety Analysis Report for that system.

3.2.17 *off-normal events or conditions—in a dry cask storage system (DCSS) for spent nuclear fuel*, the maximum level of an event that, although not occurring regularly, can be expected to occur with moderate frequency, and for which there is a corresponding maximum specified resistance, limit of response, or requirement for a given level of continuing capability. **NUREG-1536**

NOTE 3—Specific off-normal conditions to be addressed have been evaluated for each licensed DCSS and are documented in a Safety Analysis Report for that system.

3.2.18 *radiation shielding—in a dry cask storage system (DCSS) for spent nuclear fuel*, barriers to radiation, which are designed to meet the requirements of 10 CFR Parts 72.104(a), and 72.106(b), and 72.128(a.2).

3.2.19 *retrievability—in a dry cask storage system (DCSS) for spent nuclear fuel*, the ability to remove spent nuclear fuel from storage for further processing or disposal. **10 CFR Part 72.122 (1)**

3.2.20 *safety analysis report (SAR)—in a dry cask storage system (DCSS) for spent nuclear fuel*, the document that is supplied by a DCSS vendor or site specific ISFSI applicant to the NRC for analysis and confirming calculations (review and approval). **NUREG-1571**

3.2.21 *safety evaluation report (SER)—in a dry cask storage system (DCSS) for spent nuclear fuel*, the document that the NRC publishes after review of a Safety Analysis Report (SAR). **NUREG-1571**

3.2.22 *service conditions—in a dry cask storage system (DCSS) for spent nuclear fuel*, the time of service, temperatures, environmental conditions, radiation, and loading, etc. that a component experiences during storage.

3.2.23 *spent nuclear fuel (SNF), spent fuel*—nuclear fuel that has undergone at least one year of decay since being used as a source of energy in a power reactor, and has not been separated into its constituent elements by reprocessing. **NUREG-1571**

NOTE 4—In this guide, only commercial light water reactor SNF that is clad in zirconium alloy material and has been removed from service is considered.

3.2.24 *sub-criticality margin—in a dry cask storage system (DCSS) for spent nuclear fuel*, the difference between one and the allowed calculated effective neutron multiplication factor (keff), which is maintained at or below 0.95 in accordance with NUREG-1536 and NUREG-1567.

NOTE 5—An adequate margin of sub-criticality is regarded to be 0.05.

3.2.25 *thermal performance—in a dry cask storage system (DCSS) for spent nuclear fuel*, heat-removal capability having testability and reliability consistent with its importance to safety. **10 CFR Part 72.128(a)(4)**

3.2.26 *time limited aging analysis (TLAA)*—a calculation or analysis that addresses the effects of time and environmental conditions on the performance of a system or component.

4. Summary of Guide

4.1 Information in this guide deals with materials aspects of spent nuclear fuel dry storage facilities that relate to license extension beyond the original twenty-year license. Safety and retrievability of the spent nuclear fuel are to be maintained throughout the licensed period.

4.2 Topics addressed in this guide all relate to materials performance including regulations, design, environmental conditions, materials behavior under various conditions and circumstances, monitoring, evaluation, etc. References are provided to guide the user of this document to find additional information and analyses, if needed. The structure of the document is presented here to show the contents and annexes and appendices of this guide.

5. Significance and Use

5.1 Information is provided in this document and other referenced documents to assist the licensee and the licensor in analyzing the materials aspects of SNF and DCSS component performance during extended storage. The effects of the service conditions of the first licensing period are reviewed in the license renewal process. These service conditions are highlighted and discussed in Annex A1 as factors that affect materials performance in an ISFSI. Emphasis is on the effects of time, temperature, radiation, and the environment on the condition of the SNF and the performance of components of ISFSI storage systems.

5.2 The storage of SNF that is irradiated under the regulations of 10 CFR Part 50 is governed by regulations in 10 CFR Part 72. Regulatory requirements for the subsequent geologic disposal of this SNF are presently given in 10 CFR Part 60, with specific requirements for the use of Yucca Mountain as a repository being given in the regulatory requirements of 10 CFR Part 63. Between the life-cycle phases of storage and disposal, SNF may be transported under the requirements of 10 CFR Part 71. Therefore, in storage, it is important to acknowledge the transport and disposal phases of the life cycle. In doing this, the materials properties that are important to these subsequent phases are to be considered in order to promote successful completion of these subsequent phases in the life cycle of SNF. Retrievability of SNF (or high-level radioactive waste) is set as a requirement in 10 CFR Part 72.122(g)(5) and 10 CFR Part 72.122(l). Care should be taken in operations conducted prior to disposal, for example, storage, transfer, and transport, to ensure that the SNF is not abused and that SNF assemblies will be retrievable, the protective value of the cladding is not degraded and remains capable of serving as an active barrier to radionuclide release during transfer and transport operations. It is possible that cladding could be altered during dry storage. The hydrogen effects, fracture toughness of the cladding and the creep behavior are important parameters to be evaluated and controlled during the dry storage phase of the life cycle. These degradation mechanisms are discussed in Annex A2 and Annex A4.

6. Performance Requirements Related to the Design of a DCSS

6.1 Materials for extended service must meet the design and performance requirements given in 10 CFR 72. The DCSS has been designed to store spent fuel safely for a minimum of 20 years and to permit maintenance as required in the original licensed term. Structures, systems and components important to safety have been designed, fabricated, erected and tested to meet standards commensurate with their function and their importance to the safety of the overall system. The service conditions for the renewal period may be less severe than those of the initial licensing period. If the cask contains its original SNF, then the demands on materials properties for an additional 20 to 80 years of storage may be reduced due to lower temperatures and radiation levels. The general assumption put forth here regarding decreases in thermal and radiation conditions are based on the expectation that reloading of SNF does not occur. It is assumed that at the time of license renewal, the reloading of casks (with SNF different from that originally stored in a cask) is very unlikely. If new (replacement) SNF is put in the cask, then the requirements on the material properties and the ability to meet them would have to be determined using the conditions established by the properties of the new SNF.

6.2 *Structures, Systems and Components (SSC)*:

6.2.1 The functions important to safety of DCSS Structures, systems and components (SSC) are [NUREG-1536] to maintain:

6.2.1.1 Thermal performance,

6.2.1.2 Radiological protection,

6.2.1.3 Confinement,

6.2.1.4 Sub-criticality, and

6.2.1.5 Retrievability.

6.2.2 Systems, structures and components that are important to safety must be designed to accommodate the load combinations applicable to normal, off-normal and accident events with an adequate margin of safety per 10 CFR Part 72-, 122b, 122c, and 24c, 10 CFR Part 100, and 10 CFR Part 72.102(l). The DCSS must reasonably maintain confinement of radioactive material under normal, off-normal and credible accident conditions [10 CFR Part 72.236(l)]. The cask must be designed and fabricated so that the spent fuel is maintained in a sub-critical condition under credible conditions [10 CFR Part 72 Part 72.236 § C; 10 CFR Part 72.124 (a)].

6.2.3 For a license renewal, a DCSS should be analyzed to demonstrate that the SSC will continue to perform so as to ensure that SNF is maintained under conditions that meet safety requirements under design basis conditions, even for an extended storage period (up to 80 additional years).

6.2.4 The requirements of 10 CFR 72.122 (h)(1) seek to ensure safe fuel storage and handling and to minimize post-operational safety problems with respect to the removal of the fuel from storage. In accordance with this regulation, the spent fuel cladding must be protected during storage against degradation that leads to gross ruptures, or the fuel and must be otherwise confined such that degradation of the fuel during storage will not pose operational problems with respect to its removal from storage. Additionally, 10 CFR 72.122(l) and 72.236(m) require that the storage system be designed to allow ready retrieval of the spent fuel from the storage system for further processing or disposal.

6.3 *Thermal Behavior*:

6.3.1 The spent fuel cladding must be protected against degradation by thermally activated processes by maintaining the temperature below allowable limits. Spent fuel storage or handling systems must be designed with a heat-removal capability having testability and reliability consistent with its importance to safety [10 CFR Part 72.128(a)(4)]. The DCSS must be designed to provide adequate heat removal capacity without active cooling systems [10 CFR Part 72.236(f)]. The conditions in the second storage period will be less severe than in the original license term since the decay heat (as well as the radiation source term) decreases with time. The decreasing decay heat requires less heat removal capacity during the extended licensing period. Hence, the safety function related to thermal performance is a requirement to protect to fuel, that is, to ensure against the type of cladding damage mentioned in

6.2.3. At the initial licensing of a DCSS, the temperature of the fuel is limited and the cask design is important to the thermal performance requirement of the DCSS. Due to heat decay and the significant decrease in temperatures of the fuel and cask over time, this safety requirement will be met for extended licensing periods provided that the thermal properties of the cask have not been significantly degraded and the geometry of its contents have not been significantly altered.

6.3.2 Examples of components used to meet the thermal performance criteria are (*1*) cooling fins, which, for metal casks, are usually fabricated from carbon steel (SA 283 or SA 285 Grade A), copper, or stainless steel (SA 240 Type 304), so as to increase heat transfer, and (*2*) penetrations in the concrete shielding that allow air to cool the canister.

6.4 *Shielding/Radiation Protection and Confinement*—Radiological protection and confinement features that are sufficient to meet all necessary requirements of 10 CFR Part 72 should continue to be provided. The confinement canister of a DCSS provides a redundant seal. This feature is one that aids in ensuring that the confinement systems perform their safety-related functions in a reliable manner that is predictable over time. In some sub-systems the performance must be under intermittent or continuous monitoring using appropriate instrumentation and control systems. These sub-systems are expected to experience material property changes as they age under the combined influences of radiation and temperature (and in some instances chemical environment) associated with dry cask storage. Typical examples are polymer-based materials, elastomers, and organic based materials. In short, the licensee must be able to determine when corrective action needs to be taken to maintain safe storage conditions. Instrumentation and control systems deemed to be important to safety shall also remain operational during the license renewal period. Radiation exposure and dose rates to workers and the public must not exceed acceptable levels and remain as low as reasonably achievable (ALARA).

6.5 *Sub-Criticality*:

6.5.1 Subcriticality must be maintained [10 CFR Part 72.124]. The neutron multiplication factor, keff, must be maintained at or below 0.95 so as to obtain an adequate sub-criticality margin. The DCSS must be designed to ensure that this limit on the computed keff is not exceeded, under all credible conditions. Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. In an extended license period, special attention should be given to all material and components which may undergo thermal or corrosive alteration or any actions that would result in geometric rearrangement of either the boron (or other poison/neutron absorber) or the SNF.

6.5.2 Boron is the element usually added inside a DCSS to absorb thermal neutrons and to maintain neutron flux at a moderately low level. Other absorbers (Hf, Gd or Cd) may be considered for absorber applications. The level of boron is customarily specified as an areal density (which is the thickness times the volume density) for solids, such as metal alloys and polymers used for mixed neutron absorbers. The geometry or physical configuration of the fixed neutron absorbers in the system is important, and the matrix materials must not fail, corrode or degrade, so as to ensure that the absorber remains in place. If redistribution of SNF rods occurs within the canister, or if there are any significant changes or redistribution of either the absorbing material plates or the moderators of the SNF within the rod, it must be shown that the k_{eff} will remain at or below 0.95.

6.5.3 Neutron absorbing materials must continue to be effective. The license renewal application should evaluate the durability of the neutron absorbing material in its radiation, thermal, stress, and chemical environment in the cask. It should demonstrate that the material remains in place at the end of 20 years, and will remain in place for the license extension period. Consumption of neutron-absorbing materials during dry storage period is generally not a matter of concern because the neutron fluxes are low, and are almost entirely fast. Boron consumed in storage usually represents only a tiny fraction of the available boron in the system.

6.6 *Retrievability*—Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal [10 CFR Part 72.122(l)]. System conditions are set so that materials alteration does not compromise retrievability.

