RESPONSES TO PUBLIC COMMENTS ON NUREG–2224, “DRY STORAGE AND TRANSPORTATION OF HIGH BURNUP SPENT NUCLEAR FUEL, DRAFT REPORT FOR COMMENT”

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CHAPTER 1 - INTRODUCTION

On August 9, 2018, the U.S. Nuclear Regulatory Commission (NRC) issued for public comment NUREG-2224, “Dry Storage and Transportation of High Burnup Spent Nuclear Fuel, Draft Report for Comment” (83 FR 39475). NUREG-2224 provides technical background information applicable to high burnup spent nuclear fuel (HBU SNF), provides an engineering assessment of recent NRC-sponsored mechanical testing of HBU SNF, and proposes example approaches for use in potential licensing and certification applications for dry storage and transportation of HBU SNF. The initial comment period closed on September 24, 2018. In response to a request from numerous public interest groups, the comment period was re-opened on October 10, 2018 for a further 30 days (83 FR 50965). The second comment period closed on November 9, 2018. Comments on this proposed technical report are available electronically at the NRC's electronic Reading Room at http://www.nrc.gov/reading-rm/adams.html. From this page, the public can gain entry into Agencywide Documents Access and Management System (ADAMS), which provides text and image files of NRC's public documents. In addition, these comments may be viewed and downloaded electronically through the Federal e-Rulemaking Portal http://www.regulations.gov, Docket number NRC-2018-0066. Comments were received from the following individuals or groups:

	ADAMS No.	Commenter Affiliation	Commenter Name	Abbreviation
1	ML18264A196	Argonne National Lab Oak Ridge National Lab Pacific Northwest National Lab Sandia National Labs	Ned Larson (Point of Contact)	DOE
2	ML18264A197	Nuclear Information and Resource Service	Diane D'Arrigo	NIRS
3	ML18269A028	Nuclear Energy Institute	Rodney McCullum	NEI
4	ML18269A029	Nuclear Information and Resource Service	Diane D'Arrigo	NIRS2
5	ML18269A030	Private Citizen	Anonymous	ANON1
6	ML18269A031	Private Citizen	Anonymous	ANON2
7	ML18269A032	Deeper Look Data	Steven Olsen	DLD
8	ML18269A299	Electric Power Research Institute	Jeremy Renshaw	EPRI
9	ML18269A300	Pilgrim Watch	Mary Lambert	PW
10	ML18269A036	Hudson River Sloop Clearwater, Inc.	Manna Jo Greene	HRSC
11	ML18269A037	Private Citizen	Donna Gilmore	DG
12	ML18269A038	Private Citizen	Peter Wolf	PDW
13	ML18269A039	Private Citizen	Anonymous	ANON3

14	ML18291A670	Electric Power Research Institute	Jeremy Renshaw	EPRI2
15	ML18306A667	Private Citizen	Bert Moldow	BM
16	ML18310A188	Private Citizen	Anonymous	ANON4
17	ML18312A143	Citizens' Environmental Coalition	Barbara Warren	CEC
18	ML18313A186	Private Citizen	Dezi Bunio	DB
19	ML18313A187	Holtec International	Stefan Anton	HI
20	ML18317A283	NAC International, Inc	Wren Fowler	NAC
21	ML18317A288	Promoting Health & Sustainable Energy, LLC (PHASE)	Michel Lee	PHASE
22	ML18325A013	Private Citizen	Martin A. Rhodes	MAR
23	ML18325A014	NAC International, Inc	Wren Fowler	NAC2
24	ML18325A093	Committee to Bridge the Gap	Daniel Hirsch	CBG
25	ML19046A053	Electric Power Research Institute	Keith Waldrop	EPRI3

The staff reviewed and considered public comments in finalizing NUREG-2224.
This document places each public comment into one of the following categories:

1. Reponses to Specific Requests for Comments
2. Requests for Extension of Comment Period
3. Editorial and Clarification
4. Generic Comments
5. Specific Technical Comments

CHAPTER 2 - RESPONSES TO SPECIFIC REQUESTS FOR COMMENTS

In Section II of the Federal Register Notice (83 FR 39475; August 9, 2018), the NRC posed six questions for which it solicited stakeholder comments. Only one commenter responded to the questions. The following paragraphs outline these questions and the commenter's responses to them. The NRC appreciates the responses to these questions, and, while no specific changes to the document were made as a result of the responses to the questions, to the extent applicable, the responses were taken into consideration in finalizing NUREG-2224.

1. Are NRC's assumptions regarding the performance of other cladding alloys based on data obtained from high burnup (HBU) spent nuclear fuel (SNF) with Zircaloy-4 cladding for evaluating design-basis drop accidents reasonable? If not, please explain why not.

Commenter's Response [HRSC]: *No. Concerns raised by experts about the HBU fuel cladding do not appear to have been adequately addressed by NRC.*

NRC Response: The staff appreciates the response.

Since the commenter did not provide examples of concerns raised by experts, the NRC was unable to respond to the comment. See also responses to comments in Section 3.1 and 3.2 of this document for information supporting the safety of HBU SNF and the application of safety standards for the dry storage and transportation of HBU SNF.

2. Are the described licensing and certification approaches easy to follow and practical? If not, please explain why not.

Commenter's Response [HRSC]: *No. The described licensing and certification approaches are hard to follow given the absence of a long-term storage strategy for SNF.*

NRC Response: The staff appreciates the response.

The NRC licenses each Independent spent fuel storage installation (ISFSI) and certifies generic dry storage system (DSS) designs for specific periods of operation. More specifically, NUREG-2224 proposes example approaches for use in potential licensing and certification applications for dry storage of HBU SNF up to 60 years and transportation of HBU SNF that has been previously dry-stored up to 60 years. NUREG-2224 is informed, to the extent required by the regulations under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, by the need to consider eventual ready retrieval and disposal of HBU SNF. More specifically, the regulations require that ISFSIs and DSSs be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related Greater-than-Class-C (GTCC) waste for further processing or disposal (see 10 CFR 72.122(l)). In addition, DSS designs are required, to the extent practicable, to consider compatibility with removal of the stored spent fuel from a reactor

site, transportation, and ultimate disposition by the United States Department of Energy (DOE) (see 10 CFR 72.236(m)).

The staff further notes that the Nuclear Waste Policy Act, as amended, identifies specific roles and responsibilities for government agencies involved in the disposal of high-level waste (which includes spent nuclear fuel), including specific roles for the NRC, the Environmental Protection Agency, and the DOE. The activities related to disposal of high-level waste are outside of the scope of NUREG-2224, which only pertains to the transportation and dry storage of HBU SNF (per the regulatory requirements in 10 CFR Part 71 and Part 72, respectively).

3. Is the proposed approach for evaluation of vibration normally incident to transport clear? If not, please explain why not.

Commenter's Response [HRSC]: No. While the approach to evaluation of vibration normally incident to transport may be clear and otherwise adequate, it fails to address cases of abnormal vibration, shock or intentional attack.

NRC Response: The staff appreciates the response.

The staff agrees that the proposed approach in NUREG-2224 for the evaluation of vibration normally incident to transport is clear and adequate. The staff disagrees that NUREG-2224 does not consider abnormal vibration and shocks. Per Section 2.4.2 of the final NUREG-2224, an applicant would be expected to identify the strains expected for the transportation package being certified, which may include, for example, shocks associated to rail-car-to-rail-car coupling or any other postulated abnormal vibration or shocks during transport. The NRC also conducts a thorough security assessment to evaluate potential intentional attacks during transport. However, the evaluation of these scenarios is outside the scope of NUREG-2224.

4. Are the discussions on consequence analyses due to hypothetical fuel reconfiguration clear and meaningful? If not, please explain why not.

Commenter's Response [HRSC]: No. Again, discussions of fuel reconfiguration overlook consequences later in the fuel cycle, such as transfer, short-term storage, transportation and long-term storage.

NRC Response: The staff appreciates the response.

The example approaches provided in NUREG-2224 for use in potential licensing and certification applications are informed by all the regulatory requirements in 10 CFR Part 72 (for ISFSIs and DSS designs) and 10 CFR Part 71 (for transportation packages), and by existing safety review guidance for ISFSIs and DSS designs.

With respect to transfer and dry storage operations, the staff reviews the design basis of each ISFSI or DSS design to ensure that approved operations (as defined in the respective Final Safety Analysis Report) are performed in compliance with all pertinent regulatory requirements in 10 CFR Part 72. For additional information on the NRC's staff review guidance for ISFSIs and DSS designs, the commenter is referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776) and NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620).

With respect to transportation, the staff reviews the design basis of each package to ensure that the allowed SNF contents can be safely shipped in compliance with all pertinent regulatory requirements in 10 CFR Part 71. For additional information on the NRC's staff review guidance for transportation packages of SNF, the commenter is referred to NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No. ML003696262).

The staff also notes that NUREG-2224 is informed, to the extent required by the regulations in 10 CFR Part 72, by the need to consider eventual ready retrieval and disposal of HBU SNF. More specifically, the regulations require that ISFSIs and DSSs be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related GTCC waste for further processing or disposal (see 10 CFR 72.122(l)). In addition, DSS designs are required, to the extent practicable, to consider compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the DOE (see 10 CFR 72.236(m)).

The staff further notes that the Nuclear Waste Policy Act, as amended, identifies specific roles and responsibilities for government agencies involved in the disposal of high-level waste (which includes spent nuclear fuel), including specific roles for the NRC, the Environmental Protection Agency, and the DOE. The activities related to disposal of high-level waste are outside of the scope of NUREG-2224, which only pertains to the transportation and dry storage of HBU SNF (per the regulatory requirements in 10 CFR Part 71 and Part 72, respectively).

5. Are there any potential conflicts between NUREG-2215, Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities, Draft for Comment (Agencywide Documents Access and Management System (ADAMS) Accession No. ML17310A693) and this document? If so, please describe any conflicts.

Commenters Response [HRSC]: No answer. Clearwater has not undertaken a review of ADAMS Access No. ML17310A693.

NRC Response: The staff appreciates the response.

6. Is the NRC's reassessment of the ductility transition temperature as measured by ring compression testing of defueled HBU SNF segments reasonable? If not, please explain why not.

Commenters Response [HRSC]: No answer. Clearwater has not explored this reassessment.

NRC Response: The staff appreciates the response.

CHAPTER 3 – PUBLIC COMMENTS

1. Requests for Extension of Comment Period

The NRC received three comment submissions requesting an extension of the comment period. The comments expressed concern that the NRC was not providing members of the public sufficient time to review due to the complexity of the issue. After the first public comment period closed on September 24, 2018, the NRC agreed to reopen the public comment period for a further 30 days, from October 10, 2018, through November 9, 2018.

2. Editorial and Clarification

The NRC received several comments highlighting editorial and spelling mistakes, syntax errors and formatting issues in the document. In general, the NRC agreed with these comments and made the appropriate revisions to NUREG-2224.

The NRC also made a universal editorial revision, not related to any comments:
Revised "psi" values to be "psia" values.

3. Generic Comments

The NRC received many comments regarding high burnup (HBU) spent nuclear fuel (SNF) in general. Some comments addressed topics and issues outside the scope of NUREG-2224. The NRC staff grouped many of these comments into topics and has provided a universal response to these grouped comments.

