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.

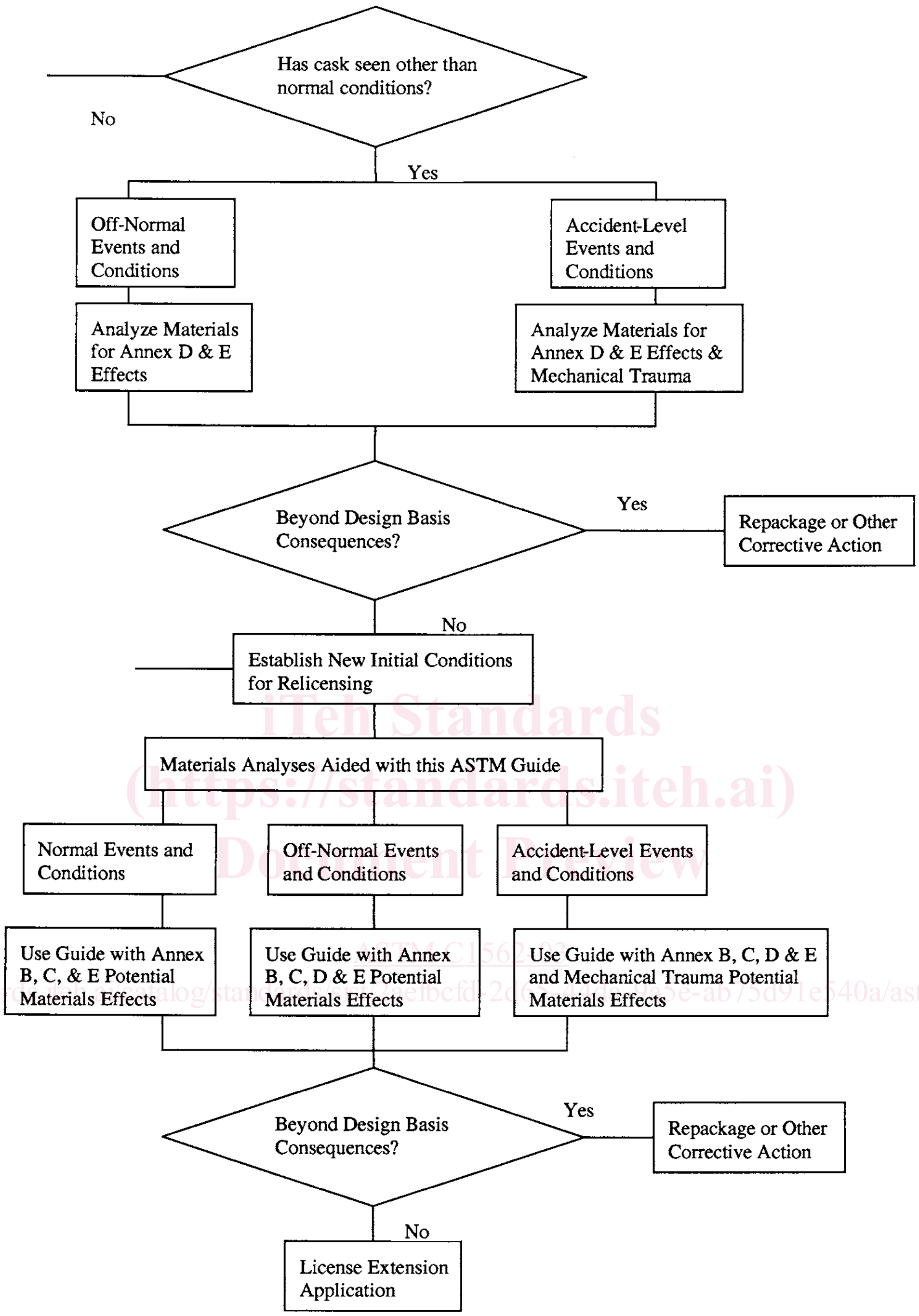

FIG. 1 Analysis Logic and Identification of Materials Conditions to be Considered in ISFSI License Renewal

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