7. The Materials Evaluation Process for Dry Storage License Renewal

7.1 Materials requirements that are important to safety must be considered for license renewal of an ISFSI. The following types of service are to be considered: normal events and conditions, off-normal events and conditions, and accident-level events and conditions. Fig. 1 illustrates an analysis logic that might be considered (in accounting for alterations of materials) during a license renewal. It begins by asking whether conditions have been other than normal, and if they have not, the user establishes the new initial conditions, which result from normal service conditions. When either off-normal or accident conditions had been experienced for a given cask system, the user is referred to appropriate Annex materials in this guide to cover the selected conditions that may require special consideration appropriate to those events. In Annex A1 the principal factors that affect materials performance in ISFSI service are briefly described under the headings of Temperature, Radiation and Chemical Environment. The effects of these overall environmental conditions, over time, on the properties of the materials may be important to the performance of the materials. For a license renewal, the materials alterations and operational events during the first 20-year storage period are considered along with the original design bases, and future materials requirements for the service conditions of the renewal period.

7.2 *Evaluation of Materials Capabilities in Relation to Service Requirements*:

7.2.1 Assess the service conditions: normal, off-normal and accident that occurred during the initial storage period.

7.2.2 Determine the profile or service history (time/temperature, radiation, chemical environment) of the components to be analyzed.

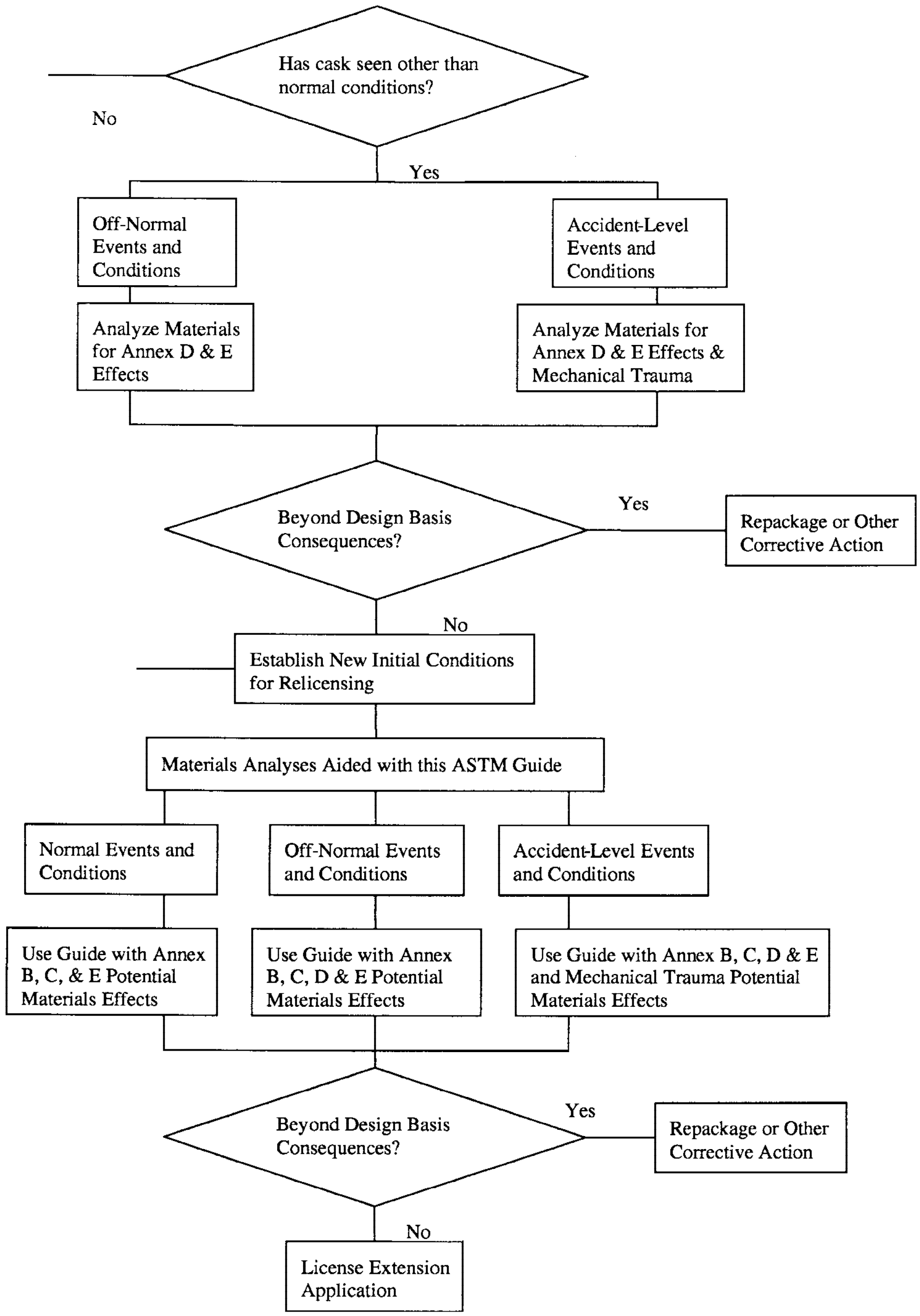

FIG. 1 Analysis Logic and Identification of Materials Conditions to be Considered in ISFSI License Renewal

7.2.3 Establish the relevant properties of SNF and DCSS materials at the start of the license renewal period based on materials alterations that may have occurred during the initial storage period.

7.2.4 Assess the capability of the materials to perform their functional and safety requirements during the renewal period.

7.2.5 Methodologies for life prediction, under the scope of this guide, are concerned with the alteration of the materials used in the sub-systems, structures and components of a DCSS. Guidance is provided on evaluating the most significant material alterations that have been observed or predicted to occur under dry storage conditions during the initial and renewal license periods. To insure system performance in each of these periods, an acceptable methodology for life prediction should (*1*) identify alteration mechanisms, (*2*) quantify the alterations, (*3*) evaluate the effects (on materials properties) of the alterations, (*4*) determine if the alterations compromise any safety functions of the system, and (*5*) determine the consequences of compromising the performance of the component, the sub-system or the system (safety, operational, economic).

The use of an acceptable methodology will help to establish the requirements for materials data and testing, monitoring and surveillance, preventive maintenance and operations management. In addition, it is noted that Practice C 1174 is expected to be a useful reference for evaluations of materials issues related to license renewals for spent fuel dry storage. This ASTM practice includes the prediction of long-term behavior, as well as methods and criteria for accelerated testing and the use of models and mechanistic understandings of alteration processes.

7.3 *Establishing Initial Conditions for License Renewal*—Almost all components of an ISFSI that are subject to license renewal will have undergone only normal service conditions during the initial license period. If off-normal or accident conditions occurred in a manner that had adverse effects on some components of an ISFSI during the initial license period the components would have been required to be restored to their original design and licensing bases. However, the effects, if any, of the off-normal or accident conditions on the ISFSI components are to be included in forming the initial conditions for materials evaluations for ISFSI license renewal and this is shown in Fig. 1. Aging mechanisms evaluated for the original licensed term must again be evaluated for the license renewal term. Similarly, any evidence gained through intermittent or routine monitoring of ISFSI components during the initial storage term that suggest accelerated or unanticipated aging must be evaluated for the license renewal term.

7.3.1 *Normal Events and Conditions*—Normal conditions include a dry and inert protective environment for (*1*) the SNF, and inner and outer surfaces of the cladding, and (*2*) inside the cask and all interior components of the storage container. Neutron absorbers must continue to be adequately effective and structural components (for example, baskets, supports, weld closures, lifting lugs and all other components) of the DCSS must have sufficient strength to meet the required performance. Seals must be maintained in accordance with requirements of the safety evaluation and safety analysis reports. If during the first licensing period for the ISFSI only normal events or conditions have occurred, then it can be assumed that there has been no air or water ingress into the storage casks and no significant damage to the spent fuel due to mechanical damage. Thus, only those material degradation mechanisms discussed for normal conditions need be considered. These include the spent fuel in Annex A2, the DCSS materials in Annex A3, and the concrete in Annex A5. All require analyses needed to establish the initial conditions for a license renewal.

7.3.2 *Off-Normal Events and Conditions*—Any off-normal events that occur during the original license term must be evaluated for their impact on materials behavior and capabilities during the extended term. Under off-normal events and conditions during the initial license period, the ISFSI may have experienced no permanent deformation or design related faults associated with a degradation of capability to perform its full function over the full license period, although operations may be suspended or curtailed. If during the initial 20-year license period, an off-normal event or condition has occurred, and that event has potentially allowed air or water ingress into the storage cask, then those material degradation mechanisms, discussed in Annex A4, must be addressed in addition to the mechanisms discussed in Annex A2, Annex A3, and Annex A5 to establish the initial material condition for the license renewal analysis.

7.3.3 *Accident Level Events and Conditions*—If an accident-level event or condition has occurred during the initial (20-year licensed) period of the ISFSI, there is a possibility of alteration or damage to the spent fuel due to air/water ingress and/or mechanical trauma to the SNF or components of, the storage cask. The material degradation mechanisms discussed in Annex A4, therefore, should be addressed in addition to the mechanisms discussed in Annex A2, Annex A3, and Annex A5 in establishing the initial material condition for the license renewal analysis.

7.4 *Consideration for Future DCSS Usage*—During the license renewal process, the applicant should assess the radiation and thermal load for which license renewal is sought. If the SNF is anticipated to remain throughout the renewal period, then the service conditions as a result of the SNF will be less severe than those of the initial license period. However, if credit is taken in the license renewal application for the less severe conditions, then the SNF permitted to be stored in the DCSS may be limited by those less severe conditions. Consideration should be given to the unlikely event that the applicant may need to reload the DCSS during the renewal period with SNF having different thermal and radiological properties than the SNF stored at the time of the initial license period. To preclude SNF loading penalties in a DCSS, the applicant may consider storing SNF having design basis properties during the renewal period. The original design basis may differ from that required in the renewal period. Therefore, the design basis used in a renewal application should correspond with the type of SNF to be stored during the renewal period.

7.5 *Degradation of SNF and DCSS Components During Extended Storage*:

7.5.1 After decades of storage, SNF in extended dry storage is expected to undergo little, if any, further alterations. For continued safe operation and to protect the SNF, the cask, neutron absorbers, shielding materials baskets, supports, closures, lifting lugs as well as other (including minor) components of the DCSS, must retain sufficient strength and other physical and mechanical properties to meet the required performance criteria, specifications, etc.

7.5.2 Alteration modes that could lead to degradation or failure of cladding in extended dry storage under normal storage conditions are discussed in Annex A2. Alteration

TABLE 1 Guide to Use of Annex A1 through Annex A4 for Material Evaluations

	Normal	Off Normal/Accident
Factors Affecting Performance	A	A
Fuel (UO_2)[A]	[B]	D
Cladding[A]	B	D
Cask Components[A]	C	D
Pad/Concrete	E	E

[A] Only corrosion is discussed in Annex A4. There is no discussion of mechanical disruption as these are unique to a given event and reports are developed to describe their relevance to safety.

[B] Under normal conditions, the fuel material is not expected to be adversely affected.

modes that could lead to degradation or failure in the other DCSS components under normal conditions are discussed in Annex A3. Other degradation mechanisms that could become important in off-normal or accident conditions are discussed in Annex A4. Mechanisms by which concrete can be altered are presented in Annex A5.

ANNEXES

(Mandatory Information)

A1. FACTORS THAT AFFECT MATERIALS PERFORMANCE IN AN ISFSI

A1.1 Introduction

A1.1.1 Factors that affect the behavior of SNF (and other components) in ISFSI service include (*a*) temperature, (*b*) radiation, and (*c*) the environment. The values of these factors change as a function of time.

A1.2 Temperature

A1.2.1 Temperature is an important factor for material performance since many degradation mechanisms are thermally activated. Over time in the DCSS, the temperature will decrease due to decreasing decay heat. Temperatures discussed in this section are fuel-cladding temperatures. While the heat decay characteristics of the fuel govern the cladding temperature, in a DCSS various factors, for example, initial enrichment, decay time (time after reactor discharge, or time in wet and dry storage), cask design, and even the fuel burnup level, affect the temperature over the time that fuel remains in dry storage. The temperature profile of the DCSS varies both radially and axially, with the maximum temperature occurring over the center 50 % of the cask **(1,2)**[9] and falling away at the outer edges.