3.1. Information Supporting Safety of High Burnup Spent Nuclear Fuel

Comment 3.1.1: *Commenters questioned the technical basis of NUREG-2224. Some of these commenters argued that NUREG-2224 is devoid of fact and ignores significant operating and experimental data related to zirconium oxides and zirconium hydrides. A commenter stated that there are too many critical unknowns about HBU SNF in dry storage and transport to have confidence it is safe. Another commenter stated that the NRC has failed to show evidence for its assumptions. Some commenters expressed the belief that the NRC's assumptions about HBU SNF are devoid of scientific bases or not reasonable and that there are critical unknowns and missing data about HBU SNF in storage and transport. A few commenters expressed concerns that many of the degradation mechanisms regarding dry storage and transport of HBU SNF are left unresolved and the NRC has made overly-optimistic assumptions. A commenter stated that the NRC does not have sufficient information to make cogent regulations concerning dry storage and transportation of HBU SNF and that there are too many "unknowns" in the document. A commenter asked if HBU SNF should be treated as damaged fuel and another*

stated that additional precautionary measures for HBU SNF should be implemented over and above what applies to low burnup spent nuclear fuel.

NRC Response: The staff disagrees with the comment.

In order to approve an ISFSI, a generic DSS design, or a transportation package, the NRC reviews the application to ensure that the design meets all applicable regulatory requirements through a robust engineering review. The review of applications for dry storage or transportation of either low burnup (i.e., less than 45 Gigawatt-day/metric ton of uranium (GWd/MTU) SNF or high burnup (i.e., greater than or equal to 45 GWd/MTU) SNF is performed with equal technical rigor. For additional information on the NRC's staff review guidance, the commenters are referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No. ML003696262).

The staff also notes that existing regulations and safety review guidance related to the dry storage and transportation of HBU SNF are based on the best available knowledge from research and operational experience. NUREG-2224 was developed to improve the existing technical basis on the performance of HBU SNF cladding, as discussed in Interim Staff Guidance 11, Revision 3 (ADAMS Accession No. ML033230335), and references therein. The improved technical basis supports the staff's conclusion that hydride reorientation will not compromise HBU SNF cladding integrity during dry storage up to 60 years and transportation of previously dry-stored HBU SNF for up to 60 years. The conclusion is supported by an engineering assessment of the results of experimental research on the mechanical performance of HBU SNF following cladding hydride reorientation, as discussed in NUREG/CR-7198, Revision 1, "Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications" (ADAMS Accession No. ML17292B057). NUREG/CR-7198, Revision 1, provides results of testing conducted at Oak Ridge National Laboratory (ORNL) to determine the ability of HBU SNF to maintain its integrity under conditions relevant to storage and transport. The intent of NUREG-2224 is to provide all stakeholders with information on the NRC's understanding of the state of knowledge of hydride reorientation and of how this information might be used in support of potential licensing and certification applications for the dry storage and transportation of HBU SNF. The NRC published the Draft Report for Comment with the explicit intention of soliciting comments "regarding errors or omissions, as well as suggestions for improvements of this NUREG."

The regulatory framework for the dry storage and transportation of HBU SNF is supported by robust regulatory guidance; voluntary domestic and international consensus standards; research and analytical studies; and processes for implementing licensing reviews, inspection programs, and enforcement oversight. As part of its regulatory program, the NRC continuously assesses the data and information relevant to the safe storage and transportation of spent nuclear fuel, as demonstrated by the HBU SNF-specific assessment provided in NUREG-2224.

Technical reports, such as NUREG-2224, support development of the NRC's evaluation of aging management programs, consideration of service life assumptions, and identification of circumstances that might require repackaging of spent nuclear fuel earlier than anticipated.

The NRC does not require that intact or undamaged HBU SNF be canned in order to meet the pertinent regulatory requirements for dry storage (per 10 CFR Part 72) and transportation (per 10 CFR Part 71) – see Interim Staff Guidance 1, Revision 2, "Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function" (ADAMS Accession No. ML071420268) for details on NRC's expectations for SNF classification. The NRC reviews each ISFSI, DSS design or transportation package for compliance with the regulations per its own design basis. The commenter is referred to response to response to Comment 3.1.7 on the technical basis for dry storage of HBU SNF.

See also responses to Comments 3.1.2 and 3.1.5 for further information.

No revisions were made to NUREG-2224 in response to the comments.

Comment 3.1.2: *Commenters expressed concern about the safety of HBU SNF, citing increased risks relative to lower burnups during transportation or storage and in spent fuel pools and dry storage systems. A commenter stated that evidence exists that moderate and high burnup fuels are unstable in storage and transport. Commenters expressed concern about the management of HBU SNF that could potentially experience damage during storage or transportation. One commenter expressed concerns about the safety of SNF during movements of lifting cranes. Another commenter expressed concerns about the safety of HBU SNF when emplaced or transported in different orientations (i.e., horizontal or vertical). Commenters expressed concern for potential safety-related and security-related accidents, including terrorist attacks, involving the storage and transportation of high-burnup fuel and how such events are accounted for in the regulations.*

NRC Response: The staff disagrees with the comment.

The NRC's current regulatory framework ensures that the risks associated with potential accidents are adequately considered to ensure that the SNF (both low and high burnup) can be safely stored and transported.

Regarding the risks associated with conditions that may be experienced during ISFSI operations (during storage of both low and high burnup), 10 CFR 72.122 defines a number of requirements for all licensees under 10 CFR Part 72 to ensure protection against environmental conditions and natural phenomena. More specifically, structures, systems, and components (SSCs) important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI to withstand postulated accidents. Further, SSCs important to safety must be designed to withstand the effects of natural phenomena such as

earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions.

In addition, the staff notes that the materials used in SSCs important to safety are evaluated to ensure they will perform adequately when exposed to the range of operating environments encountered during fuel loading, transfer, and storage operations (see 10 CFR Part 72.120(d), 72.122(b)(1) and (c), 72.124(b), 72.236(h)). This review considers the allowed contents during dry storage operations (i.e., whether it is low or high burnup fuel) in the performance of the materials used in the SSCs important to safety, and whether the SNF contents may experience reconfiguration (or damage) during the range of operating environments.

Regarding the risks associated with the transportation of SNF (both low and high burnup), NRC regulations require that transportation packages be evaluated per rigorous tests for normal conditions of transportation and hypothetical accident conditions, as defined under 10 CFR 71.71 and 10 CFR 71.73, respectively. Per 10 CFR 71.73, each transportation package for SNF (both low and high burnup) is evaluated for hypothetical accident conditions based on sequential application of multiple tests to determine their cumulative effect on a package or array of packages, including an evaluation of the effects of a free drop of the package through a distance of 9 m (30 ft) (see 10 CFR 71.73(c)(1)) and a fire test with an average flame temperature of at least 800°C (1475°F) (see 10 CFR 71.73(c)(4)). The rigorous test sequence provides reasonable assurance that, in the unlikely event that an accident was to occur during transport, the public health and safety will be protected.

In addition, the staff notes that the materials used in the fabrication of transportation packages are evaluated to ensure they will perform adequately when exposed to the range of operating environments encountered (see 10 CFR 71.31(a)(2), 71.35(a), 71.43(d) and (f), 71.51(a)(1), 71.55(b)(1), 71.55(d)(3), 71.55(e)(1)(2) and (f), 71.87(a through c, f and g)). This review considers the allowed contents for the transportation package (i.e., whether it is low or high burnup fuel) in the performance of the package materials, and whether the SNF contents may experience reconfiguration (or damage) during the range of operating environments.

The staff also reviews the movements of dry storage systems and transportation packages by lifting cranes as part of its safety review. Operations have typically been evaluated for design-basis drops of six inches or more. For higher lifts, the defense-in-depth approach to load handling established by NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants: Resolution of Generic Technical Activity A-36” (ADAMS Accession No. ML070250180), provides reasonable assurance that more energetic drops would be avoided.

It is also noteworthy that the NRC has previously evaluated the radiological impact of a confinement breach in a DSS. In 1988, the NRC analyzed a postulated accident involving the removal of the lid of a DSS with the assumption that all the fuel rods were damaged (NUREG-1140, "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees, Final Report," ADAMS Accession No. ML062020791). The analyses considered the release of krypton and iodine gases and reported resultant doses that

were below the Environmental Protection Agency's protective action guidelines (see "PAG Manual: Protective Action Guides and Planning Guidance for Radiological Incidents," Environmental Protection Agency, EPA-400/R-17/001, January 2017, http://www.epa.gov/radiation/protective-action-guides-pags)[1] for taking protective action after an accident. In later studies published in 2007 and 2014, the NRC also calculated the radiological risks of spent fuel dry storage and transportation, as discussed in NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," (ADAMS Accession No. ML071340012) and NUREG-2125, "Spent Fuel Transportation Risk Assessment," (ADAMS Accession No. ML14031A323), respectively. Those analyses considered the post-accident release of gases and particulates from breached DSSs containing damaged fuel assemblies. More specifically, the analyses considered the portion of gases and particulates that would be released from the breached DSSs (i.e., the release fractions). In these studies, the sizes of the breaches in the primary confinement/containment following an impact accident were either calculated or assumed to allow for radionuclide release. These studies concluded that the risks of dry storage and transportation are low. Though these analyses did not specifically address HBU SNF, Sections 3.2.2 and 4.2.2 of NUREG-2224 consider bounding release fractions (source term) from HBU SNF as determined through modeling and available experimental data. Based on those release fractions, the NRC considers the risk assessments discussed in NUREG-1864 and NUREG-2125 are generally applicable to HBU SNF. The staff continues to monitor ongoing experimental and modeling work in this area.

Regarding the security of spent fuel, the NRC regulations in 10 CFR Parts 72 and 73 provide security requirements for physical protection of spent fuel storage and transportation. An ISFSI licensee must comply with these security requirements in 10 CFR Parts 72 and 73; how a licensee complies is described in the licensee's Physical Security Plan. The NRC reviews and approves each licensee's Physical Security Plan in evaluating the adequacy of the licensee's on-site security measures. The NRC also inspects ISFSIs to ensure the licensees' complete and correct implementation of the features of the approved Physical Security Plan.

The NRC has concluded that there is reasonable assurance that a terrorist attack would not lead to a significant radiological event at an ISFSI. The bases for this conclusion include: (1) the NRC's continual evaluation of the threat environment, in coordination with the intelligence and law enforcement communities, which provides, in part, the basis for the protective measures currently required by applicable regulations and orders; (2) the protective measures that are in place to reduce the chance of an attack that leads to a significant release of radiation; (3) the robust design of dry storage systems, which provides substantial resistance to penetration; and (4) the NRC security assessments of the potential consequences of terrorist attacks against ISFSIs, that inform the decisions made regarding the types and level of protective measures. Since dry storage began in the United States, there have been no known or suspected attempts to sabotage, or to steal, radioactive material in dry storage systems at

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[1] The reference pertains to the current, 2017, revision of the protective action guidelines from the Environmental Prorection Agency. The analyses in NUREG-1140 are based on a 1988 revision of the protective action guidelines, which have the same maximum allowable doses as the current, 2017, revision.

ISFSIs, or to directly attack an ISFSI. Nevertheless, the NRC is continually evaluating the threat environment, to determine whether any specific threat to ISFSIs exists. The results of these security assessments contain sensitive unclassified information and therefore are not publicly available.

Regarding the security of SNF transportation packages, NRC regulations in 10 CFR Part 71 and Department of Transportation (DOT) regulations (49 CFR Parts 171 – 178) provide for rigorous standards for the design and construction of transportation packages to ensure safe and secure shipment of radioactive material. After September 11, 2001, the NRC issued Orders to licensees requiring increased security for the transportation of specific types of radioactive materials, including spent nuclear fuel. The NRC incorporated these Orders and other additional security requirements into a 2013 rulemaking (78 FR 29520) that amended the NRC regulations in 10 CFR Part 73 for the physical protection of irradiated reactor fuel in transit. NRC regulations in 10 CFR Parts 71 and 73 and DOT regulations in Title 49 of the Code of Federal Regulations (49 CFR) Parts 107, 180, 390, and 397, as appropriate to the mode of transport, require spent nuclear fuel shippers to use approved routings and to implement safeguarding measures, including the use of armed escorts and emergency response plans. (See also response to Comment 3.3.1, which provides references for analyses that consistently show that the accident risks from transportation of spent nuclear fuel are extremely low). All these regulations are applicable to the transportation of HBU SNF.