A1.2.2 The temperature of the various components of a particular DCSS depends on the burnup, initial enrichment, and decay time of the spent fuel and the design (that is, orientation, heat removal capability) of the DCSS. The temperature profile of the DCSS varies both radially and axially, with the maximum temperature occurring over the center 50 % of the cask **(1,2)** and falling away sharply at the outer edges. Temperature drop over time has been calculated using cask heat transfer codes and decay heats from ORIGEN **(2-4)**. In general a temperature drop from 380°C to 100°C was calculated (for a typical 5 year, 30 GWD/MTU SNF) for the first 10 years, with the temperature remaining at about 100°C for the next 90 years **(2)**.

A1.2.3 One methodology for assessing the effect of temperature on material performance was given by Peehs et al. **(5-7)** who suggests four phases (modes) that define the temperature range over a given period for evaluation of expected degradation mechanisms. Rates of temperature change are principally a function of the age of the SNF. Duration and temperature of these phases are functions of the specific fuel and cask conditions. Peehs et al. **(5-7)** dealt with commercial light water reactor fuel with Zircaloy™ cladding.

A1.2.3.1 *Phase I*—Temperatures above 300°C are characterized by a rapid decrease in temperature. Phase I is a short term stage, typical of the first two years in dry storage for SNF out of the reactor for less than seven years. The duration of this stage in dry storage is, of course, is a function of the initial time in wet storage.

A1.2.3.2 *Phase II*—Temperatures between 175 and 300°C are characterized by a medium rate of decrease in temperature occurring later in interim storage (usually from two to five years in dry storage).

A1.2.3.3 *Phase III*—Temperatures between 120 and 175°C are characterized by a moderate rate of decrease in temperature.

A1.2.3.4 *Phase IV*—Temperatures below 120°C, characterized by a negligible decrease in temperature.

A1.2.3.5 Phases III and IV are characteristic of the temperatures expected for extended dry storage.

A1.2.4 Thermal conditions external to the cask can be important to the alteration of properties to the concrete components. When concrete is used as shielding the design temperature range is given in the Safety Analysis Reports for the DCSS system. A general discussion of the effect of temperature on concrete is found in Annex A5.

A1.3 Radiation

A1.3.1 After 20 years of dry storage, the fast neutron fluence at the interior of the DCSS is typically on the order of 10^{14} n/cm^2 and the cumulative gamma dose is on the order of 10^9 rad. The radiation shielding within a DCSS absorbs neutrons and decreases the exposure levels and the potential damage to the materials of the exterior components but, in general, at this fluence level the effects on materials of interest are small. These levels of neutron fluence could potentially have some effects on mechanical properties of steels, but not for any austenitic materials. The ferritic materials would require at least several orders of magnitude greater neutron fluence to have any significant effect on mechanical properties **(8)** and the effects would be limited to those on impact properties, that is, on either on the upper-shelf energy absorption or on the transition temperature behaviors.

A1.3.2 While these levels of gamma radiation (10^9 rad) are not significant for materials used inside a cask system, their absorption by the shielding materials is important to the radiation protection afforded to people.

A1.3.3 For discussion of materials used in seals, see A3.3.3.3.

[9] The boldface numbers in parentheses refer to the list of references at the end of this standard.

A1.4 Chemical Environment

A1.4.1 The potential chemical environments to be considered are: backfill gases which may be air, nitrogen, helium, or argon, residual water remaining in the cask after drying, zinc vapor if internal components are galvanized, and (potentially) fission products. The effects of radiolysis on the composition of the internal atmosphere should be assessed in a license renewal, whenever concern exists over the presence of nitrogen or water.

A2. POTENTIAL DEGRADATION MECHANISMS AND BEHAVIOR OF SPENT NUCLEAR FUEL CLADDING UNDER NORMAL CONDITIONS

INTRODUCTION

Alteration modes that could lead to degradation or failure of cladding in extended dry storage under normal storage conditions are discussed here.

A2.1 Potential Degradation Mechanisms of Spent Nuclear Fuel Cladding

A2.1.1 Under normal storage conditions, the major degradation mechanisms of spent nuclear fuel (SNF) cladding that have been hypothesized to result in (lead to) failure include creep, hydrogen mechanisms, stress corrosion cracking, and diffusion controlled cavity growth.

A2.1.2 *Creep*:

A2.1.2.1 It is widely held that the enveloping criterion for consideration of cladding integrity during inert dry storage is creep. Creep is the progressive deformation of a material under an applied stress. Creep occurs in three stages. The primary stage has rapid deformation and a decrease in creep rate over time, the secondary stage has a constant creep rate and the tertiary stage has a rapid creep rate increase with time until fracture occurs. The creep behavior of unirradiated cladding may be a function of many variables including chemical composition, metallurgical structure and processing conditions. For irradiated cladding, radiation effects overshadow these fabrication and chemical effects. The two principal factors in the creep behavior of irradiated cladding are the hoop stress and the temperature. The hoop stress results from the rod internal pressure, a combination of the original fill gas and the fission gas release during service, and the temperature results from the decay heat of the fuel assemblies. At low temperatures and stresses, the deformation (strain) is negligible and can be ignored; at high temperatures and stresses the strain can be substantial. For typical fuel cladding hoop stresses, strain may be detected at temperatures above about 300°C although significant strain (for zirconium alloy cladding) is not expected to occur until the temperature is well in excess of 350°C. Over long storage times, both the pressure and the temperature decrease thus the strain rate tends to zero. At temperatures below 300°C, creep may be considered to be immeasurably slow. The creep strain rate and strain at failure of spent nuclear fuel cladding are affected by material parameters like alloy composition, fabrication steps (for example, cold work, solution anneal, recrystallization anneal), hydride content, and radiation fluence. Irradiation effects are predominant, in irradiated materials. In general, the creep strain can be calculated from creep equations, but their applicability for a particular material or set of materials parameters should be questioned and not applied without consideration of all important factors. At temperatures of the drying, transportation and initial storage operations, there may be significant recovery of mechanical and irradiation damage, which will affect the creep behavior **(9,10)**.

A2.1.2.2 To avoid degradation of cladding, the strain needs to be limited. The strain calculated to occur in storage should be determined to be less than the creep strain to failure. In creep tests at temperatures between 250 to 400°C of Zircaloy™ cladding irradiated up to burnup of 64 GWd/MtU, no failures have been observed below 2 % strain **(11,12)**. Therefore, a conservative cladding strain limit of 1 %, has been used in several countries, including Germany and the U.S.A.

A2.1.3 *Hydrogen-Related Mechanisms and Effects*—Zirconium alloys absorb hydrogen during corrosion with water. The quantity of hydrogen absorbed into the matrix depends primarily on the environmental conditions and the composition of the alloy. The quantity of hydrogen absorbed, determined as a fraction of the total hydrogen generated, is known as the hydrogen pick-up fraction. For Zircaloy™ in either BWR or PWR service this fraction is typically in the range of 10 %, or equivalent to less than about 500 ppm. As with most materials, the solubility of hydrogen in zirconium alloys increases with increasing temperature in the unirradiated condition **(13-15)**. Irradiation does not appear to have a significant impact on this behavior. The solubility of hydrogen in Zircaloy™ at room temperature is significantly less than 1 microgram per gram. At 400°C the calculated solubility is in the 170 to 300 microgram per gram range. These values compare to typical hydrogen concentrations of 15 to 20 microgram per gram in the as-received condition. As a result of corrosion during irradiation the hydrogen concentration can increase to values in excess of 300 microgram per gram (for higher burnup fuels, the concentrations may be considerably higher) and hence result in hydride formation and precipitation. The zirconium hydrides formed can impact the mechanical properties of the Zircaloy™, generally increasing the strength and decreasing the ductility but may also produce hydride embrittlement and delayed hydride cracking (DHC) **(16-19)**.

A2.1.3.1 *Hydride Embrittlement*—Hydride embrittlement is due to the formation of hydrides sufficient to cause detrimental effects to mechanical properties, including tensile ductility, fracture toughness and ultimate fracture strength. The amount of ductility degradation depends on hydride orientation, concentration and distribution. As the fraction of hydrides with radial orientation increases, the effects of cladding hoop stress on cladding ductility become more significant. Hydrides typically precipitate as platelets (with thickness to length aspect ratios of 0.02 to 0.1) along specific crystallographic planes. Hydride platelets oriented normal to the stress direction cause large reductions in strength and ductility, whereas hydride platelets oriented parallel to the stress direction have little effect. In cladding for commercial SNF, the radial hydride orientation is regarded as more detrimental (than circumferential hydrides) and is important upon cooling under high circumferential stress. Except for some of the earlier (~pre-1980) fuels, commercial fuel cladding is fabricated with a specific texture (that is, preferred orientation of hexagonal close-packed grains) that results in hydrides predominately oriented in the circumferential direction **(20)**. However under sufficient stress, hydrides will reorient to the radial direction **(2, 20-22)**. The amount of hydrogen necessary to severely reduce ductility of SNF cladding depends on the storage service temperature and the orientation of the hydrides. Small amounts of hydrogen (as low as 30 ppm may be required to reduce ductility at room temperature **(16,23)** and, in general, at higher temperatures a higher concentration (over 600 ppm **(23)**) may be required at 300°C. Thus, the minimum amount of hydrogen that has been reported (for a reduction in ductility) may be very low at room temperature **(16,23)** and may be very high at 300°C **(23)**. The combined effects of hydrides and prior irradiation on ductility in Zircaloy™ cladding are a complex function of temperature. The current understanding suggests that at temperatures around 300 to 400°C radiation damage determines ductility loss, while at room temperature the effects of hydrogen and radiation damage are additive **(24)**.

A2.1.3.2 *Delayed Hydride Cracking*—Delayed hydride cracking (DHC) is a process that occurs by diffusion of hydrogen atoms to a flaw region. The fracture process from hydrides includes an incubation period, the formation of a hydride zone, growth of a flaw, and subsequent fracture of the brittle hydride zone. The tensile stress region at the flaw provides the driving force for diffusion of hydrogen atoms. When the hydrogen concentration exceeds the solubility limit, hydrides will start to form and grow. When this hydrided zone reaches a critical size under a sufficient tensile stress, fracture through this zone can occur. The repeated process can eventually lead to failure. Initiation of DHC occurs only if the stress intensity is above a threshold value, and stress intensities in inert dry storage conditions are expected to be lower than this critical stress intensity.

A2.1.4 *Stress Corrosion Cracking*—This mechanism involves chemical corrosion of a crack tip with crack extension being driven by a stress on the cladding. Pescatore **(25)** reviewed the testing done on SCC of irradiated Zircaloy™ and determined that under conditions expected for dry storage that failure would not occur by this mechanism. Even under conditions of higher than expected stress (270 MPa), only pinhole breaches were observed. It is shown in A2.2.3.1 of this guide, that under normal storage conditions that gas and/or volatile fission product release from the fuel pellets to the gap of an intact rod will be negligible. Therefore, the stress on the cladding is highest at the start of storage and decreases with storage time due to a decreasing temperature. In addition, it is expected that no corrosive fission products will be released from the fuel pellets to the cladding gap during the storage period. Therefore if the unexpected SCC is going to occur in a particular fuel rod, it would likely occur early in storage, when stress is highest, not later during the extension period.

A2.1.5 *Diffusion Controlled Cavity Growth*—Diffusion related phenomena manifest themselves as voids formation and migration, ion migration, grain boundary alteration and enrichment, and formation and migration of reaction products from the site of generation. Diffusion processes accelerate at elevated temperatures, and the kinetics of the processes generally follows an Arrhenius rate law. A temperature threshold sometimes exists below which the kinetics of the process may be too slow to be of any concern even for a dry storage period of up to 100 years. Diffusion controlled cavity growth is a potential mechanism for mechanical degradation, but one that has never been observed on actual SNF cladding and its potential effect on degradation of the cladding in a ISFSI is expected to be very small. The occurrence of failures from DCCG has not been observed to date, either in Zircaloy™ or in any components of dry storage facilities.

A2.2 Driving Forces for Cladding Degradation

A2.2.1 The driving forces for cladding degradation are stress and chemical effects. These driving forces act independently and synergistically depending on the particular degradation mechanism. The SNF fuel rods have an internal stress due to the internal gas pressure. Discussions given below indicate that the chemical effects that may lead to SNF cladding degradation include fission products, hydrogen, and zinc vapor.

A2.2.2 *Stress*—Stress is a driving force of the potential SNF degradation mechanisms of creep, delayed hydride cracking, stress corrosion cracking and diffusion controlled cavity growth (see A2.1). The performance limiting stress of SNF cladding is the hoop or tangential stress. The axial stress is half of the hoop stress; therefore, the resultant axial strain is small compared to the hoop strain. The hoop stress, σ_ϕ, is defined as $\sigma_\phi = pr/t$, where p is the pressure, r is the radius of the cladding, and t is the wall thickness of the cladding. During the dry storage period, the hoop stress can (potentially) be changed due to any of several factors (*1*) an increased diameter due to creep, (*2*) a decreased cladding thickness due to corrosion, (*3*) possibly any extension of existing flaws, and (*4*) changes in the internal rod pressure.

A2.2.2.1 *Internal Gas Pressure*—The internal gas pressure of commercial SNF is due to as-fabricated fill gas and the fission gas released from the fuel. The pressure at beginning of storage depends on the temperature and internal void volume of the SNF fuel rods. Fill gas pressurization was introduced as a design feature of commercial LWR fuel rods in the late 1970s to increase the heat transfer in the rods. Usually the fill gas is inert He, although some of the early PWR rods were not evacuated of air prior to backfill. The nominal fill gas pressure of pressurized LWR fuel rods has evolved overtime, being between 1.5 to 3.5 MPa (200 to 500 psia) at 20°C for PWRs and 0.3 to 0.7 MPa (50 to 120 psia) at 20°C for BWRs. The source of fission gas contributing to internal gas pressure is discussed in A2.2.3.1.

A2.2.2.2 *Void Volume*—The void volume of as-fabricated LWR fuel rods consists of the plenum above the fuel column, the annular gap between fuel pellet column and cladding inner diameter, and the void spaces at pellet-to-pellet interfaces due to chamfers and dishes. Two fuel performance phenomena reduce void volume during in-reactor service. First, the cladding creeps inward due to the higher external coolant pressure (N.B. referred to as cladding creep down) until it contacts the fuel. Second, the fuel swells (radially and axially) and this reduces the annular gap between the cladding and fuel and decreases the plenum void volume, respectively. The pressure inside the cladding during dry storage exceeds the canister fill pressure, resulting in cladding creep at sufficiently high cladding stresses and temperatures. The void volume will increase during dry storage if the cladding creeps.

A2.2.3 *Chemical Effects*—The environment referred to here includes conditions such as moisture, chemistry and other effects of the surroundings (within, throughout and outside all components of the DCSS) of the ISFSI. The environment can affect the durability and performance of the fuel, cladding and all materials used in structures, systems and components. The environment can affect the protection provided by coatings, for example, oxides on the materials or interactions of coatings or coating products. Materials must withstand effects of environmental conditions associated with normal operation and storage conditions and postulated accidents. Accidental and environmental hazards causing breach of a DCSS could compromise the ability of the system to perform at the intended design level. If chemical effects have been properly considered in the original license application, the period of a license renewal is expected to present no new safety issues.

A2.2.3.1 *Fission Product Release to Fuel/Cladding Gap*—Fission products are not likely to cause degradation of SNF cladding in ISFSI service **(26)**. The release of metastable implanted fission products occurs by diffusion processes. Since diffusion is a thermally activated process, the temperature distribution in the fuel will be of importance when considering fission product releases. Data and operations experiences are available to describe the release of the fission gases from the fuel during reactor operation. Typically, the fuel surface temperature during reactor operation is around 400 to 500°C and its centerline temperature is between 1300 and 1800°C, depending on the heat rating and burn-up. For temperatures above 1000°C the diffusion coefficients are only dependent on the temperature. Those coefficients are decreasing between 1800 and 1000°C from 10^{-12} cm^2/s to 10^{-16} cm^2/s, that is, by 4 orders of magnitude. Between 1000 and 400°C the pure thermal diffusion is combined in-core with a radiation supported term only decreasing by 2 orders of magnitude. The radiation level in storage is negligible in comparison to in-core conditions; hence, only pure thermal diffusion will contribute to fission gas release. The diffusion coefficient at storage temperatures is at least 8 orders of magnitude less when compared with that at fuel centerline temperature or 5 orders of magnitude less than that of the fuel average temperature. Therefore under dry storage conditions for fuel temperatures less than 400°C, it is expected that no fission gas will be released at all, even under extended periods of storage time. Investigation of the Cs and I release **(5-7,26)** from irradiated fuel has shown that the release is diffusion controlled and the diffusion coefficients of those fission product isotopes are similar to those of the fission gases. It was found that a temperature exists under which there is no release at all. This temperature decreases with increasing burn-up and will be above 800°C to 900°C. This corresponds to the findings for the fission gas release as discussed above. "Release of iodine from UO_2 generally is low at in-reactor fuel temperatures (1000 to 1800°C) and will be negligible at inert dry storage temperatures (<400°C)" **(2,26)**. The threshold concentration of iodine available to the cladding, which is the amount needed for promoting stress corrosion cracking, is greater than 5×10^{-6} g/cm^2 **(2,27-29)**. Iodine at this concentration level will not be present in inert dry storage. Therefore, it can be concluded that neither fission-Cs-isotopes nor fission-I-isotopes will be released under normal dry storage conditions from the fuel to the gap of a spent LWR fuel rod. New fission products are not expected to be released during storage and those released at reactor temperatures (for example, Cs and I) are expected to become immobilized through chemical reactions (for example, formation of Cs compounds). Therefore, no cracking is expected because both stress and iodine concentrations are expected to be low in the SNF of a DCSS.

A2.2.3.2 *Hydrogen*—See A2.1.3 for discussion mechanisms and effects.

A2.2.3.3 *Zinc Vapor*—In addition, it is noted that when zinc is present inside the cask system, it forms a vapor at modest temperatures. The vapor can react with Zircaloy™ cladding **(30)** and perhaps other materials present inside the system. Reaction products may continue to form until the available zinc vapor is consumed. As the ductility and strength properties are not as favorable as those of the zirconium alloy cladding, it may be important to understand the extent of these reactions under actual storage cask conditions where their effects on the behavior of stored cladding materials may become important **(31)**. Further research has been conducted to clarify this area **(32)**, but it too has left some questions unanswered so that further research will be required to close this potential issue for realistic cask loading and storage conditions.

A2.2.3.4 *Radiation Effects*—In general, in-pile irradiation causes an increase in strength and a decrease in both ductility and fracture toughness. This occurs as a result of radiation hardening, and in Zircaloys™ it reaches a saturation level within the first cycle of reactor operation. During subsequent cycles, mechanical properties of Zircaloy™ are not affected significantly. Drying, and dry storage may lead to partial recovery in Zircaloy™ resulting in some improvement in ductility. The radiation doses received during dry storage will be approximately four to seven orders of magnitude lower than received during reactor irradiation. Therefore, no new radiation damage is expected to occur in dry storage.

A3. POTENTIAL DEGRADATION MECHANISMS AND BEHAVIOR OF DCSS MATERIALS UNDER NORMAL CONDITIONS

INTRODUCTION

The properties of non-fuel components of the system may be affected or altered by the normal service conditions encountered in the initial storage period. For these components the importance of the service requirements and the effects of the service conditions are described here. The potential degradation mechanisms that may alter materials properties are important to the assessment. Industry practices related to plant license renewal provide a good basis for informational needs required in a DCSS license extension and these too are described here.

The materials analysis for the license extension should address the effects of normal events and conditions during the license extension period on the spent fuel and DCSS materials. These service effects must be analyzed from the perspective of the initial condition of the materials at the beginning of the license extension period. These conditions are likely to be much less demanding (that is, lower temperature, radiation, etc.) when compared with initial licensing conditions. If the ISFSI has not experienced off-normal or accident-level events during the initial license period, it would be expected that those potential degradation mechanisms operative during the initial license period would be considered for the license extension period. Thus, the analysis of the normal condition for a license renewal would focus on the degradation mechanisms and aging factors given in Annex A2 (for SNF cladding) and Annex A3 (for the cask and other components). Under normal storage conditions, the selected degradation and/or failure mechanisms to be considered would include sensitization, and hydrogen effects for the cask and components.

A3.1 DCSS Materials and Components

A3.1.1 Tables A3.3 and A3.4 **(33)** give examples on some components, basket assemblies and the types of materials used in the primary confinement vessel for some DCSS. Typical materials that penetrate shell lids, etc. are shown in Table A3.5 **(33)**, and materials used for lifting devices are shown in Table A3.6. Examples of materials used in gamma and neutron shielding are shown in Table A3.7. The vessel consists of a metal shell with one to three metal lids sealed by multi-pass welds or by bolts with metal seals. Each of these materials may have reactions relating to exposure time to high temperatures, irradiation and hydrogen effects. These reactions should be analyzed for the specific materials used in a given ISFSI in terms of stability and performance of intended function.

A3.2 DCSS Materials and Components Classification

A3.2.1 *Classification of Materials by Function*—A convenient way to identify the DCSS potential degradation mechanisms for components other than SNF and cladding is to examine the components by their function **(33)**. Six primary functions for cask components are: (*1*) confinement, (*2*) criticality control, (*3*) shielding, (*4*) heat transfer, (*5*) structural integrity, and (*6*) operations support. It is noted that most of the components of safety classification A, fall under the first three functions. The A classification is for components critical to safe operation. The safety classifications for the functions and components are based on terminology of NUREG/CR-6407.

A3.2.2 *Safety Related Components*—A table of safety related materials used in the storage system may be tabulated as an aid for conducting the review of expected behavior, safety analyses, problems and properties under extended service

TABLE A3.1 Examples of Dissimilar Materials in Contact

Material	Contact Material	Comment
Ni, SS, shell/lid, coatings	Al, SS, ethylene, propylene, silicone rubber, PTFE, Viton®	Metallic or elastomer O-rings in contact with sealing surface
Structural (shell, basket, etc.)	Boral®, BISCO®, lead	Shielding material in contact with structural component
Non-coated materials	Coatings	Coated material in contact with non-coated or coated material (basket/shell, etc.)

TABLE A3.2 Examples of Coatings Used in DCSS

Component	Coating	Comment
Inner cask surfaces	Electroplated Ni (1.5 mm)	Also O-ring seating surfaces
	Zn/Al, Ti/Al	Metallic flame-sprayed
	Everlube 812 & 823, Carbo Zinc 11, Dimetcote 6	Radiation and corrosion resistant up to 1200F if completely dry, but generally rated at 750-800°F for continuous exposure
O-ring seating surfaces	SS weld overlay	
Outer surfaces, fins	Epoxy resin, paint	
Basket assembly (Al)	Al oxide	500 A thick
Basket assembly (C-steel)	Metallic	Thermally applied

TABLE A3.3 Examples of Materials Used for Confinement Components in DCSS

Shell	Lids	Bolts/Studs	Seals
Nodular cast iron	GGG40	304L SS	Metal
Carbon steel	515 Gr 70	193 Gr B7	Inconel X730© spring, Al jacket
Low alloy steel	515 Gr 70	320 Gr L43	Nimonic 90© spring, Al jacket
	516 Gr 70	SB637 Gr NO7718	Inconel X750©, Al, 304L SS jackets
Forged steel	508 Cl 4b	SA564 Type 630 H1150	
	350 Gr LF3		Elastomer[A]
Stainless steel, Concrete	304, 304L, 630		Ethylene propylene copolymer, silicone rubber, propylene, Viton, PTFE (polytetrafluoroethylene)

[A] Generally used for leak testing and do not have a confinement function.