No revisions were made to NUREG-2224 in response to the comments.

Comment 3.1.3: *Commenters expressed concern that canisters providing primary confinement in dry storage systems (DSS) cannot be regularly inspected for degradation and that canister integrity may be compromised because of the lack of regular inspections. Some commenters requested the NRC require more robust DSS canisters and casks so spent nuclear fuel and its confinement subcomponents can be inspected and maintained. Other commenters expressed concern regarding DSS canisters potentially developing cracks during operations. A commenter requested the NRC provide detailed information about the NRC's specific requirements related to the monitoring of helium in dry storage.*

NRC Response: The staff notes the comment.

Dry storage systems (DSSs) are required to be monitored using inspections, tests, or other means to demonstrate that safe storage conditions are maintained (see 10 CFR 72.122(f)). To meet this requirement, maintenance programs typically include monitoring activities (such as radiation, pressure, and temperature monitoring); periodic visual inspections of accessible surfaces for defects that could reduce confinement effectiveness; periodic visual inspections of air flow vents for blockages that could reduce thermal performance; and other testing, as applicable, to verify that the radiation shielding, thermal, and confinement capabilities of the DSS are maintained (see NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620) and

NUREG-1567, “Standard Review Plan for Spent Fuel Dry Storage Facilities” (ADAMS Accession No. ML003686776)).

To date, specific licenses for ISFSIs and Certificates of Compliance (CoCs) for DSSs have been issued for initial terms of 20 years, although current NRC regulations allow for an initial 40-year storage period. Specific licenses and CoCs can be renewed for additional terms not to exceed 40 years, with appropriate aging management programs (AMPs) in place to detect the effects of aging and ensure that adequate corrective actions are initiated to maintain the safety functions of the DSSs. Similar to the approach used for nuclear power reactor high pressure piping, any identified areas of canister degradation are expected to trigger additional examinations that characterize the extent and severity of degradation. Both specific and general licensees must either demonstrate that the DSSs continue to fulfill their safety functions or pursue repair or other options to ensure that the spent fuel is safely confined, and all other pertinent regulatory requirements are met.

No revisions were made to NUREG-2224 in response to the comments.

Comment 3.1.4: *Concerns were also expressed about the safety functions of damaged fuel cans (DFCs), stating that these don’t fully replace the cladding as a barrier and thus do not confine the release of fission gases or volatiles. A commenter noted that DFCs are vented on both ends and questioned whether the safety analyses for DFCs consider the releases of fission gases from SNF during dry storage operations. The commenter also stated that the NRC should require the use of helium-filled sealed tubes for the dry storage of damaged SNF.*

NRC Response: The staff notes the comment**.**

Cans used for damaged fuel (DFCs) are typically metal enclosures sized to confine a single assembly and are generally placed in a limited number of positions inside the DSS canister/cask or transportation package containment system. The use of these cans provides reasonable assurance that fuel-specific and system-related requirements—as discussed in Interim Staff Guidance 1, Revision 2, “Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function” (ADAMS Accession No. ML071420268)—are satisfied. Consistent with Interim Staff Guidance 1, Revision 2, fuel-specific requirements include those specified in 10 CFR 72.122(h)(1), 72.122(l) (for dry storage), and 10 CFR 71.33, 71.55(d)(2) (for transportation). Further, system-related requirements include those specified in 10 CFR 72.124(a), 72.128, 72.236(h),(m) (for dry storage) and 10 CFR 71.55(e)(1), 71.71, 71.73 (for transportation). Compliance with these requirements provides reasonable assurance that all intended functions of the DSS design or transportation package (i.e., criticality safety, radiation shielding, confinement/containment, heat removal, structural performance, and retrievability, if applicable) are maintained.

The safety review of a DSS design or transportation incorporating the use of a DFC considers the specific design and features of the DFC, including whether the DFC is vented. The NRC

does not prescribe specific designs for DFCs, but defines performance-related requirements in its regulations. Therefore, the NRC evaluates each DFC on a case-by-case basis to ensure compliance with the appropriate regulations (including the aforementioned fuel-specific and system-specific regulatory requirements).

The staff also notes that the safety review of a DSS canister conservatively assumes the entire failure of all SNF rods in the DSS canister (including those in DFCs), which provides reasonable assurance that the DSS confinement boundary is maintained during normal, off-normal and accident conditions of storage (even if there were to be releases of gases from the SNF rods to the inside of the DSS canister).

No revisions were made to NUREG-2224 in response to the comments.

Comment 3.1.5: *A commenter identified statements made in a 2010 US Nuclear Waste Technical Review Board (NWTRB) Report ("Evaluation of the Technical Basis for Extended Dry Storage and Transportation of Used Nuclear Fuel") that, in the commenter's opinion, document a "fundamental lack of knowledge available" regarding the long-term storage of high burnup spent nuclear fuel (HBU SNF) and questioned whether the NWTRB's recommendations for filling these knowledge gaps are being adequately addressed. Another commenter inquired regarding the data available for HBU SNF when NUREG-2224 was prepared.*
Several commenters also expressed concerns that the cladding oxide and cladding hydrogen absorption (the latter identified in the 2010 NWTRB report) are ignored by the NRC.

NRC Response: The staff disagrees with the comment.

The NRC disagrees that there is a fundamental lack of knowledge supporting the safety of HBU SNF, and that the condition of SNF cladding (e.g., oxidation, hydriding) is ignored in the safety review of applications for dry storage and transportation of HBU SNF.

The NRC's regulatory framework for storage is supported by well-developed regulatory guidance; voluntary domestic and international consensus standards; research and analytical studies; and processes for implementing licensing reviews, inspection programs, and enforcement oversight. NUREG 2224 considers a wide range of recent studies and activities evaluating HBU SNF, including reports issued by the NRC, such as NUREG/CR-7198, Revision 1, "Mechanical Fatigue Testing of High-Burnup Fuel for Transportation Applications," (ADAMS Accession No. ML17292B057); and NUREG/CR-7203, "A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages" (ADAMS Accession No. ML15266A413).

Additionally, there are other activities and published information that provide support for NRC's regulatory framework that are not related exclusively to HBU SNF, such as:

1) The Extended Storage Collaboration Program (ESCP) has produced several relevant documents including:
    - "Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless Steel Canisters for Dry Cask Storage Systems," EPRI-3002005371, Electric Power Research Institute, 2015;
    - "Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless Steel Canisters," EPRI-3002008193, Electric Power Research Institute, 2017;
    - "Dry Canister Storage System Inspection and Robotic Delivery System Development," EPRI-3002008234, Technical Update, Electric Power Research Institute, May 2016; and
    - "Flaw Growth and Flaw Tolerance Assessment for Dry Cask Storage Canisters," EPRI-3002002785, Technical Update, Electric Power Research Institute, October 2014;

2) NUREG-2214, "Managing Aging Processes in Storage (MAPS) Report, Draft Report for Comment" (ADAMS Accession No. ML17289A237)

3) The Institute of Nuclear Power Operators (INPO) has established a new database: AMID (Aging Management INPO Database) to collect and disseminate operating experience of aging of dry storage systems.

4) The NRC is currently following Temporary Instruction TI 2690/011, "Review of Aging Management Programs at Independent Spent Fuel Storage Installations" (ADAMS Accession No. ML17167A268) to inspect licensees' implementation of AMPs. TI 2690/011 will inform the development of a new Inspection Procedure to inspect AMP implementation.

Due to uncertainties associated with potential aging effects, the NWTRB report did identify areas for further research to confirm that spent fuel can be safely stored at ISFSIs for extended periods. However, as noted by the references to reports and activities defined above, significant work has been completed since the issuance of that report nine years ago. Additional work continues, both nationally and internationally, for enhancing the understanding of the degradation of dry storage systems – both through experimental testing as well as through the collection of operating experience during operations. Enhancing the understanding of potential degradation mechanisms associated with dry storage systems is consistent with NRC's role as an effective regulator.

The NRC will continue its regulatory control and oversight of spent fuel storage at both operating and decommissioned reactor sites under both specific and general licenses issued under 10 CFR Part 72. Decades of operating experience and ongoing NRC inspections demonstrate that licensees continue to adequately meet their obligation to safely store the spent fuel in accordance with the NRC's regulatory requirements. If the NRC were to find a noncompliance with these requirements or to otherwise identify a concern with the safe storage

of the spent fuel, it would evaluate the issue and take necessary action, including potential revisions to its regulatory framework to protect the public health and safety and the environment.

The staff notes that the oxidation and hydriding of HBU SNF during reactor operations and its impacts to dry storage and transportation are well addressed in the current staff review guidance (see NUREG-1567, “Standard Review Plan for Spent Fuel Storage Facilities” (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility” (ADAMS Accession No. ML101040620), and NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear fuel” (ADAMS Accession No. ML003696262)).

For additional details, the commenters are also directed to the discussions on cladding oxidation and hydriding in NUREG-2214, “Managing Aging Processes in Storage, Draft Report for Comment” (ADAMS Accession No. ML17289A237). Chapter 1 of NUREG-2224 also provides a historical discussion of the evaluation of SNF cladding performance in dry storage and transportation, and an adequate discussion on the considerations of oxidation and hydriding in HBU SNF cladding. NUREG-2224 provides appropriate references in those discussions.

No revisions were made to NUREG-2224 in response to the comments

Comment 3.1.6: *Commenters expressed concerns about pyrophoricity and fire hazards related to spent fuel in dry storage systems.*

NRC Response: The staff notes the comment.

<u>Regarding Fire Hazards</u>
The staff reviews the design bases of an ISFSI or DSS design to ensure that materials will remain stable, meaning they do not adversely lead to the generation of flammable gases (i.e., hydrogen generation due to radiolysis in confined spaces), and that pyrophoric reactions are not credible during transport and storage operations. Two inherent requirements for pyrophoricity of the spent fuel in the confinement cavity (where the fuel is contained) are (1) the need for an oxidizer/flammable compound at sufficiently high concentrations and (2) a source of ignition, neither of which is present under normal and off-normal conditions of storage. The staff has developed safety review guidance to ensure these two conditions are not present in the transportation package or dry storage system (see NUREG-1567, “Standard Review Plan for Spent Fuel Storage Facilities” (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility” (ADAMS Accession No. ML101040620), and NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel” (ADAMS Accession No. ML003696262)).

In particular, the NRC staff reviews operations used for draining and drying the cask/canister to ensure the relevant Safety Analysis Reports describe adequate procedures for reducing

oxidizing compounds in the confinement cavity to an acceptable level. The staff has accepted vacuum drying methods comparable to those recommended in PNL-6365 (Knoll, R.W. and E.R. Gilbert, “Evaluation of Cover Gas Impurities and Their Effects on the Dry Storage of LWR Spent Fuel,” PNL 6365, Pacific Northwest Laboratory, Richland, WA, 1987). PNL-6365 evaluates the effects of oxidizing impurities on the dry storage of light-water reactor spent fuel and recommends limiting the maximum quantity of oxidizing gasses (such as O_2, CO_2, and CO) to a combined total of 1 gram-mole per cask. Moisture removal is inherent in the vacuum drying process, and moisture levels at or below those evaluated in PNL-6365 (about 0.43 gram-mole H_2O) are expected if the cask/canister is drained of as much water as practicable and evacuated to less than or equal to a pressure of 4.0×10^2 pascals (Pa) (5.8×10^2 pounds per square inch absolute (psia)). After evacuation, adequate removal of moisture and other oxidizing gases is verified by maintaining a constant pressure over a period of 30 minutes without vacuum pump operation.