TABLE A3.4 Examples of Materials Used for Basket Assemblies in DCSS

Component	Material	Comment
Basket assembly	304 SS with Boron	Cu plating, Al rails
	6061-T651 Al	500 A oxide coating
	SA 705 Type 630 SS	Al heat transfer fins
	SA 516 Gr 70	
	240 SS	Cell structure
	479 SS	Rail structure
Neutron absorbers	Metal matrix composites (B4C/Al matrix, Al clad)	In SS wrappers welded to panels
	Castable borated shielding	Encased tubes

TABLE A3.5 Description of Devices that Penetrate Confinement Shell/Lids

Component	Material	Seal
Drain port plug	SA240 Type 304 SS, SA36, SA516 Gr 70	Metallic
Vent port plug	SA240 Type 304 SS, SA36, SA516 Gr 70	Metallic
Pressure port plug	SA240 Type 304 SS	PTFE O-ring
Interlid port plug	SA705 Type 304 SS	PTFE O-ring
Pressure sensor	Ni-plated Cu Be diaphragm	Electronic (Ag, Au, SS), ceramic and weld seals

TABLE A3.6 Examples of Materials Used for Lifting Lugs/ Trunnions/Grapples

Component	Material	Comment
Lug/trunnion/grapple	304 SS, 508 Class 2a, 564 Type 630, A537, SA705 Type 630 SS	
Lug/trunnion bolts	304L SS, 193 Gr B7, 320 Gr L43, 4340	Generally same as lid bolts

TABLE A3.7 Examples of Materials Used in Gamma and Neutron Shielding

Component	Material	Comment
Gamma shielding	Shell material (thick-walled)	See Table A3.3
	B29 chemical lead	Between inner and outer shell
	Reinforced concrete (0.28 MPa)	29-in.-thick overpack, A36 inner liner
	Reinforced concrete (0.35 MPa)	Used for overpack. A 615 or A 706 Gr 60 steel for inner liner
Neutron shielding	Polyethylene, polypropylene	In shell wall and lids
	BISCO© (borated polymer)	Cast into metal containers (Al, Fe)
	RX 277© (castable borated shielding)	In lids

conditions, and some of the information would be available from the initial licensing. The table could include the service conditions for those materials. The primary class functions, could be correlated with the associated components such as cask, seals, lids, baskets, plugs, neutron absorbers, support pads, support structures, etc. The list could include the type of material, specifications and standards, vendor, mechanical properties, surface finish, internal surface coatings, temperature/pressure exposure, welds, etc. This table could facilitate and expedite the review process by permitting a quick reference for each component. The listed information is a useful starting point for specialists (materials, shielding, etc.), in assessing potential problem areas. The table would illustrate various types of information that the reviewer may expect to be available throughout the original licensing application and original SAR. In an application for a license renewal, tabular information can be organized and serve as an aid in determinations of the suitability of the materials for their continued service. The table would give the names of each component part of the DCSS and, where applicable the function, the material specification(s) to which it is produced, and the nominal values for the following parameters: strength, thickness, surface finish or coating, and dissimilar materials which may be in direct contact. If materials are welded, the list could include the welding process and filler metal, preheat treatment, and post-weld heat treatment, if any, or the governing code. Other tabulations could include the stress (nominal and maximum) in service, the residuals (chemicals/foreign matter) on the surface of the component after loading and after storage, the service temperatures (for the storage period, as well as during loading and during unloading), the internal pressure (min., max.) and the type/composition of gas or liquid in the container. The tabulation should include all materials used for components with an important-to-safety function, for example, confinement, confinement systems, transport, criticality control, shielding. In addition, materials that coat or in other ways support or interact (physically, chemically, or electrochemical) with the components with important-to-safety functions could be tabulated. Information in this table can aid the reviewer in formulating the types of performance-related questions that are important for each component of a storage system. Additional information and guidance on the materials, parts and performance are available in Annex A1, Annex A2, Annex A4, and Annex A5.

A3.3 Potential Degradation Mechanisms of DCSS Materials

A3.3.1 Materials used in the DCSS must meet performance requirements needed in the respective functions. A table showing functions, materials, properties, and specifications of components could be developed similar to that discussed earlier **(33)**. Examples of potential degradation mechanisms associated with the materials of the DCSS include mechanical failure, sensitization in welds, corrosion, thermal effects and environmental interactions. Degradation mechanisms would be specific to the materials used and type of exposure and service in the DCSS. Fast neutron irradiation of metals increases the yield strength and reduces ductility. For example, the increase in Charpy-V transition temperature as a function of fast neutron fluence for various pressure vessel steels has been reported **(8,34)**. Extrapolation of the results in these references to the 10^{14} n/cm^2 range indicates that no measurable embrittlement would occur in steel during 20 years of storage. Neutron flux levels are even less for the period from 20 to 80 years of extended storage. In many DCSS, the primary confinement vessel is metallic and designed with some of the same American Society for Mechanical Engineers (ASME) codes used to design nuclear reactor pressure vessels. However, some allowances and modifications take into account the different requirements of storage vessels and nuclear reactor pressure vessels. Several DCSS use reinforced concrete as the primary shielding material. A methodology for degradation assessment has been developed in the nuclear industry for the concrete containment building. Thus, some aspects of the methodology for life extension developed for nuclear reactor systems might be appropriate for DCSS.

A3.3.2 *General Degradation Mechanisms*:

A3.3.2.1 *Galvanic Corrosion of DCSS Materials*—When dissimilar metals are connected electrically in the presence of an electrolyte, a galvanic cell is established and electrochemical actions occur. For example, if bolts are anodic to a large component of a cask system, the bolts may corrode quickly and impair their ability to function successfully as a fastener in the cask system. The galvanic series lists metals in terms of their electrochemical potential and may be useful in establishing potential problems in either aqueous systems or vapors of moderate to high humidity. As many different metals are used within a cask system, and some may be exposed to water vapor and therefore subject to its electrochemical effects, it is important to note the possible interactions for dissimilar-metal systems and to pass judgment on the possibilities for unfavorable interactions in relation to functions that are important to the safety of the systems. Some examples of dissimilar metals in contact in the DCSS are given in Table A3.1 **(33)**. A relatively active metal such as zinc, in the presence of large areas of ferrous surfaces that are cathodic to zinc, will corrode at a rate greater than that for zinc alone. The products of this reaction are gaseous hydrogen, and ions and compounds of zinc. These products must not impair any safety functions under the service conditions of the cask system. In addition, it is noted that when zinc is present inside the cask system, it forms a vapor that can interact with the Zircaloy™ cladding **(30-32)** and other materials present inside the system. Applicants for license extension could demonstrate either, that the effects of these interactions will not impair the safety functions or that such interactions are not possible under the service conditions. Coatings are a source of dissimilar metals. Coatings that are used include paint, epoxy, and oxides, as well asthermally sprayed metal, electroplated metal and weld overlays. The source of the zinc discussed in A2.2.3.3 was paint on the inside of the cask. Typical examples of coatings used are shown in Table A3.2 **(33)**.

A3.3.2.2 *Corrosion of Canister, Cask and Other Components*—Corrosion of external (or exposed) components of storage systems is detectable to inspectors during routine monitoring of the storage system. As corrections needed for this type of corrosion would occur in the normal operation of an ISFSI, this corrosion is not regarded to be a concern that is specific to the license renewal process. Of direct interest to license renewal would be any corrosion that would reduce the effectiveness of a confinement boundary (see Annex A4 for further discussion). The presence of water could increase potential for initiation of corrosion degradation of the metal components of the DCSS. However, with routine monitoring, it is unlikely that a failure of a confinement boundary would go undetected for long. For components that can not be monitored routinely, evaluation methods including analysis or remote inspection might be considered for assessment of their condition.

A3.3.3 *Component Specific Degradation Mechanisms*—Components, under higher loads of stress, whether residual or applied, are more likely to be affected by ISFSI service conditions than components having low levels of stress. Thus, welds, bolts and seals are of concern in this regard.

A3.3.3.1 *Welds*—Welds could have degraded during the first 20 years and should be evaluated. Welding material specifications, fabrication procedures and properties of fabricated components, along with available inspection results, should be reviewed in license extension applications. This information may be given by reference to the code that was used in the weld. For ISFSI casks licenses, most welds are inspected volumetrically using methods (ultrasonic and radiographic) that ensure that defects and cracks don't exceed acceptable levels. Welds that have not had the benefit of post-weld-stress-relief heat treatment have high levels of residual stress and a greater potential for extension of flaws during the first licensing period. Closure welds are in this category and special treatment is given to them during fabrication, for example, PT examinations of various weld passes and the use of redundant seals (as mentioned in 6.4), so as to ensure performance over the life of the system. Other concerns for welds include sensitization caused by heating during welding, the tendency for divergent microstructures and residual stress fields to promote precipitate formation and growth that can affect corrosion behavior, and even strength level, in some welds. Additional evaluation of welds of materials that, usually, are not welded, may be needed to ensure that the welding process and materials chosen have yielded a durable component that has not become adversely affected during the licensing period. Records describing the conditions of welds at the onset of the first (original licensed) storage period could be used as a guide in determinations concerning the need for inspections or repair.

A3.3.3.2 *Bolts*—The service life of a bolt in a DCSS is based on a 20 year to 30-year historical database. On questioning the acceptability of a bolt for extended service of an additional 20 to 80 years of service, the original acceptability of the bolts should be checked and the potential failure problems unique to bolts must be analyzed. Bolts that are improperly heat-treated may crack in service under normal conditions (if tempered too little) or under off-normal (accident) conditions (if tempered too much). Factors that may affect their performance, for example, environmental effects of corrosion or creep, must be considered and discussed in an application for extended service. Materials and their specifications used in bolt selection should be stated along with assessments of the current condition of the bolts at the time of application for license extension. Bolts may be removed tested and replaced, as necessary.

A3.3.3.3 *Metallic and Elastomeric Seals*—Metallic seals can be used for primary and secondary confinements, but elastomeric seals can only be used for secondary confinement. Elastomeric seals are more sensitive to radiation and temperature than metal seals. Radiation will cause polymerization (cross-linking) in elastomers, and doses above about 106 rad will result in an increase in strength and a loss of elasticity (embrittlement). Radiation also tends to reduce permeability for the same reason; at least initially higher doses reverse these trends. Radiation may also produce a chemical breakdown of the elastomer, and the decomposition products may interact with other components of the DCSS. Elastomers may also start to degrade at higher temperatures and extended exposure times. Metallic seals should not be affected by radiation because of the high threshold value required to alter the mechanical properties of metals. However, time and temperature effects on metallic seals should be carefully assessed to ensure continued safe performance. Any schedules for these assessments should be carefully reviewed and updated in relation to their applicability during the license extension period and in accordance with the information developed after the original SER was completed. Elastomeric seals, designed to be replaced at intervals, should be specified in the SAR. The SAR should be checked to determine the procedures to be followed to ensure continued safe performance. Most elastomers are resistant to absorbed radiation doses up to 10^6 rad **(35,36)**. The radiation exposure conditions for safety-related elastomeric o-rings, seals, and neutron shielding materials should be determined and compared with the radiation resistance for the particular elastomer. Some studies **(37)** conclude that for neutron shielding materials, thermal damage is far more significant than radiation damage. In addition, the release of corrosive gases, for example, fluorine from an elastomeric seal, must be considered for any potential effect of the gas on DCSS components.

A3.3.3.4 *Shielding Materials*—The metal shell for thick-walled monolithic metal canister systems provides gamma radiation shielding, by lead in multi-shelled metal casks, and by the concrete in thin-walled metal casks. Neutron shielding can be located in the metal shell(s) or exterior to the shells. Typical materials used for gamma radiation shielding include thick walled steel, B29 chemical lead and/or reinforced concrete (0.28 MPa and 0.35 MPa). Materials for neutron shielding include polyethylene, polypropylene, borated polymer, and concrete. Shielding materials are shown in Table A3.7 **(33)**. During extended storage, the shielding materials must not creep or slump to any extent that critically impairs the safety function. The effect of radiation on any polymeric materials is of concern because of potential changes in the polymers configuration by slumping or granulation. Radiation can cause chain scission or crosslinking of the polymer structure. Chain scisssion can result in a reduction in molecular weight and an increased tendency of the polymer to creep. Crosslinking can increase the molecular weight of the polymer, increasing its hardness and resistance to creep. These radiation effects are sensitive to the applied dose rate and the presence of oxidants and other reactive species that make polymer performance application specific. The use of accelerated radiation testing results may be misleading in evaluating polymer performance and useful life. Shielding performance can be directly verified by field measurements of dose rates.