The staff has also approved the use of forced helium dehydration for the drying of SNF (Singh, K., “Forced Gas Flow Canister Dehydration,” United States Patent 7,096,600 B2. August 29, 2006). The adequacy of drying is verified using specific procedures. For example, in order to meet the acceptance criterion for the forced helium dehydration system either gas temperature exiting the demoisturizer must be less than -6.1 °C (21 °F) for a minimum of 30 minutes or the gas dew point exiting the multipurpose canister must be less than -5.2 °C (22.9 °F) for a minimum of 30 minutes (Holtec International, “Certificate of Compliance No. 1040 Appendix A: Technical Specifications for the Hi-Storm UMAX Canister Storage System,” January 6, 2017 (NRC ADAMS Accession No. ML16341B100).

Once the drying operation is complete (by either standard vacuum drying or forced helium dehydration), the confinement cavity is then backfilled (overpressurized above atmospheric pressure) with a highly pure, inert gas (e.g., helium) for applicable pressure and leak testing. Care is taken to preserve the purity of the cover (inert) gas and, after backfilling, the cover gas purity is verified by sampling. The leak tightness of the canister is then verified by helium leak testing per American National Standards Institute (ANSI) 14.5 (ANSI N14.5, “American National Standard for Radioactive Materials — Leakage Tests on Packages for Shipment,” American National Standards Institute, Inc. New York, NY, 1997).

An evaluation conducted by the Center for Nuclear Waste Regulatory Analyses (CNWRA) (Jung et. al, "Extended Storage and Transportation: Evaluation of Drying Adequacy” (ADAMS Accession No. ML13169A039)) has indicated that most of the hydrogen generated during dry storage occurs due to radiolysis of any residual water. Some of the hydrogen generated will be absorbed by the cladding as hydrides. The CNWRA evaluation of the potential for a flammable hydrogen gas concentration was based on assumed amounts of residual water remaining inside a spent fuel storage system after drying and backfilling with helium. This included free water either as liquid water such as water trapped in a damaged fuel rod or water vapor as well as chemisorbed water such as water in a hydrated metal oxide. The CNWRA evaluation showed that for 1 atmosphere helium backfill pressure (14.7 psia), a flammable hydrogen concentration could occur with a residual water amount of 17.4 moles (313 grams of water). With 5.5 moles of

water (99 grams), no flammable hydrogen concentration was expected. When the helium-backfill pressure was increased to 5 atmospheres (73.5 psia), no flammable hydrogen concentration would occur even with a residual water amount of 55 moles of water (990 grams).

According to the evaluation by the CNWRA, the minimum helium backfill pressure of a representative DSS design would prevent the formation of a flammable hydrogen concentration unless the residual water remaining after drying was greater than 17.4 moles (313 grams). Recent analyses of gas samples by Sandia National Laboratories (Bryan, C.R., R.L. Jarek, C. Flores, and E. Leonard, “Analysis of Gas Samples Taken from the High Burnup Demonstration Cask,” SAND2019-2281, Sandia National Laboratories, Albuquerque, NM, February 2019) show that residual water after spent fuel drying was approximately 100 g (5.55 moles) or just slightly more than the 5.5 mol amount of residual water estimated in the CNWRA study which did not result in the formation of a flammable hydrogen mixture.

Because the amount of residual water remaining in a dry storage cask is low, hydrogen produced by the radiolysis of the residual water will not exceed 4%, which is the lower flammability limit for hydrogen (Airgas Incorporated, “Material Safety Data Sheet for Hydrogen and Liquefied Hydrogen,” Document #001026, Radnor, PA: Airgas, Inc. October 22, 2002). Therefore, hydrogen generation from residual water poses no credible flammability risk during dry storage of SNF.

<u>Regarding pyrophoricity hazards:</u>
The staff also recognizes that commercial nuclear fuel is manufactured as uranium dioxide (UO_2) pellets enclosed in a zirconium-alloy cladding. These materials are non-pyrophoric at room-temperature prior to their use in a reactor. Upon discharge from the reactor, spent fuel remains non-pyrophoric as long as decay heat is adequately removed, and proper measures are taken for controlling the chemistry of the fuel environment (in either wet or dry storage). The staff recognizes that the UO_2 fuel pellets in commercial spent fuel may be fractured. However, in addition to the cask/canister confinement, the cladding serves as an initial defense-in-depth confinement barrier for preventing the release of pellet fragments. Per Interim Staff Guidance 1, Rev. 2, “Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function” (ADAMS Accession No. ML071420268), spent fuel that has been classified as damaged for storage must be placed in a can designed for damaged fuel, or in an acceptable alternative. The cladding intact/undamaged fuel is also passivated by an oxide layer, which limits further oxidation of the fuel pellets. Even in the event that small particulates were to be released into the confinement cavity through a non-gross breach (i.e., defined as a pinhole leak or hairline crack), as stated before, the absence of oxidizing species and a source of ignition prevents any pyrophoric concern during dry storage. This is supported by operating experience and loading of almost 3000 dry storage systems (for both low burnup and HBU SNF) in the United States without any pyrophoric reaction ever being observed.

In summary, the NRC’s review examines the use of defense-in-depth approaches to ensure that the spent fuel will be maintained in an analyzed configuration and the conditions for pyrophoric reactions will not be present during dry storage operations. These include: (1) ensuring that

materials used in the dry storage system do not result in the generation of flammable gases; (2) adequate drying processes for limiting the presence of oxidizers/flammable gases in the confinement cavity; (3) over-pressurization of the canister/cask with a highly pure, inert gas to mitigate the entry of oxidizing gases into the canister/cask; and (4) ensuring cladding integrity or canning of grossly-breached spent fuel to maintain a known fuel configuration.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.1.7: *Various commenters questioned the technical basis for NRC's reasonable assurance that HBU SNF is safe for storage and transport (including HBU SNF dry stored up to 60 years and HBU SNF dry stored beyond 60 years). One commenter asserted that the NRC's Continued Storage of Spent Nuclear Fuel GEIS concluded safety would be maintained indefinitely.*

Various commenters also questioned what is the technical basis for the NRC's safety review guidance for peak (maximum) cladding temperatures in SNF cladding.

Another commenter also requested clarification on the technical basis for the minimum acceptable cooling time for HBU SNF prior to dry storage and requested clarification on NRC's inerting atmosphere requirements.

NRC Response: The staff notes the comment.

The technical bases for the NRC's reasonable assurance for the safe dry storage and transport of HBU SNF aged up to 60 years is delineated in the licensing or certification basis for each approved transportation package, DSS design, or ISFSI. The licensing or certification basis includes, but is not limited to, applicable NRC regulations and appendices; orders; license or Certificate of Compliance (CoC) conditions; exemptions; technical specifications; and design basis information as documented in the Final Safety Analysis Report for the transportation package, DSS design, or independent spent fuel storage installation. The NRC has developed Standard Review Plans (SRPs) and Interim Staff Guidance for the review of the design basis information to ensure compliance with 10 CFR Part 71 and Part 72 requirements. These SRPs, Interim Staff Guidance, and references therein, as well as other regulatory guides and technical reports, are used by the staff to ensure the safety review is comprehensive and to reach reasonable assurance of adequate protection of the public health and safety in the dry storage and transport of spent fuel (see NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No ML003696262)).

The staff notes that the technical basis in support of the NRC's safety review guidance for peak (maximum) cladding temperatures in SNF cladding may be found in Interim Staff Guidance-11

Revision 3 (ADAMS Accession No. ML033230335). The staff also notes that the licensing or certification basis for each approved transportation package, DSS design, or ISFSI, defines the minimum cooling time and inerting atmosphere requirements for the approved SNF contents. The staff reviews the safety of the ISFSI, DSS design or transportation package, per the applicant-defined minimum cooling time and inerting requirements (which are defined in the Technical Specification of the ISFSI, DSS, or transportation package). The NRC provides oversight to ensure that loadings by the licensees comply with these Technical Specifications.

Per the technical basis discussed in the NRC's safety review guidance (including that in Interim Staff Guidance 11, Revision 3 (ADAMS Accession No. ML033230335)), the NRC currently has an adequate technical basis that provides reasonable assurance for the safe dry storage of HBU SNF up to 60 years. This technical basis has provided the basis for renewals of ISFSIs and DSSs up to 60 years. Although the NRC has concluded that the dry storage of HBU SNF can be conducted safely, the staff has recognized that additional research on the performance of HBU SNF should be monitored and assessed by licensees and CoC holders to ensure that corrective actions are taken if future research results indicate that the design basis of the ISFSI or DSS may not be maintained due to aging of the HBU SNF (see Appendices B and D, NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" (ADAMS Accession No. ML16179A148)). Therefore, the U.S. DOE in conjunction with industry is currently conducting additional research. The research is designed to ensure that the data supporting the existing technical basis for safe dry storage of HBU SNF (e.g., short-term accelerated testing) is accurate as the fuel gets older. The NRC expects that these results will confirm that the HBU SNF will remain safe for transport even after extended storage, as the existing technical basis considers all credible aging effects during dry storage up to 60 years.

More specifically, the DOE is sponsoring two research programs on HBU SNF. The first, the High Burnup Dry Storage Cask Research and Development Project, run jointly with the nuclear industry with regulatory oversight by the NRC, is currently underway. In this study, HBU SNF was loaded into a dry storage system (referred to as the Research Project Cask) fitted with instruments to provide temperature readings and allow sampling of the gas inside. These readings, combined with tests on the fuel assemblies and inspection of the cask's interior after years of dry storage, will enhance our understanding of what happens to HBU SNF in a storage system as it cools over time. Additional information on this program may be found at https://www.energy.gov/ne/downloads/high-burnup-dry-storage-cask-research-and-development-project-final-test-plan.

The second DOE-sponsored program, the High Burnup Spent Fuel Data Project - Sister Rod Test Plan, focuses on the characteristics, material properties, and performance of HBU SNF rods with similar irradiation histories to those loaded in the Research Project Cask under the other DOE-sponsored program. These "sister rods" will be tested against the conditions as measured in the Research Project Cask as well as against conditions modeled for other loaded DSSs with different thermal profiles and histories. All this work will help system designers, users, and regulators better understand how to ensure HBU SNF will remain safe in dry storage

over the long term and during eventual transportation to a centralized storage or disposal facility. Additional information on this program may be found at https://www.energy.gov/ne/downloads/high-burnup-spent-fuel-data-project-sister-rod-test-plan-overview.

The staff notes that NUREG-2224 does not specifically address dry storage and transportation for HBU SNF dry-stored longer than 60 years. Regarding the Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel (GEIS) (NUREG-2157, Volume 1, 2014, ADAMS Accession No. ML14196A105), because the timing of repository availability is uncertain, the GEIS analyzed potential environmental impacts over three possible timeframes: a short-term timeframe, which includes 60 years of continued storage after the end of a reactor's licensed life for operation; an additional 100-year timeframe (i.e., 60 years plus 100 years) to address the potential for delay in repository availability; and a third, indefinite timeframe to address the possibility that a repository never becomes available. Environmental Impacts of Continued Storage over the three possible timeframes are summarized in Section 8.1 of the GEIS.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.1.8: *A commenter commended the efforts of the NRC and national laboratories to contribute data to the state of engineering knowledge about the behavior of HBU SNF. Another commenter stated that Draft NUREG-2224 successfully captures much of the scientific and technical information that has been acquired through the collective efforts of the NRC, the DOE, various national laboratories, and the Electric Power Research Institute (EPRI). The same commenter also stated: "It is particularly important that the extensive technical basis that now exists on HBU SNF – which demonstrates that safety margins for the storage and transportation of HBU SNF are significantly greater than previously understood – be deployed in a timely manner to support more efficient and better risk-informed management of used fuel. This information is especially valuable towards expediting the safe decommissioning of the increasing number of shutdown nuclear plants, by more expeditiously transferring all fuel to dry storage, and facilitating the near-term transportation of used fuel to the two interim storage sites now actively seeking NRC licenses."*

NRC Response: The staff notes the comment.

The staff will also continue to monitor the efforts of the U.S. DOE's national laboratories to contribute data to the state of engineering knowledge about the behavior of HBU SNF.
No revisions were made to NUREG-2224 in response to the comments.

3.2. Application of Safety Standards

Comment 3.2.1: *Commenters expressed concern regarding the application of safety standards for transporting spent nuclear fuel to interim storage facilities and expressed concern that safety standards are being 'relaxed.' One other commenter expressed concern that radiation exposures along transportation routes would impact particularly vulnerable individuals (e.g., pregnant women, infants, and children). Another commenter requested the NRC do everything in its power to ensure public health and safety and to protect the environment and existing infrastructure. That same commenter also asked the NRC to not dismiss or minimize the complexities of nuclear waste storage or transport and the perceived risks associated with high burnup fuel.*

NRC Response: The staff disagrees with the comment.

NUREG-2224 does not relax the safety standards for the transportation of SNF per the regulatory requirements in 10 CFR 71. Further, in the development of NUREG-2224, the NRC did not dismiss or minimize the complexities of dry storage and transportation of HBU SNF, nor the importance of the subject to our stakeholders. As discussed in the responses to Comment 3.1.1 and Comment 3.1.2, NRC's current regulatory framework, together with the licensee's compliance with the regulations and NRC's oversight, ensures that the SNF can be safely stored and transported. NUREG-2224 does not make any changes to 10 CFR Part 71's regulatory requirements for transportation of SNF. NUREG-2224 explicitly states that "[n]nothing contained in the report is to be construed as having the force or effect of regulations". The report is intended to provide all stakeholders with information on the NRC's understanding of the state of knowledge concerning hydride reorientation and of how this information might be used in support of potential licensing and certification applications for dry storage and transportation of HBF SNF. The NRC published the Draft Report for Comment with the explicit intention of soliciting comments "regarding errors or omissions, as well as suggestions for improvements of this NUREG."

The NRC recognizes that particularly vulnerable individuals could potentially be present along transportation routes, however, NRC's regulations ensure adequate protection of the public, including those individuals that are considered more vulnerable. The NRC based its dose limits and calculations on a descriptive model of the human body referred to as the "standard man," but has always recognized that these limits must be informed and adjusted in some cases for other factors (e.g., age and gender). NRC dose limits are also much lower for members of the public, including children and elderly people, than for adults who receive radiation exposure as part of their occupation (e.g., nuclear power plant personnel). Additional information is also available on the NRC's website, at http://www.nrc.gov/about-nrc/radiation/rad-health-effects.html.

No revisions were made to NUREG-2224 in response to the comments.

Comment 3.2.2: *A commenter inquired as to the length of time a spent fuel canister can be certified and asserted that canisters in dry storage systems can only be certified for a total of up to 60 years (i.e., an initial 20 years with a 40-year renewal).*

NRC Response: The staff disagrees with the commenter's assertion that dry storage systems can only be approved for a total of up to 60 years. NRC regulations for dry storage allow for approval of an ISFSI specific license or CoC of a dry storage system for initial and renewal periods of up to 40 years, each. The current regulatory framework for dry storage of spent fuel also allows for multiple license and CoC renewals if an applicant can adequately demonstrate compliance with the pertinent 10 CFR Part 72 requirements during the requested period of operation. Therefore, approvals are not limited to a maximum of 60 years per the current regulations, as suggested by the commenter.

The NRC further notes that the dry storage of spent fuel by both specific and general licensees remains subject to ongoing NRC inspection and enforcement to ensure continued compliance with the pertinent 10 CFR Part 72 requirements.

The staff revised the definition for "Renewal of a license or CoC (dry storage)" in the Glossary of NUREG-2224 to be consistent with the definition provided in NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" (ADAMS Accession No. ML16179A148). To clarify that the current regulatory framework for storage of spent fuel allows for multiple license renewals, subject to aging management analysis and planning, the following text has also been added: "The current regulatory framework for storage of spent fuel allows for multiple license or CoC renewals, subject to aging management analysis and planning."

Comment 3.2.3: *A commenter expressed concern that the NRC has allowed exemptions for "incorrectly packed Holtec canisters" at more than one nuclear power plant site that add to the "dangers and unknowns" for safe storage of spent nuclear fuel.*

NRC Response: The staff disagrees with the comment. The NRC has a robust inspection and oversight program to identify any misloads that have occurred at an ISFSI. The NRC conducts inspections and provides oversight during mock testing and actual loadings to verify that dry storage systems (DSSs) are loaded according to the respective license or CoC Technical Specifications. The NRC also verifies that proper loading of the DSSs is done by either direct observation during the loading or by reviewing records of how the SNF is selected for each DSS, including procedures governing the selection and verification of SNF assemblies prior to each loading. The NRC conducts the oversight review based on the maximum or allowable design values, as approved by the staff per the safety review of the ISFSI or DSS design. It is the licensee's responsibility to ensure compliance with the specific license or CoC Technical Specifications (e.g., that head load values are properly calculated to comply with the Technical Specifications). If a licensee has not loaded a DSS according to the license or CoC Technical Specifications and opts to request an exemption from an NRC regulation, the staff performs a

robust safety review of the request to ensure adequate protection of public health and safety, and the environment.

No revisions were made to NUREG-2224 in response to the comment.

3.3. Transportation

Comment 3.3.1: *Commenters expressed concern that the nation's transportation infrastructure is not capable of providing for safe transportation of spent nuclear fuel due to issues such as backlog of rail freight improvements; uncertainty in funding; lack of training for state inspection officials and emergency responders; deteriorating roads, rails, and bridges.*

NRC Response: The NRC disagrees with the comment. The NRC recognizes the public's concerns about aging of transportation infrastructure and the associated challenges. While infrastructure challenges are outside the scope of NUREG-2224 and beyond the scope of NRC's regulatory authority, the NRC is confident that radioactive materials can be transported safely based on existing safety practices and regulations.

In the unlikely event an accident does occur, the defense-in-depth framework of NRC and DOT safety requirements in 10 CFR Parts 71 and 73, and 49 CFR Parts 107,171-180, 390-397, as appropriate to the mode of transport (including testing and approval of packaging, proper placarding and labeling, and limiting the dose rate from packages and conveyances) provide adequate protection of the public to limit the potential consequences of an accident. The transportation risk analyses conducted to date have used state-of-the-art methods to account for the probability of accidents of different severities and the response of a transportation package under modeled accident conditions (see below references). In particular, regarding potential infrastructure effects, the most recent transportation risk analysis, conducted by the NRC (NUREG–2125, "Spent Fuel Transportation Risk Assessment," (ADAMS Accession No. ML14031A323)) evaluated the consequences of an elevated highway collapsing directly on a spent fuel transportation package. This and other analyses consistently show that the accident risks from transportation of spent nuclear fuel are low (see also response to Comment 3.1.2).

Transportation Risk Analyses:

1. NRC (U.S. Nuclear Regulatory Commission). 2014. *Spent Fuel Transportation Risk Assessment Final Report.* NUREG–2125 Washington, D.C. ADAMS Accession No. ML14031A323.

2. NRC (U.S. Nuclear Regulatory Commission). 1977. *Final Environmental Statement on Transportation of Radioactive Material by Air and Other Modes.* NUREG–0170, Washington, D.C. ADAMS Accession Nos. ML022590265, ML022590348.

3. Sprung, J.L., D.J. Ammerman, N.L. Breivik, R.J. Dukart, F.L. Kanipe, J.A. Koski, G.S. Mills, K.S. Neuhauser, H.D. Radloff, R.F. Weiner, and H.R. Yoshimura. 2000. *Reexamination of Spent Fuel Shipment Risk Estimates.* NUREG/CR-6672, Sandia National Laboratory, Albuquerque, New Mexico. ADAMS Accession No. ML003698324.

No revisions were made to NUREG-2224 in response to the comments.

Comment 3.3.2: *Commenters expressed concern about the thermal output of HBU SNF and inquired about the appropriate time for the fuel to cool prior to transportation, including time spent in fuel pools as well as dry storage systems. A commenter suggested that there should be a period of continuous temperature monitoring at air outlets of casks to ensure that recommended temperatures are not exceeded.*

NRC Response: The staff agrees with the comments to the extent that appropriate regulatory controls are necessary for ensuring the thermal output of spent nuclear fuel is managed safely. The NRC considers the current regulatory framework as ensuring that spent nuclear fuel is appropriately cooled prior to either dry storage or transportation and temperatures are maintained within regulatory limits for both low burnup and high burnup spent nuclear fuel. In particular:

1. The NRC conducts inspections and oversight during mock testing and actual loadings to ensure that DSSs are loaded according to the respective specific license or CoC Technical Specification (TS). The NRC verifies that proper loading of the DSS, including compliance with heat load TS, is done by either direct observation during the loading or by review of select fuel selection records, including procedures governing the selection and verification of fuel assemblies prior to each loading. The NRC conducts the oversight review based on the maximum or allowable design values, as approved by The staff per the safety review of the ISFSI or DSS design. It is the licensee's responsibility to ensure compliance with the specific license or CoC TS (e.g., that head load values are properly calculated to comply with the TS).

2. Prior to transportation, a dry storage system canister must be inspected to verify its integrity and ensure the contents meet the description and conditions as given in the CoC for the transportation package (per 10 CFR 71.87).

See also response to Comment 3.1.3 for further information on monitoring and inspection of DSSs.

No revisions were made to NUREG-2224 in response to the comments.

Comment 3.3.3: *Commenters expressed concerns that SNF may remain at the reactor sites and not be transported from the current locations. The concerns included the potential for damaged DSS canisters to not be transportable to another storage or disposal facility and DSS canisters that are not approved for transport. Some commenters expressed concern that damaged fuel may need to be repackaged and some sites no longer have a spent fuel pool that would allow for repackaging. A commenter stated that, despite NRC regulations and technical specifications requiring no defects in the package prior to transport, the NRC is not inspecting such packages and approves the packages unless they are leaking. Another commenter stated that one of the consolidated interim storage facilities currently under review by the NRC will not have the infrastructure for repackaging HBU SNF.*

NRC Response: The NRC notes the comments.

Prior to transport, the licensee must inspect a DSS canister previously in dry storage to verify its integrity, if relied for primary containment, and ensure that the contents meet the CoC description and conditions for the transportation package (see 10 CFR 71.87). Consistent with the licensee's approved Quality Assurance Program under Subpart H of 10 CFR Part 71, if safety issues are identified during this inspection, the user of the transportation package must pursue corrective actions to ensure that the SNF can be transported safely and per the CoC conditions. As part of its regulatory oversight, the NRC evaluates whether the corrective actions are effective and sufficient to ensure safe transportation. Any potential corrective actions for transportation would be case-specific. The NRC does not prescribe how licensees would take corrective action with respect to specific DSS canister designs. The NRC evaluates whether the corrective action taken is effective and sufficient to maintain the intended functions of the important-to-safety structures, systems, and components (SSCs) of the transportation package and remain compliant with the applicable regulatory requirements per 10 CFR Part 71.