A3.3.4 *Time Limited Aging Analyses*—An important consideration in a license renewal may be a list of time-limited aging analyses (TLAA), similar to the one defined in 10 CFR 54.3. The analysis might demonstrate the following:

A3.3.4.1 The analyses remain valid for the period of extended operation;

A3.3.4.2 The analyses have been projected to the end of the period of extended operation; or

A3.3.4.3 The effects of aging on the intended function(s) will be adequately managed for the period of extended operation.

A4. CONSEQUENCES AND POTENTIAL DEGRADATION MECHANISMS UNDER OFF-NORMAL/ACCIDENT CONDITIONS

INTRODUCTION

If an off-normal or accident level event has occurred during the initial licensing period, the materials analysis for the license renewal must address the potential consequences of off-normal and accident-level events and conditions as defined in NUREG-1536 for the initial condition of the SNF after 20 years of storage, and potential behavior of the system during the extended storage period. The materials degradation mechanisms of Annex A4 should be analyzed in addition to the mechanisms of Annex A2 and Annex A3.

Accident level events and conditions may compromise the ability of selected components to perform at the intended design level. Some accident conditions are cask drop, cask tip-over, airflow blockage, and fire that could lead to fuel damage, leakage of the confinement boundary or explosive over-pressure. Some natural phenomena events include flood, tornado, earthquake, burial under debris, lighting, seiche, tsunami, and hurricane. Effects of each accident should be analyzed for its possible consequences during the extended storage service.

If off-normal/accident conditions exist, the storage conditions of the SNF could change due to either air or water ingress, or excessive temperature excursion. In addition, the possibility of fuel damage by mechanical trauma also may occur. In the event of off-normal/accident conditions, retrievability must be considered and reevaluated by the licensee. If there is air ingress, oxidation of the Zircaloy™ cladding and fuel may occur.

A4.1 Air Ingress

A4.1.1 Oxidation of both the fuel and cladding are thermally activated processes governed by the time at temperature and the amount of oxidant available. During the initial license period, when temperatures within the DCSS are higher, the potential for adverse consequences due to oxidation is greater than during the extended license period. The total amount of air admitted to the DCSS must be considered in addition to the temperature and the total time that the off normal or accident condition existed when determining the consequences of air ingress. Limited access of air can be assumed if the seals of the DCSS failed during the first (original licensed) dry storage period or if the system assessment indicates the potential of the sealing system leaking in the license renewal period. Small leaks, where the amount of air ingress has been minimal, should result in only minor oxidation of either the fuel or cladding, with limited consequences.

A4.1.2 *Oxidation of UO_2*—Oxidation of the fuel, UO_2, is possible only if the fuel is exposed directly to air in the DCSS. A comprehensive review of the mechanisms and kinetics of fuel oxidation has been performed **(38)**. Typical spent fuels oxidize by first forming either U_3O_7 or U_4O_9, phases that are more dense than the original UO_2. This first transition is accompanied by a net contraction of the fuel, relieving mechanical stress on the cladding. Oxidation typically occurs first along the fuel grain boundaries. Thus, the release of fission gases, such as ^{85}Kr, to the DCSS relatively early during this transition is possible. Upon further exposure to air, the fuel is oxidized to U_3O_8, a phase which is approximately 36 % less dense than the original fuel. The swelling of the fuel as U_3O_8 forms has been shown to supply sufficient mechanical stress to split the cladding **(39,40)**. U_3O_8 formed by oxidation of the fuel is a fine powder that spalls from the fuel surface. The release of fines and/or fuel relocation from split cladding must be evaluated if U_3O_8 formation is suspected. The extent of oxidation of irradiated UO_2 is a time and temperature-dependant phenomenon. In addition, the oxidation of spent fuel to U_3O_8 has been shown **(41)** to be burnup-dependent. As burnup increases, the soluble fission products and actinides stabilize the matrix and inhibit the formation of U_3O_8. For example, at 229°C, the reported incubation time for crack propagation due to fuel oxidation was approximately 1000 h for a rod with an assembly average burnup of about 12 MWd/kg U, whereas a rod with an assembly average burnup of about 30 MWd/kg showed no sign of splitting or dilation after 5962 h **(39)**. At the lower temperatures expected during extended storage, oxidation of even low-burnup fuels will take sufficiently long, at least months or years, that detection and mitigation of the off-normal condition should prevent fuel oxidation from being a concern. At higher burnup levels, under the conditions of extended storage, sufficient oxidation to propagate cladding defects should not occur and no mitigative measures are required. Excessive oxidation of defected SNF should be considered in transferring SNF.

A4.1.3 *Oxidation of Zircaloy*™—Excessive oxidation of the Zircaloy™ cladding combined with an internal stress can potentially breach the cladding. Cladding failure could result in a risk of contamination of the interior of the confinement vessel and needs to be evaluated for retrievability. Oxidation of the Zircaloy™ **(42)** is a thermally induced process. Dry storage under inert gas conditions leads to no further increase in the oxide layer over and above the condition of final discharge from the reactor, since the storage conditions rule out the presence of oxidizing substances. In the event of air ingress, dry oxidation at storage temperatures below 300°C is probably of no concern since the extent of oxidation is much smaller than the cladding thickness. Above 300°C, oxidation of the cladding should be evaluated, depending on the temperature and amount of air. Oxidation of Zircaloy™ has been discussed.

A4.2 Water Ingress

A4.2.1 Water ingress may raise the potential for degradation of the cladding, canister, cask and other components due to the creation of a humid atmosphere. Possible effects of the humid atmosphere include: radiolysis of the moisture to create highly oxidizing radicals, corrosion of the cladding and cask components, and hydrogen ingress into the cladding. The presence of moisture can enhance or alter existing degradation mechanisms. For example, if the relative humidity inside the DCSS exceeds 50 %, then the oxidation of the fuel can be accompanied by formation of hydrated phases of the fuel **(38)**, accelerating the oxidation process. A thermal excursion will accelerate the rate at which the phenomena will occur. During a thermal excursion inside a storage cask, oxidation and corrosion reactions of a uniform nature could result in decreases in the cladding wall thickness that would weaken the cladding mechanically. However, it is expected that any remaining water in the cask would uniformly corrode the Zircaloy™ cladding. Because there is a large available cladding surface and only a small amount of available water, the overall effect would be insignificant. Defects in the cladding that could serve as initiation sites for cracks or reactions may have occurred during reactor service or wet storage. The effects from reactor service and storage would be primarily those pertaining to forms of localized corrosion and their contribution to the hydrogen content of the cladding. Pitting is a severe form of localized corrosion and occurs at sites of breakage in protective surface oxide films, defects in materials, and other discontinuities. Zircaloy™ is resistant to pitting attack, but is not immune. Environments conducive to Zircaloy™ pitting include halide solutions containing ferric or cupric ions **(43)**. Casks are designed to maintain an atmosphere that precludes the presence of these environmental constituents under normal service and these conditions are not likely to form even under off-normal conditions. Stress corrosion cracking (SCC) may occur in stainless steels and other metallic materials used in the DCSS components and SNF cladding. Stress corrosion cracking is a complex form of localized corrosion that occurs in the presence of a corroding environment and a tensile stress. If present, halides can also contribute to SCC. SCC has been discussed in detail in the literature **(28)**. Galvanic corrosion is of consequence only for instances of liquid water ingress or under conditions of equilibrium saturation pressure. It is not expected that any buildup of the hydrogen content of cladding would occur in storage, unless a thermal excursion with water ingress would occur under accident conditions. The applicant may need to evaluate the potential for hydride-induced failures. It is important to note that hydrogen can be absorbed by cask materials, for example, ferrous materials, other than the cladding.

A5. DURABILITY AND PROPERTIES OF CONCRETE STRUCTURES AND COMPONENTS

A5.1 Concrete Structure and Components

A5.1.1 Reinforced concrete structures and components may play multiple roles in Independent Spent Fuel Storage Installations (ISFSI) sites. These many and varied roles include providing radiological shielding or forming ventilation passages, weather enclosures, structural supports, access denial, foundations, earth retention, anchorages, floors, walls, removable shields, bulk fill, and protection against natural phenomena and accidents. Bulk fill of concrete or other materials may be emplaced with an enclosing structure to provide additional shielding and strength.

A5.1.2 All concrete used in an ISFSI isreinforced, regardless of the functional role or need for structural strength or integrity [NUREG-1536]. The structural design of the reinforced concrete structures shall withstand the effects of credible accident conditions and natural phenomenon events without impairing their capability to perform safety functions. The principal safety functions stated in the initial license include maintaining subcriticality, containing radioactive material, providing radiological protection for the public and workers, and maintaining retrievability of the stored spent nuclear fuel (SNF).

A5.2 Codes and Standards

A5.2.1 Certain codes and standards have been accepted for reinforced concrete structures associated with DCSS. ANSI/ANS-57.9 generally applies to ISFSI design and also to construction of DCSS and its associated ISFSI. The NRC has not accepted the use of a set of criteria selected from multiple standards and codes, except when the selected criteria meet the most limiting requirements of each code. However, in recognizing a graded approach to quality assurance, the NRC has approved the use of ACI 349 for design and material selection of reinforced concrete structures important to safety (not confinement), and has allowed the optional use of ACI 318 as an alternating standard for construction [NUREG-1536]. The principal codes and standards used in DCSS include the following:

A5.2.1.1 Concrete Containment ACI 359, also designated as Section III, Division 2, of the ASME Boiler and Pressure Vessel Code, Subsection CC.

A5.2.1.2 Reinforced Concrete Structures Important to Safety ACI 349, except that ACI 318 may be allowed for construction provided conditions and limitations described in NUREG-1536 are met.

A5.2.1.3 Other Reinforced Concrete Structures ACI 349 or ACI 318; if ACI 349 is used for design, the NRC accepts use of ACI 318 for construction [NUREG-1536]. In addition, an alternative to the temperature requirements of ACI 349 is provided in NUREG-1536.

A5.3 Aging-Related Effects of Concrete

A5.3.1 The concrete typically used in DCSS consists of Type II Portland cement because of its higher sulfate resistance and lower heat of hydration relative to the general purpose Type I Portland cement. However, the potential aging-related degradation effects of both types of concrete should still be evaluated. The various admixtures that are used to improve air entrainment and workability, to modify the hardening or setting characteristics, to aid in curing process, to reduce evolution of heat, or to provide other material property enhancements should also be included in the scope of review.

A5.3.2 The reinforcements that act together with the concrete in resisting forces include any embedded bars, wires, strands, or other slender members and should also be included in the review for aging-related degradation effects.

A5.3.3 Portland cement has been used in concrete structures since the early 1800s, and its proven durability to aging-related effects is one of the main reasons for its widespread use in buildings and industrial facilities. However, concrete that is exposed for an extended period of time to extreme environments such as freeze-thaw, aggressive chemicals, flowing abrasive fluids, and elevated temperature may experience significant aging-related degradation in performance. The aging-related degradation effects that are evaluated for all concrete and the steel reinforcements include loss of material, cracking, and change in material properties. As part of the license renewal program, various aging degradation mechanisms and the applicability within bounds of the specific material and environments for concrete structures or components are evaluated to determine if a particular aging degradation mechanism is significant to cause detrimental degradation of its intended safety function.

A5.3.4 Loss of material in concrete structures or components is characterized as scaling, spalling, rust staining, pitting, and erosion. These effects result from one or more of the following aging-related degradation mechanisms: freeze-thaw, abrasion and cavitation, elevated temperature, aggressive chemical attack, and corrosion of reinforcement steels.

A5.3.5 Cracking in concrete structures or components is characterized as general cracking, map cracking, hairline cracking, pitting, and erosion. These effects result from one or more of the following aging degradation mechanisms: freeze-thaw, reaction with aggregates, shrinkage, settlement, elevated temperature, irradiation, and fatigue.