The staff also notes that, per 10 CFR 72.236(h), spent fuel DSS designs "must be compatible with wet or dry spent fuel loading and unloading facilities." If a DSS canister or cask needs to be opened as a result of a corrective action, the licensee must ensure the fuel remains in a subcritical condition with adequate radiation shielding to the occupational workers and the public. The industry has decades of operating experience with wet transfer of new and spent nuclear fuel, which involves similar spent nuclear fuel handling equipment (e.g., cranes) and procedures to what would be used in a dry transfer system (DTS), at a site that did not have a spent fuel pool. Further, the operation of a DTS would be similar to the operations conducted at current reactor sites with licensed ISFSIs where spent fuel is loaded in dry storage systems. While spent fuel transfer operations can present challenges to operators (e.g., working with damaged fuel) as described in Section 4.17.2 of NUREG 2157, ("Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel", 2014, ADAMS Accession No. ML14196A105), these operations routinely maintain public and occupational doses well within existing requirements. This is done despite variations in the facilities, equipment, and the characteristics of the spent fuel being transferred. While these characteristics may vary, the safety regulations do not; therefore, the variation in equipment and fuel characteristics do not present insurmountable challenges or preclude a generic approach to analysis of impacts. In

addition, the NRC requires that facilities and equipment are maintained to ensure safety functions are maintained. Finally, the NRC inspects operating facilities to verify compliance with requirements.

With respect to the availability of transportation packages, the NRC has certified transportation packages for the transport of spent fuel. Some of these packages have been authorized for shipment of HBU SNF. The NRC approves designs only after a robust safety review. Based on these reviews, the NRC has certified package designs to transport HBU SNF currently in dry storage (e.g., the NAC-UMS, HI-STAR 100, and MP-197HB). Transportation of spent fuel must be accomplished in accordance with NRC regulations (i.e., 10 CFR Parts 20 and 71) and applicable DOT requirements.

With respect to potential repackaging of HBU SNF at a consolidated interim storage facility, the staff notes that NUREG-2224 does not address specific licensing actions currently under review by the NRC. However, the staff notes that, the same regulatory requirements in 10 CFR 72.236(h) (mentioned above with regard to corrective actions) also apply to potential repackaging at a consolidated interim storage facility.,

See also response to Comment 3.1.3 for further information on monitoring and inspection of DSSs.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.3.4: *Various commenters stated that radial hydrides result in cladding embrittlement during dry-storage conditions. A commenter expressed concern regarding the effects of vibration during transportation and how this might affect potentially embrittled cladding.*

NRC Response: The NRC disagrees with the comment.

The NRC disagrees that the cladding of HBU SNF rods behaves in a brittle manner due to the presence of hydrides, per the engineering assessment of experimental results on the mechanical performance of HBU SNF, as discussed in Chapter 2 of NUREG-2224.

The staff notes that, per the requirement in 10 CFR 71.71(c)(5), a transportation package for HBU SNF is to be evaluated per the expected vibration loads during shipment. In Section 2.4.2 of the final NUREG-2224, the NRC has defined an approach where an applicant would be expected to identify the strains expected during transportation of HBU SNF subject to hydride reorientation and assess the cumulative damage due to vibration loads. In support of a better understanding of vibration during transportation, Sandia National Laboratory (SNL) recently instrumented surrogate SNF assemblies that was transported by rail from Baltimore to the Transportation Technology Center, Inc. (TTCI) near Pueblo, Colorado. Additional testing was conducted at the TTCI utilizing the same instrumented railcar, transport package, and fuel

assembly. The rail transport shock and vibration data has been published (Kalinina, et al., "Data Analysis of the ENSA/DOE Rail Cask Tests," Sandia National Laboratory, SFWD-SFWST-2018-000494, November 19, 2018, https://www.osti.gov/biblio/1532526-data-analysis-ensa-doe-rail-cask-tests). The staff expects that the approach delineated in NUREG-2224 in combination with the better understanding of vibrations during transport, per the SNL experimental results, will allow applicants to demonstrate reasonable assurance of adequate performance of HBU SNF during vibration normally incident to transport.

No revisions were made to NUREG-2224 in response to the comment.

3.4. Miscellaneous

Comment 3.4.1: *A commenter suggested that alternative storage sites are needed and recommended that such sites be located away from high density populations, away from moist regions, and where security can be assured.*

NRC Response: The comment is out of the scope of NUREG-2224.

NUREG-2224 does not address the construction of alternative dry storage facilities (e.g., a consolidated interim storage facility). NUREG-2224 does not discuss generic siting issues for SNF, but instead expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that 10 CFR Part 72 sets overall requirements for the performance of DSSs at both specific-licensed and general-licenses ISFSIs. More specifically, 10 CFR 72.122 specifies that structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. 10 CFR Part 72 also provides dose limits (10 CFR 72.104 and 72.106) to be met for individuals beyond the site boundary regardless of the location of the ISFSI and provides requirements for ensuring the physical security of an ISFSI (10 CFR Part 72, Subpart H).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.2: *Some commenters expressed the view that the NRC staff seems motivated just to get the spent fuel storage problem behind them, even though the waste will remain hazardous long into the future. One commenter also expressed concern regarding the lack of a coherent strategy or national policy for spent fuel storage generally, which urgently needs to be remedied.*

NRC Response: This comment is out of the scope of NUREG-2224 .

NUREG-2224 does not address issues regarding the national policy for SNF storage generally. The NRC notes that, as an independent regulator, it does not set national policy for SNF disposal. That responsibility lies with Congress and the President. The NRC is responsible for regulating the nation's civilian use of byproduct, source, and special nuclear materials to ensure adequate protection of public health and safety, to promote the common defense and security, and to protect the environment. To fulfill this responsibility, the NRC has established requirements for safe spent fuel storage which provide a sound technical basis for protecting public health and safety as well as the environment.

See also response to Comment 3.2.2 regarding the NRC’s approval of periods of operation for DSSs and ISFSIs.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.3: *Some commenters stated that the decision to use HBU SNF was made by the industry to increase profits. A commenter added that the nuclear industry’s economic struggles do not justify increasing the risks. A commenter stated that the nuclear industry is pushing the NRC to approve higher burnup fuels without addressing potential problems and that the NRC has weakened, rather than strengthened, enforcement of its own safety regulations. A commenter stated that packing fewer HBF assemblies in a canister and using smaller canisters would cost money but will be safer.*

NRC Response: The staff notes the comment.

The NRC does not consider the industry motivations when performing its statutory mission of adequate protection of public health and safety, and the environment. The staff conducts its safety reviews to ensure compliance with the established regulatory requirements of the transportation package, DSS design, or ISFSI, as submitted by the applicant.
See also response to Comment 3.2.1.

No revisions were made to NUREG-2224 in response to the comments.

Comment 3.4.4: *A commenter stated that waste packages have not been licensed for disposal and the geology of the ultimate disposal site should determine the appropriate package. The same commenter also stated the burden is on the NRC to determine how waste should be packaged for the geology of a disposal site.*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not address issues regarding packaging of high-level waste for disposal. NUREG-2224 expands the technical basis of safety review guidance regarding hydride

reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation. Based on this assessment, the staff developed example licensing and certification approaches for evaluating the performance of HBU SNF during dry storage (per the regulatory requirements in 10 CFR Part 72) and transportation (per the regulatory requirements in 10 CFR Part 71). NUREG-2224 is informed, to the extent required by the regulations in 10 CFR Part 72, by the need to consider eventual ready retrieval and disposal of HBU SNF. More specifically, the regulations require that ISFSIs and DSSs be designed to allow ready retrieval of spent fuel, high-level radioactive waste, and reactor-related Greater-than-Class-C (GTCC) waste for further processing or disposal (per 10 CFR 72.122(l)). In addition, DSS designs are required, to the extent practicable, to consider compatibility with removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the United States Department of Energy (DOE) (per 10 CFR 72.236(m)) .

The staff further notes that the Nuclear Waste Policy Act, as amended, identifies specific roles and responsibilities for government agencies involved in the disposal of high-level waste (which includes spent nuclear fuel), including specific roles for the NRC, the Environmental Protection Agency, and the DOE. The activities related to disposal of high-level waste are outside of the scope of NUREG-2224, which only pertains to the transportation and dry storage of HBU SNF (per the regulatory requirements in 10 CFR Part 71 and Part 72, respectively).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.5: *A commenter provided a portion of contention 14 from the "Sierra Club's Petition to Intervene and Request for Adjudicatory Hearing before the US Nuclear Regulatory Commission in the Matter of Holtec International -- Consolidated Interim Storage, Docket No. 72-1050, Sept. 14, 2018."*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not discuss specific licensing and certification actions pending before the NRC, but instead expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.
No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.6: *A commenter inquired about the "Performance-Based Emergency Core Cooling System Acceptance Criteria" rulemaking, which incorporated Petition for Rulemaking-50-84.   Specifically, the commenter stated: "The rulemaking package was delivered to the Nuclear Regulatory Commission in March 2016, but the Commission has not taken action on the proposal for more than 2 years. We would appreciate an explanation for the delay in taking action."*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not address the rulemaking referenced in the comment, but instead, expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that rulemaking actions can be monitored at the following website: https://www.nrc.gov/reading-rm/doc-collections/rulemaking-ruleforum/active/RuleIndex.html.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.7: *Some commenters expressed concerns about DSS canisters being stored near salt-water environments such as the Pacific Ocean. The commenters questioned the effect that the proximity to salt air will have on the canisters' lives. A commenter stated that "...storing the canister in a concrete silo just feet above the high water mark invites future failure. The oceans are rising due to global warming and if no future storage site is identified, there is increased risk of water penetration, which could result in a catastrophic event. Although the likelihood of a tsunami is low, it is still a possibility. It has happened before and must be considered as part of a risk evaluation."*

NRC Response: The comments are outside the scope of NUREG-2224.

NUREG-2224 does not address the issues raised in the comments, but instead, expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that the NRC's existing regulatory framework accounts for environmental conditions that could cause or accelerate degradation of ISFSIs or otherwise challenge the safety of spent fuel storage. The NRC's regulatory program, including a robust environmental review of licensing actions and the inspection and oversight of licensees' ISFSI operations, such as implementation of aging management programs, provides reasonable assurance of the protection of public health and safety as well as the environment.

See also response to Comment 3.1.2.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.8: *One commenter expressed concerns regarding the existence of faults surrounding dry storage facilities and how these faults could exacerbate the possibility of a canister failure.*

NRC Response: The comment is outside the scope of NUREG-2224.

This NUREG does not address seismic issues as they pertain to DSS canister performance during operations. NUREG-2224 expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that current regulations require an applicant for a spent fuel DSS design to provide the design bases and design criteria, including the bounding natural hazards (e.g., seismic, flooding, extreme weather). The NRC also requires an applicant to evaluate the DSS design to demonstrate it will withstand these natural hazards and continue to store spent fuel safely. Further, in order to ensure that the DSS design is suitable for use at its ISFSI location, the licensee must evaluate the natural hazards for its ISFSI site and demonstrate that the relevant storage system's design bases bound the ISFSI site's natural hazards. The NRC inspects a licensee's evaluation of its ISFSI location and selection of a storage system, in addition to inspecting ISFSI construction and operations, to ensure continued compliance with all applicable regulatory requirements. The NRC also inspects the manufacturing of the dry storage systems to ensure compliance with all applicable regulatory requirements.

Additionally, in terms of updated natural hazard information, the NRC considered the applicability of the lessons learned from the Fukushima Dai-ichi accident (e.g., seismic reevaluation) to facilities other than operating power reactors, including ISFSIs. The results of NRC's evaluation are presented in SECY-15-0081, "Staff Evaluation of Applicability of Lessons Learned from the Fukushima Dai-ichi Accident to Facilities Other than Operating Power Reactors" (ADAMS Accession No. ML15050A066).