A5.3.6 Change in material properties is evidenced in concrete structures or components as increases in permeability and porosity of concrete material, and reductions in pH value, tensile strength, compressive strength, modulus of elasticity, and bond strength of concrete. These effects result from one or more of the following aging degradation mechanisms: leaching of calcium hydroxide, aggressive chemical attack, elevated temperature, irradiation, and creep. In addition, the potential change in material properties of the reinforcement steel is due to irradiation.

A5.4 Aging-Related Degradation Mechanisms of Concrete

A5.4.1 The license renewal technical evaluation consists of four parts. First, the component evaluation basis is established, which includes component descriptions, the general component design bases, and relevant component operating history. Second, the aging-related degradation mechanisms that could affect this component are described and their potential significance to component safety function performance as defined in the original license of the DCSS is evaluated. Third, the effectiveness of existing programs, which also helps manage the facility aging phenomena, is examined to determine if credit can be taken for the existing programs. Fourth, for case where the existing programs cannot be shown to adequately manage the effects of age-related degradation, aging management options for plant specific programs are recommended for the extended operation. The following aging-related degradation mechanisms are identified from a review/evaluation of nuclear power plant operating experience, relevant laboratory data, and related experience in other industries [EPRI-1994]:

A5.4.2 *Freeze-Thaw*:

A5.4.2.1 Repeated cycles of freezing and thawing may change the mechanical properties and physical form of concrete [EPRI-1994]. The durability of concrete to freeze-thaw damage mechanism depends primarily on the porosity characteristics of aggregates, the presence of moisture to saturate the fines pores in aggregates, and the permeability of the harden cement mortar matrix to the passage of water. Freeze-thaw damage typically occurs on relatively flat surfaces such as pavement, where water remains in contact with the concrete. Freeze-thaw damage often originates at the surface and therefore is readily detected by visual inspections. The effects of freeze-thaw damage are local in nature and by itself will not affect the strength of the concrete.

A5.4.2.2 Concrete structures or components located in areas of the country that experiences numerous freeze-thaw cycles with significant winter rainfall are more likely to exhibit damage than areas in milder climates. Specification C 33 groups the areas of the U.S. into severe, moderate, and negligible weathering regions on the basis of the weathering index. This index is the product of the average annual number of freezing cycle days and the average annual winter rainfall in inches. If concrete structures or components located in a geographic region subject to negligible weathering conditions, that is, a weathering index of less than 100 day-in. per year, the freeze-thaw is not a significant aging degradation mechanism for the concrete, and requires no further evaluation [EPRI-1994].

A5.4.2.3 Freeze-thaw damage is not a significant aging degradation mechanism for concrete structures and components, if the concrete is of an appropriate mix and construction quality or is located in geographic regions not subject to severe weather conditions with significant freeze-thaw cycles [EPRI-1994]. Also, concrete structures are designed in accordance with the ACI 318-63, a later version of ACI 318, or ACI 349-85, and constructed in accordance with ACI 301-66 or later using materials conforming to ACI and ASTM standards are not subject to freeze-thaw aging degradation mechanism [EPRI-1994].

A5.4.3 *Leaching of Calcium Hydroxide*:

A5.4.3.1 When water passes through cracks, inadequately prepared construction joints, or inadequately consolidated areas during concrete placement, the calcium compounds in the concrete can be dissolved due to presence of small amount of calcium ions in water. Since the most readily soluble calcium compounds in concrete is calcium hydroxide or lime, water can leach lime from concrete. When calcium hydroxide or lime in concrete is leached away, other cementitious constituents become exposed to chemical decomposition, eventually leaving behind silica and alumina gels with little or no strength **(44)**. The significance of leaching effect depends on salt content and temperature of water [EPRI-1994]. The areas of leaching in concrete can be found in concrete structures or components subject to flowing water, ponding, or hydraulic pressure. The leaching action of the water can only occur if the water passes through the concrete.

A5.4.3.2 Leaching over long periods of time can increase the porosity and permeability of concrete, making it more susceptible to other forms of aggressive attack and reducing strength. Leaching can also lower the pH of the concrete and affect the integrity of the protective oxide film of reinforcement steel [EPRI-1994].

A5.4.3.3 The dissolving and leaching actions of the percolating water are related to permeability of concrete. Resistance to leaching action can be enhanced by using a concrete with low permeability [EPRI-1994]. A dense concrete with a suitable cement content that has been well cured will be less susceptible to leaching of calcium hydroxide because of its low permeability and low absorption. Guidance to assure a dense and well-cured concrete is provided in ACI 201.2R-67.

A5.4.4 *Aggressive Chemicals*:

A5.4.4.1 Concrete is highly alkaline ($pH > 12.5$) and therefore is vulnerable to degradation by strong acids **(45)**. Acid attack can increase porosity and permeability of concrete, reduce its alkaline nature at the surface of the attack, reduce strength and render the concrete subject to further degradation **(45)**. Portland cement concrete, regardless of its composition, will not withstand exposure to highly acidic fluid for long periods of time [EPRI-1994]. Sulfates solutions of potassium, sodium, and magnesium sometimes found in groundwater may attack concrete over time. Sulfate attack can produce significant expansive stresses within the concrete, leading to cracking, spalling, and strength loss. Groundwater chemicals may also damage the foundation concrete [EPRI-1994].

A5.4.4.2 A dense concrete designed and constructed in accordance with ACI 318-63 and ACI 301 with materials conforming to applicable ACI and ASTM standards will result in a concrete with low permeability and high resistance to aggressive chemical attack [EPRI-1994]. If concrete structures or components are not exposed to an aggressive chemical environment, the aging degradation of concrete strength due to chemical attack is considered to be insignificant [EPRI-1994].

A5.4.5 *Reactions with Aggregates*:

A5.4.5.1 Chemical reactions may develop between certain mineral constituents of aggregates and alkalis that compose the Portland cement **(46)**. These alkalis are largely introduced in the concrete by cement, but may also be present from improper admixtures and salt-contaminated aggregates [EPRI-1994]. Seawater and solutions of deicing salt can also inject alkalis into concrete by action of penetration. Three types of chemical reactions may occur depending upon the composition of the aggregates. They are alkali-aggregate reaction, cement-aggregate reaction, and expansive alkalis-carbonate reaction [EPRI-1994].

A5.4.5.2 Alkali-aggregate reaction, also known as alkali-silica reaction, occurs when aggregates that contain silica get in contact with alkaline solutions. All silica minerals have the potential to react with alkaline solutions, but the degree of reaction and ultimate damage incurred can vary significantly [EPRI-1994]. Alkali-aggregate reaction can cause expansion and severe cracking of concrete structures **(47)**. A map and data showing geographic areas known to yield natural aggregates suspected of, or known to be capable of, alkali-silica reaction is included in ACI 201.2. The reactivity of such aggregates might not be recognized until the structures are over 20 years old, even if used in combination with high alkali cement [EPRI-1994].

A5.4.5.3 Cement-aggregate reaction, which is a second type of reaction similar to alkali-aggregate reaction, occurs between the alkalis in cement and siliceous constituents of the aggregates, and is complicated by environmental conditions that produce high concrete shrinkage and alkali concentrations on the surface due to drying (aging) of the concrete [EPRI-1994].

A5.4.5.4 Expansive alkali-carbonate reaction is a third type of reaction between certain carbonate aggregates and alkalis, which in some instances produces expansion and cracking of concrete. Certain limestone aggregates have been reported as reactive [EPRI-1994]. Aggregates that react with alkalis can induce expansion stress in the concrete to a degree of varying severity, and in some cases, the expansion stresses may be high enough to produce cracking of the concrete structures. This cracking is irregular and has been referred to as "map cracking" [EPRI-1994].

A5.4.5.5 Operating history of nuclear power plant concrete structures do not indicate that structural integrity of these concrete structures is significantly affected by alkali-aggregate reactions. Aggregates used in the concrete can be tested and petrographically examined in accordance with Practice C 295 and Test Method C 227 to determine potential for reactivity with alkalis. When aggregate reactivity is considered a possibility, limitations may be imposed on cement alkalis during construction, and/or an effective pozzolan is used in combination with the cement. Reaction with aggregates is not a significant aging degradation mechanism for concrete structures designed and constructed in accordance with provisions given in the ACI 201.2.

A5.4.6 *Corrosion of Embedded Steel*:

A5.4.6.1 Corrosion is an electrochemical process involving metal, oxygen and an electrolyte that results in the formation of ferric oxide, that is, rust. The oxide product which has a significantly greater volume than the original metal can result in tensile stresses and eventually cause hairline cracking, followed by rust staining, spalling and more severe cracking in the concrete surrounding the embedded steel [EPRI-1994]. Typically, the high alkalinity ($pH>12.5$) nature of concrete provides an environment around embedded steel and steel reinforcement, which protects them from corrosion. If the pH is lowered (for example, $pH<10$) due to leaching of alkaline products through cracks, intrusion of acidic materials, or carbonation, corrosion may occur **(44)**. Chlorides could also be present in constituent materials of the concrete mix (that is, cement, aggregates, admixtures, and water), or they may be introduced environmentally. The severity of corrosion is influenced by the properties and type of cement and aggregates, as well as the concrete moisture content [EPRI-1994].

A5.4.6.2 The degree to which concrete will provide satisfactory protection for embedded steel is, in most instances, a function of the quality of the concrete and the depth of concrete cover over the steel. The permeability of concrete is also a major factor affecting corrosion resistance. Concrete of low permeability contains less water under a given exposure and hence is more likely to have low electrical conductivity and better resistance to corrosion [EPRI-1994]. Such concrete also resists absorption of salts and their penetration to the embedded steel and provides a barrier to oxygen, which is an essential element of the corrosion process. Low water-to-cement ratios, low aggregate to cement ratios, and adequate air entrainment increase resistance to water penetration and thereby provided resistance to corrosion [ACI 201.2R-67].

A5.4.6.3 Corrosion of embedded steel can be an aging degradation mechanism for concrete structures. If the concrete is degraded by other aging mechanisms, which may reduce the protective cover of the steel, corrosion may occur at a significantly higher rate. Adequate management of the other aging mechanisms will help control the corrosion of the embedded steel [EPRI-1994].

A5.4.6.4 If steel-reinforced concrete is exposed to aggressive groundwater ($pH<11.5$ or 500 ppm chlorides with availability of oxygen) for extended periods, aging degradation due to corrosion of the embedded steel should be considered a potentially significant aging degradation mechanism [ACI 201.2R-67]. Use of concrete with a low water-to-cement ratio, low aggregate to cement ratios, and adequate air entrainment, which yields low permeability, and sufficient reinforcement distribution to minimize crack development, will prevent corrosion of the embedded steel and degradation of the concrete [EPRI-1994].

A5.4.6.5 Induced or stray currents can cause rapid corrosion of embedded steel. Induced currents are of special concern if the DCSS is located near a power station or transmission line. Cathodic protection systems used to protect other components of the ISFSI can be a source of stray currents.

A5.4.6.6 If the concrete structures are designed and constructed in accordance with the provisions given in ACI and ASTM standards-which provide a good quality, dense, and low permeability concrete that provides adequate concrete cover over the embedded steel-corrosion of embedded steel is not a significant aging degradation mechanism [EPRI-1994].

A5.4.7 *Elevated Temperatures*:

A5.4.7.1 The concrete mechanical properties such as the compressive strength, tensile strength, and modulus of elasticity may be degraded when subjected to prolonged exposure to elevated temperature **(48)**. Test data suggests that reductions in excess of 10 % begin to occur in the temperature range of 180 to 200°F **(48)**. However, concrete age affects the magnitude of strength loss—the older the concrete, the lower the strength loss. ASME Boiler and Pressure Vessel code, Section III, Division 2 [6] indicates that aging degradation mechanism due to elevated temperature is not significant so long as concrete temperature is less than 150°F. ACI 349 allows local area temperature to reach 200°F before special provisions are required [ACI 349]. Long term exposure to high temperature, that is, greater than 150°F in general area, or greater than 200°F in localized area, may cause a change in material properties. If these temperature limits are exceeded for a long period of time, the effect of aging-related degradation should be evaluated to assure that any change in concrete mechanical strength is not excessive for extended operation [ACI 349].

A5.4.7.2 Long-term exposure to elevated temperature above 300°F may cause concrete surface scaling and cracking. Since very few localized areas in DCSS systems would experience temperatures in this range, aging degradation due to elevated temperature above 300°F is not required for consideration.

A5.4.8 *Irradiation*:

A5.4.8.1 When exposed to high levels of fast and slow neutrons, concrete may exhibit aggregate growth, decomposition of water, and thermal warming of concrete. High gamma radiation may affect the cement paste portions of the concrete; producing heat and causing free water migration and exchange with atmosphere **(49)**. With loss of free water, the mechanical properties of concrete, that is, compressive strength, tensile strength, and modulus of elasticity may be subjected to aging degradation mechanism [ANS-6.4-1985], **(50)**. Impact of direct radiation on the mechanical properties of concrete can be found in ACI publication SP-55 **(51)**. The hydrogen atoms present in water can influence the moderation of fast neutrons, thus affecting the neutron shielding characteristics of concrete if significant water evaporates. Concrete degradation due to neutron and gamma irradiation is not always observable. The most likely physical manifestation of excessive irradiation is cracking and/or spalling of concrete caused by thermal stress. ANSI/ANS-6.4-1985 reports that for incident energy fluxes less than 1010 MeV/cm^2-s, nuclear heating is negligible. This standard also indicates that for concrete temperature maintained below 65°C (149°F), the amount of degradation to be expected is minimal, and need not be given consideration.

A5.4.8.2 In DCSS systems, the level of irradiation over the extended operation is not expected to reach a level that is sufficient to cause significant mechanical strength reduction of concrete, therefore, irradiation effect is not considered to be a significant aging degradation mechanism.

A5.4.9 *Creep*:

A5.4.9.1 Creep is the time-dependent increase of strains in hardened concrete that has been subjected to sustained stress, primarily compressive in nature. The sustained stress results from dead load, live load, and pre-stress on the structure and from temperature effects. Creep deformation is a function of loading history, environment and material properties of the concrete. The time-dependent creep deformation of concrete under compressive load consists of cumulative strain resulting from progressive cracking at the aggregate-cement paste interface, from moisture exchange with the atmosphere, and from moisture movement within the concrete **(52)**.

A5.4.9.2 Crack in the concrete due to creep is typically small and not visible since it causes cracking at the aggregate-cement paste interface [EPRI-1994]. Creep degradation is not significant enough to result in concrete degradation or in exposure of reinforcing steel to environmental effects. Creep is significant when new concrete is subject to load; the effects of creep decrease exponentially with time. Any degradation due to creep can be detected in the first few years of the concrete service life. According to ACI 209R-82, approximately 78 % of creep occur within the first year, 93 % within 10 years, 95 % within 20 years, and 96 % within 30 years. Therefore, creep is not a significant aging degradation mechanism.

A5.4.10 *Shrinkage*—Excess water is typically added to the concrete mix to improve its workability during forming of concrete structure. Shrinkage of concrete occurs initially during curing as the result of excess water leaving the concrete. This curing period typically continues several months after placement. As excess water evaporates into the surrounding environment, tensile stresses are induced in the concrete due to internal pressure from the capillary action of water movement, and cracks develop. Subsequent drying and shrinkage occurs in concrete for up to about 30 years. Over 90 % of the shrinkage occurs during the first year and about 98 % in the first five years [ACI-209]. Therefore, shrinkage is not a significant aging degradation mechanism for licensing renewal.

A5.4.11 *Managing Aging-related Degradation Effects of Concrete Structures*—Any age-related degradation mechanism is considered significant if, when allowed to continue without any additional prevention or mitigation measure, it cannot be shown that the component would maintain its safety function during the license renewal period or extended operation following the initial license. The potential for significant age-related degradation of specific component evaluated is dependent upon design features, design basis, operating history, and the extent to which they are susceptible to the age-related degradation mechanism(s). If it can be shown that a DCSS component is not affected by the degradation mechanism under consideration, or is only affected to such a small degree that the component safety function is not adversely affected throughout the license renewal term, then the component/degradation combination is not significant. Otherwise, the combination is potentially significant. If a potentially significant problem (component/degradation mechanism combination) is adequately addressed by effective existing programs, then the issue is not a license renewal concern as it is considered to be resolved on the basis that the degradation is managed acceptably. Combination of mechanisms and components for which existing programs cannot be shown to manage potentially significant age-related degradation will require plant specific enhanced monitoring program to effectively manage the age-related degradation phenomena.

APPENDIX

(Nonmandatory Information)

X1. BACKGROUND ON LICENSES, MATERIALS, CRITICALITY

X1.1 Type of License

X1.1.1 The use of DCSS for the storage of SNF may be conducted under either a site-specific or general license for operating an ISFSI. From the perspective of a utility, advantages and disadvantages may be associated with each type of license.

X1.1.2 The ISFSI licenses are of two types, general and site specific. Title 10 of the Code of Federal Regulations, Part 72 (10 CFR Part 72 subpart K) grants a general license for operation of an ISFSI to power reactor sites operating under a current 10 CFR Part 50 license. The ISFSI must use only storage systems with a Certificate of Compliance (COC) issued under 10 CFR Part 72 subpart L. The 10 CFR Part 72 .236 gives requirements that must be met to obtain a COC for a cask to be used at an ISFSI. Once a cask has been approved, the NRC issues a COC to the designer/vendor. See 10 CFR Part 72, Subpart K for a list of approved casks. The license for the storage of SNF in each cask terminates 20 years after the first use of that cask, or 20 years after the COC renewal, whichever comes later (10 CFR Part 72.212(a) 3). The cask COC re-approval is governed by 10 CFR Part 72.240.

X1.1.3 A general license is issued under 10 CFR Part 72.6, Subparts A, B, and C provide the basis for issuance of a general ISFSI license. A power reactor site operating under 10 CFR Part 50 may use a general license. The site-specific application must have an approved Safety Analysis Report (SAR) that demonstrates how the requirements given in Part 72, Subpart B (License Application, Form and Content) are to be met at the ISFSI site. A general licensee may use storage systems with a COC. They may use a cask without a general license by incorporating the safety analysis in their ISFSI application, or by referencing a topical safety analysis report for that cask. The SAR may include justifications for modifying the storage system or using it outside of its original design basis, as needed by the site. The ISFSI license must be renewed within 20 years from the date of issuance, in accordance with 10 CFR Part 72.42.

X1.1.4 A Site-Specific License can be issued, for example, to a utility for a particular storage system at a specific site. This license permits the user to tailor a DCSS to particular needs (fuel type, burn-up level, thermal, or other considerations). For example, if a plant were operated under a general license that does not cover a particular fuel, one option would be to obtain a site-specific license to cover the characteristics of this fuel. The site-specific license also permits the development of alternative (such as a vault system) to the common cask designs. Site-specific licenses have been issued for use at plants with plans to decommission such as Fort St. Vrain, Rancho Seco and Trojan. Site-specific licenses have been issued to plants with no plans for decommissioning such as Oconee, Surry, Prairie Island, Calvert Cliffs, and North Anna. The options for a plant that plans to decommission are (*1*) apply for a site-specific license or, (*2*) maintain the Part 50 license and use the general license under Part 72.

X1.1.5 A General License is restricted to sites with a 10 CFR Part 50 license. A general license has been issued for each of the storage casks listed in 10 CFR Part 72.214, with each cask being approved for storage of spent fuel under the conditions specified in its COC. The main advantage associated with the use of a certified DCSS is that its use requires neither an SAR nor a new Safety Evaluation Report (SER), when the storage system is to be employed by a Part 50 licensee, that is, at a particular utility. The utility need only verify that the conditions for the general license will be satisfied at the site. For example, the site conditions must be bounded by the conditions specified in the SAR for the cask design. Thus, under the general license, no approval (or re-approval) that is specific to the site is required for the use of an approved DCSS design. A practical consequence of this it that the utility may save costs that would have been associated with having a cask design re-approved on a site-specific basis. It is reasonable to anticipate that, under future regulations related to license renewal, this same notion may become a cost saver for utilities that use cask designs that have been re-approved for continued storage in a license renewal term of 20 to 80 years beyond the initial licensing period. Another advantage of a general license is that for the general license does not require the period of public hearing that may be associated with a site-specific license application. This too may facilitate license extension, as only one re-approval may be required vis-à-vis re-approval for each site.

X1.2 Review of SNF and Dry Storage System Component Performance Requirements for License Renewal

X1.2.1 Pertinent information on the SNF, the materials used in the storage system, their functions, expected behavior, performance, etc. that was provided in the license application should be reviewed along with any data taken during the storage period to assess their significance in license extension. As with the original application, in which design events were considered, the reviewer must consider the suitability of materials for their functions as components of safety protection systems. The materials and components of principal concern are those used in safety arguments that satisfy regulatory requirements. The properties and characteristics that are important to these requirements are important to license extension and often must be demonstrated to be adequate for the remainder of the storage life cycle.

X1.2.2 Normal, Off-normal, and Accident events should be considered for the extended service period. Where historical precedents are available, they may furnish reasonable assurance that the material is suitable for the extended service of a given component in a cask application, provided that the service conditions are sufficiently similar to those of the precedent. Where such analogies are not available, the applicant must present arguments that will be assessed by the reviewer. When additional information is required, a brief literature search may suffice. When necessary, the required information must be developed, through research, monitoring, inspections, or other means.

X1.2.3 The history of the SNF should be known, and then a step-by-step procedure for review of the dry storage system under normal conditions (dry, inert environment) should be provided, and areas that need emphasis for reassessment should be discussed. Information is provided in this document for the discussion of normal conditions guidance is provided for use in evaluating SNF and materials that have undergone off-normal or accident conditions. Information for evaluating SNF is followed by information on other materials, including those used in shielding, welds, bolts, seals, O-rings and other applications. Verification of these features constitutes part of the process of assuring that the design basis can be extrapolated. A service period of up to 100 years, including the first twenty years, is assumed in this guide.

X1.2.4 *Cladding Performance Criteria*—The spent fuel cladding must be protected during dry storage against degradation that leads to gross ruptures, or the fuel must be otherwise confined so that degradation of the fuel during interim storage will not pose operational safety problems with respect to its removal from storage. Damage modes and failure mechanisms can be tolerated as long as they result in a loss of fuel rod internal pressure and termination of the damage mechanism. A small breach in the cladding, such as a pinholes or a hairline crack that would not release irradiated UO_2 particles is not be considered a gross rupture. Descriptions of selected degradation modes are given in Annex A1 and Annex A2 to serve as a guide in determining the probability of occurrence and the extent of the damage that could be anticipated.

X1.3 Maintaining Sub-Criticality

X1.3.1 The principal function of boron or other neutron absorbers is to absorb thermal neutrons, which are required for the critical multiplication of neutrons in UO_2 fuel. Since fast neutrons are not significant to this process, the designs are such that the behavior of boron-containing materials during dry storage is not important unless and until water or some other neutron moderator is present. Thus, until the fuel is placed in a situation wherein (non-borated) water engulfs the fuel, the neutron multiplication process is not an issue. When the fuel is in the presence of the water (or other) moderator, specific minimum amounts of boron (the level of which is specified as a concentration or areal density) may be required to inhibit any neutron multiplication or to limit keff to calculated values that are acceptable. This level of boron is obtained from either borates in the pool water or some form of boron in neutron absorbing plates.

X1.3.2 Factors affecting the systems ability to meet criticality safety requirements include fuel geometry, geometry of absorbing material, and water incursion (if full water flooding was not assumed in the initial safety analysis). Some DCSS for PWR fuel include no fixed neutron absorber materials in the basket, and depend entirely on water exclusion for criticality control during dry storage, and dissolved boron in the pool water for criticality control during loading and unloading.

X1.4 Supplemental Information on Effects of Temperature

X1.4.1 Under normal conditions, at long times, as temperature decreases, the wall thickness and fission gas content and pressure remain nearly constant. The strain rate tends to approach zero, and stress levels are insufficient to promote further creep.

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