In conducting that evaluation, the staff considered whether ISFSI licensees should reevaluate and upgrade, as necessary, the design bases for seismic protection. The NRC staff did not identify a need for additional analysis in this area because The staff considers the consequences of an earthquake to be bounded by the non-mechanistic accident event that is analyzed for ISFSIs and dry storage systems.

See also response to Comment 3.1.2.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.9: *One commenter stated that, after draining and drying of storage canisters, any residual water can contribute to oxidation of fuel and cladding as well as hydrogen absorption into the cladding.*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not revise the acceptance criteria used for the draining and drying operations of DSSs, as discussed in the current Standard Review Plans for ISFSIs, DSS designs and transportation packages (see NUREG-1567, “Standard Review Plan for Spent Fuel Storage Facilities” (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility” (ADAMS Accession No. ML101040620), and NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear fuel” (ADAMS Accession No. ML003696262)).

Notwithstanding the out of scope nature of the comment, the staff notes that the impacts of residual water on fuel oxidation and hydrogen absorption in the cladding have been addressed in both NUREG-2214, “Managing Aging Processes in Storage Report, Draft Report for Comment” (ADAMS Accession No. ML17289A237) and an NRC-sponsored technical report produced by the Center for Nuclear Waste Regulatory Analyses (“Extended Storage and Transportation: Evaluation of Drying Adequacy,” ADAMS Accession No. ML13169A039).
No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.10: *A commenter mentioned a contamination event at Battelle Memorial West Jefferson Facility in Ohio when a failed fuel assembly in a concrete cask from the Connecticut Yankee nuclear reactor was sent for evaluation. The commenter expressed concern with the method of decontamination of the cask, specifically that the hydrofluoric acid used could cause rapid deterioration of the concrete.*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not address nondestructive examination (NDE) methods for concrete structures or decontamination approaches. NUREG-2224 instead expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that the facility referenced by the commenter is a DOE facility, and not governed by NRC regulations. The staff also notes that NDE technologies for the inspection of dry storage systems already exist and have been used in the nuclear industry for decades. Methods to apply existing NDE techniques to welded stainless-steel canisters and bolted casks have been developed and implemented by both the Electrical Power Research Institute (EPRI) and dry storage system manufacturers. Further, methods for evaluating the structural performance of nuclear-related concrete

structures are well developed and have been implemented in dry storage systems. The commenter is referred to NUREG-2214, “Managing Aging Processes in Storage Report, Draft Report for Comment” for additional aging management considerations of DSSs (ADAMS Accession No. ML17289A237).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.11: *A commenter stated: “Technical specifications in 10 CFR 72.236 include minimum acceptable cooling time of SNF prior to storage in a storage cask, maximum heat designed to be dissipated, and maximum loading limit.”*

NRC Response: The staff agrees with the comment.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.12: *Various commenters asked questions regarding licensing and certification actions currently under NRC review. A commenter asked questions on the activities and plans of a current vendor of DSS designs.*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not discuss specific licensing and certification actions, but instead, expands the technical basis in support of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.13: *A commenter had several questions regarding the study documented in NUREG-7198, Revision 1 (ADAMS Accession No. ML17292B057), including questions about the age of the fuel, when it was placed in the reactor and how long it was stored in a pool prior to being dried and placed in dry storage. The commenter also stated that “Given the importance of degradation processes to HBF, it would be important to identify the difference in performance of rods with relatively low oxidation and hydrogen absorption to those with higher levels.” The commenter also provided their review of the dynamic testing results and conclusions documented in NUREG-7198, Revision 1 (ADAMS Accession No. ML17292B057).*

Another commenter had several questions and comments, including:
“For the rods tested in this project, what was the age of the fuel? When was it first placed in a reactor? When was it removed and placed in a pool? How long was it in the pool before being

dried and placed in dry storage? How long was it in dry storage before being used in this testing project? Were there any important events during this time?
Were the canisters found to contain helium at appropriate levels upon arrival with no signs of leakage? Were any weld failures noted? The importance of maintaining helium in canisters has been emphasized by the NWTRB in 2010, but we have seen no evidence that frequent monitoring has been required.

What was the condition of the fuel rods upon arrival? Were there any pertinent observations pertaining to the fuel rods prior to cutting them into segments to be tested?

How much heat were the assemblies or fuel rods giving off at the time they arrived at ORNL? Temperature?

What was the actual cladding wall thickness of the rods, not including the oxide layer? (We believe there was no attempt to identify the thickness of the actual metal remaining in the cladding. It was merely calculated based on subtracting the inner diameter from the outer diameter.) Since a standard has been established for metal wall thickness not counting the oxide layer, it would be important to know whether the rods tested still met the standard. 10 CFR 50 defines the maximum cladding oxidation limit to be that nowhere on a fuel rod can the oxidation thickness be more than 17 percent of the original thickness. Given the importance of degradation processes to HBF, it would be important to identify the difference in performance of rods with relatively low oxidation and hydrogen absorption to those with higher levels."

NRC Response: The NRC notes the comment.

The staff notes that the characteristics and specifications of rod segments (irradiated at the H.B Robinson Steam Electric Plant) used in the NRC-sponsored HBU SNF test program are well described in Table H.2 of NUREG/CR-7198, Revision 1 (ADAMS Accession No. ML17292B057). These include specifications of the fuel discharge date, cladding hydrogen content, oxide layer thickness, segment dimensions, and rod average burnups. Table 2-1 of NUREG-2224 also provides the hydrogen content and rod average burnups for the test results evaluated in Chapter 2 of NUREG-2224, which the NRC considers as the relevant parameters for assessing the effects of hydride reorientation.

In addition, the commenter is referred to EPRI Report 1001558, "Design, Operation, and Performance Data for High Burnup PWR Fuel from H. B. Robinson Plant for Use in the NRC Experimental Program at Argonne National Laboratory," 2001 (publicly-available online at https://www.epri.com/#/pages/product/1001558/?lang=en-US), which provides additional information on the HBU SNF rods tested under the NRC-sponsored research program. This report compiles and summarizes the available key information on the fuel rods. The information provided includes design drawings and dimensions, fabrication data, operating history data (rod linear – peak and average — heat generation rates), and fluences, as well as results from the periodic poolside examinations. In addition, calculated neutronics-related values—such as

decay heats and isotopic activities, which are needed for shipment of the fuel—are also provided.

See also response to Comment 3.4.24.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.14: *Commenters questioned the validity of the predicted release of Cesium (Cs)-137 under accident conditions involving fire and impact conditions. One commenter asked that the NRC explain its release projections for Cs-137 under accident conditions. The commenter made reference to Table 4.1 of the Draft Report for Comment, in which non-leaktight containment is evaluated with its release fractions.*

NRC Response: The staff notes the comment.

Cesium exists predominately as cesium iodide in the spent fuel. Cesium iodide is a solid within the typical range of temperatures experienced by spent fuel in storage and transportation. In Table 4.1 of NUREG-2224, the NRC provides bounding release fractions for volatile elements in HBU SNF, including cesium, ruthenium and strontium. Moisture in air would not facilitate a significant release of cesium from the fuel, since most of the cesium, even under accident conditions, would not be exposed to that atmosphere. At high burnups, it has been noted (Einziger and Beyer, "Characteristics and Behavior of High-Burnup Fuel That May Affect the Source Terms for Cask Accidents", *Nuclear Technology*, 159:2, 134-146, 2007) that the cesium migrates to the pellet-pellet interfaces and that the release fraction tracks that of xenon at high temperatures. The cesium comes mostly from the central and mid-radius portion of the pellet. The small rim grains almost completely retain their cesium, although some may be released at very high burnup. The iodine and cesium, generally found on the fuel surface and open cracks when there is significant fission gas release, is not generally volatile at room temperature but does become volatile at higher temperatures. The release fractions for volatile elements in HBU SNF, as listed in Table 4-1 of NUREG-2224, consider the above information. Section 4.2.2 provides appropriate reference citations for the release fractions in Table 4-1.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.15: *A commenter expressed concern regarding NRC's plan for revisions of multiple Interim Staff guidance documents and Standard Review Plans, stating that the NRC intends "...to remove all of them from the public domain prior to providing the fully revised documents. We understand that the revised Standard Review Plans will be available for the public to comment on. However, given the number of documents being altered and combined into single documents this plan poses extraordinary obstacles for the public."*

The commenter recommended "...providing the revised documents but continuing to make the original documents available to the public, in order to facilitate review. Indexes that identify the changes and how the documents are organized would also be helpful. This plan creates an extraordinary burden on the public unless all the original documents are still available so that they can be searched to clearly understand the changes and the new organization."

NRC Response: The staff notes the comment.

The Interim Staff Guidance (ISG) documents were developed for use until such time that the NRC staff's positions could be formally incorporated into comprehensive staff review guidance (i.e., Standard Review Plans (SRPs)). In the process of incorporating the ISGs into the SRPs, the NRC's technical position as described in the ISG will not change without an opportunity for public comment. The updated SRPs are the documents the NRC staff will use for its safety reviews and evaluations of applications for dry storage and transportation of SNF, rather than the previously-developed ISGs. At the time of final publishing of the updated SRPs, the NRC will review the accessibility of the current ISGs with the goal of minimizing any confusion regarding NRC's expectations for its safety reviews and evaluations.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.16: *Commenters requested further engagement with the NRC regarding the specific questions that were asked in the Federal Register notice. A commenter stated a public meeting was needed because the answers will depend on how the information contained in the NUREG-2224, Draft Report for Comment is applied to risk-inform the dry storage and transportation regulatory framework.*

NRC Response: The NRC disagrees with the comment. Since efforts to risk-inform the dry storage and transportation regulatory framework are ongoing, discussions on how NUREG-2224 will be applied with regard to the regulatory structure are premature.

See also response to Comment 4.3.42.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.17: *A commenter stated that further interaction between NRC and the industry is considered crucial for a successful implementation. The commenter stated that the examples in Chapter 3 and 4 of NUREG-2224 allow a wide range of options in licensing of HBU SNF; however, many of those will need additional justification, and there appears to be no clear guidance what level of justification would be needed and what the acceptance criteria would be. The commenter also stated that resolving this on a case-by-case (i.e. application by application) basis does not appear to be an efficient regulatory process. The commenter stated that developing a broad consensus in that respect would help both applicants and NRC.*

NRC Response: The staff agrees, in part, with the comment. The NRC agrees that dialogue between the NRC and industry should continue on the technical issues associated with dry storage and transportation of HBU SNF. However, the NRC disagrees that this cannot be done, for the most part, in the context of licensing and certification actions for dry storage and transportation. NUREG-2224 provides a generic methodology for demonstrating compliance with the pertinent regulatory requirements by providing example licensing and certification approaches. An applicant would be expected to demonstrate that the implementation of these approaches is adequate for each specific DSS or transportation package. Further, an applicant has the flexibility to implement other approaches that demonstrate that the pertinent regulatory requirements are satisfied.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.18: *A commenter stated: "Casks take roughly 500% more damage, depending on the degree of burnup. And transportation is much more dangerous."*

NRC Response: The NRC disagrees with the comment.

The staff evaluates the stress loads for each transportation package or DSS per its own design basis (e.g., allowed contents, fabrication materials, center of gravity, basket geometry, etc.). The regulations in 10 CFR Part 72 are performance-based, and do not specify that bolted-cask DSS are to be resistant to five-times the postulated stress loads relative to canister-based DSSs. The generalization that cask-based DSS designs can either receive or sustain more damage during operations is inaccurate.

See also response to Comment 3.1.2 regarding the evaluation of transportation packages under normal conditions of transport and hypothetical accident conditions.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.19: *A commenter stated: "To put it in perspective, each average nuclear plant produces the radiation, EVERY DAY, to each a medium size nuclear bomb that we have to watch over for thousands of years. Each 2 reactor plant creates around 750 nuclear bombs of radiation in 1 year, 7500 nuclear bombs of radiation in 10 years. This is more than all the bomb tests that were ever performed!"*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 expands the technical basis for the safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that the NRC has no authority over whether nuclear power continues to be part of the energy strategy of the United States. As long as nuclear power is part of the nation's energy strategy and radioactive materials are used in the United States, the NRC is responsible for licensing and regulating the nation's civilian use of radioactive materials to provide reasonable assurance of adequate protection of public health and safety, to promote the common defense and security, and to protect the environment.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.20: *A commenter stated: "And while they sit in water covered spent fuel pools, they present a grave risk. In a wide area catastrophe, there is no way we can guarantee that the plant would remain staffed, and the electric circulation pumps needed to keep the used fuel cool will continue to be supplied. Examples could be a nuclear war, or an EMP that could knock the grid and large transformers out for months, or a Heliophysics based CME from the Sun, which are star sent a few "samples" just in the last few years that missed us, but a modern day Carrington type event could end up with several plants and spent fuel pools going on fire from their self-generated decay heat. Picture 20,000 nuclear bombs worth of radiation being released at the same time."*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 expands the technical basis for the safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation. Notwithstanding the out of scope nature of the comment, the staff notes that, to the extent the comment is related to spent fuel pools, additional information on this topic may be found at https://www.nrc.gov/waste/spent-fuel-storage/pools.html.

Comment 3.4.21: *A commenter stated: "HBF began being used in reactors around 2004-2005. 90% of spent fuel above 45 GWd/MTU is in fuel pools and has been sitting there for more than a decade. The owners of the fuel pools have resorted to dense compaction (re-racking). That creates severe overcrowding and higher risk of criticalities, so the assemblies in the pools are surrounded with neutron absorbers. When the NRC made the decision to increase the use of HBF and double the irradiation time, it did so without regard for long-term storage. NRC has no technical data to support long-term storage that deals with potential failures in storing or transporting it, Alvarez said."*

NRC Response: The staff disagrees with the comment.

The NRC's approval of a particular ISFSI or DSS is based upon a review of that specific application and a demonstration that the ISFSI or DSS will meet all pertinent NRC requirements

during the approved period of operation. The NRC regulations currently provide for an initial term up to 40 years and renewal terms up to 40 years, each. See also the response to Comment 3.1.3 regarding the renewal of a specific license or a certificate of compliance (CoC) under 10 CFR Part 72, and the response to Comment 3.1.7 regarding the NRC's technical basis for dry storage and transportation of HBU SNF.

The staff notes that NUREG-2224 does not specifically address dry storage and transportation for HBU SNF dry-stored longer than 60 years. Regarding the Generic Environmental Impact Statement for Continued Storage of Spent Nuclear Fuel (GEIS) (NUREG-2157, Volume 1, 2014, ADAMS Accession No. ML14196A105), because the timing of repository availability is uncertain, the GEIS analyzed potential environmental impacts over three possible timeframes: a short-term timeframe, which includes 60 years of continued storage after the end of a reactor's licensed life for operation; an additional 100-year timeframe (i.e., 60 years plus 100 years) to address the potential for delay in repository availability; and a third, indefinite timeframe to address the possibility that a repository never becomes available. Environmental Impacts of Continued Storage over the three possible timeframes are summarized in Section 8.1 of the GEIS.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.22: *A commenter stated: "Page 8, [HISTORM COC] NRC says, "The staff found that the material properties of structures, systems, and components important to safety will be maintained during normal, off-normal, and accident conditions so that the spent nuclear fuel can be safely stored for the minimum required years and maintenance can be conducted as required." It is not clear what NRC means by "maintenance?"*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not discuss specific licensing and certification actions pending before the NRC, but instead expands the technical basis in support of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that, a maintenance program describes actions that the licensee needs to implement during the storage period to ensure that the DSS or ISFSI, and their associated features, perform their intended functions. Maintenance programs include actions such as inspections, tests, repairs, and replacements. For additional information on the NRC's staff review guidance for maintenance programs, the commenters are referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No. ML003696262).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.23: *Some commenters stated the NRC should require spent fuel management practices implemented in other countries.*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not evaluate strategies implemented by other countries for the storage of HBU SNF, but instead expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that the NRC actively participates in international working groups and regularly engages with other international regulatory authorities to discuss progress on various countries' initiatives and exchange lessons learned regarding safe spent fuel management.

See also response to Comment 3.4.19 regarding the NRC's role with respect to the energy strategy of the United States.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.24: *A commenter stated that NUREG-2224 relies on insufficient operating data, and data that does not reflect normal operating conditions of fuel burnup. The commenter concludes that the HBU SNF rods from H.B. Robinson do not reflect normal operating conditions of fuel burnup and therefore don't provide valid operating data. The commenter stated that the H.B. Robinson rods were handled in an abnormal way. The commenters further stated that the rods were moved to new fuel assemblies after each cycle, rather than continuing to burn in fuel assemblies with other higher burnup rods. The commenter concluded that cherry-picking operating data to reach a conclusion that high burnup fuel is safe in storage and transport is not acceptable science. Therefore, the commenter concluded that NUREG-2224 should be rejected.*

NRC Response: The staff disagrees with the comment.

Table 2-1 of NUREG-2224 provides the specifications of rod segments used in the NRC-sponsored HBU SNF test program discussed in NUREG/CR-7198, Revision 1 (ADAMS Accession No. ML17292B057). Further, as discussed in Section 2.3.4 of NUREG-2224, the NRC has determined that:

1. the rod-average burnup of the tested hydride-reoriented Zircaloy-4-clad HBU SNF segments is conservative per HBU SNF irradiated in commercial reactors in the United

States (i.e., burnups ranged between 63.8 and 66.8 GWd/MTU, well exceeding 45 GWd/MTU)), and

2. the average hydrogen content of the tested hydride-reoriented Zircaloy-4-clad HBU SNF segments is bounding to other M5®-clad HBU SNF irradiated in commercial reactors in the United States, and conservative to the average hydrogen content of other Zircaloy-2, Zircaloy-4 and ZIRLO™-clad HBU SNF irradiated in commercial reactors in the United States (as determined by metallographic characterization of the cladding in the samples).

Therefore, the NRC considers that the Zircaloy-4 HBU SNF tested (irradiated at the H.B Robinson Steam Electric Plant) with the Cyclic Integrated Reversible Fatigue Tester (CIRFT) provides representative results for other HBU SNF generated from commercial reactor operations in the United States.

The staff also notes that the burnups listed in Table 2-1 are for the specific rod segments tested, and not an average for the assemblies where the rods were obtained. Therefore, any movement of the rods between assemblies is irrelevant to the conclusions based on CIRFT results. The NRC disagrees that the staff is selecting specific operating data to reach a conclusion that HBU SNF is safe in storage and transport. NUREG-2224 is based on sound technical analyses of the experimental results discussed in NUREG/CR-7198, Revision 1 (ADAMS Accession No. ML17292B057).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.25: *A commenter expressed concern that nuclear waste storage canisters in the United Stated are "thin-wall (mostly ½" inch welded stainless steel)" and that these canisters have no pressure monitoring or pressure release valves for use during dry storage or transport. The commenter also stated that the "...NRC gives exemptions to ASME pressure vessel standards that require this. That may have been acceptable for 20-year storage, but not for the longer time periods that the NRC admits may likely occur. This is a major concern of the NWTRB in their December 2017 report to Congress on Spent Nuclear Fuel."*

NRC Response: The staff disagrees, in part, with the comment.

The staff agrees that welded canister-based DSSs in the United Stated generally do not have pressure monitoring of the primary confinement system during dry storage operations. The staff disagrees that the thickness of these canister is inadequate for ensuring their safe dry storage and transport. Further, the staff notes that each DSS design or transportation package is reviewed against the guidance in current Standard Review Plans to ensure adequate performance of the confinement and containment systems per 10 CFR Part 72 (for dry storage) and 10 CFR Part 71 (for transport) requirements, whether the design is consistent with American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code criteria or designed to alternative DSS-specific criteria. The staff's safety review considers the

performance of the confinement and containment systems throughout the entire period of operation, including any extended period of dry storage allowed under a renewal. For additional information on the NRC's staff review guidance, the commenters are referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No. ML003696262. The staff notes that neither 10 CFR Part 72 or 10 CFR Part 71 regulations do not explicitly require a DSS design to be consistent with ASME B&PV Code requirements. Alternative design criteria are reviewed and approved by the staff on a case-by-case basis.

The staff also notes that the ASME Code requirement in Section III, Division 1, Subsection NB for pressure relief valves is specified for active energized systems used at operating nuclear power plants where pressure can continuously increase. By contrast storage and transportation casks/canisters are passive systems in which the increase in pressure due to fuel rod rupture is a known and a limited value. Therefore, the need for pressure relief valves in storage and transportation casks/canisters is evaluated on a case-by-case basis.

No revisions were made to NUREG-224 in response to the comment.

Comment 3.4.26: *A commenter stated that a "2011 IAEA Nuclear Energy Series document regarding Impact of High Burnup Uranium Oxide and Mixed Uranium– Plutonium Oxide Water Reactor Fuel on Spent Fuel Management, discusses damages to uranium oxide high burnup fuel. Was any of this considered? The grain size changes within high burnup fuel as you proceed from the central portion to the outer rim of the fuel. The major portion of high burnup fuel will have a grain size similar to (unchanged from) the as-fabricated grain size of approximately 10 μm typical of commercial fuel. The central portion of the fuel may have some grain growth (up to a factor of 2). The rim portion of high burnup fuel will have much higher burnups than the pellet average and forms restructured fine sub-grains at pellet average burnups > 40 GWd/t U. The sub-grain sizes are generally between 0.1 μm to 0.3 μm [39.49–51]. As the burnup of the [fuel pellet] rim increases the original as-fabricated grain boundaries begins to disappear as the sub-grain structure becomes dominant. This restructured rim is not present in the older fuel where rod or bundle burnups did not exceed 33 GWd/t U."*

NRC Response: The staff notes the comment.

As stated in Section 3.2.2 of NUREG-2224, HBU SNF fuel has different characteristics than low burnup SNF with respect to cladding oxide thickness, hydride content, radionuclide inventory and distribution, heat load, fuel pellet grain size, fuel pellet fragmentation, fuel pellet expansion, and fission gas release to the rod plenum. These characteristics have been considered in NUREG-2224, as they may affect the mechanisms by which the fuel cladding may breach, and the amount of fuel that can be released through these breaches. Additional details may be

found in Appendix C.5 to NUREG/CR-7203, “A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages,” issued September 2015 (NRC, 2015) (ADAMS Accession No. ML15266A413). This reference is provided in NUREG-2224.
No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.27: *A commenter stated that “NRC’s 1997 NUREG-1536 Standard Review Plan for Dry Cask Storage Systems defines high burnup as 40 GWd/MTU, but in the July 2010 NUREG-1536 revision it redefines it as >45 GWd/MTU. What evidence was used to redefine high burnup fuel from > 40 GWd/MTU to >45 GWd/MTU when there is evidence showing damage starting above 33 GWd/MTU? The scope of the high burnup NUREG-2224 should evaluate moderate burnup fuel, also.*”

NRC Response: The comment is out of the scope of NUREG-2224.

NUREG-2224 does not revise the definition of HBU SNF per the current staff review guidance, but instead expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

Notwithstanding the out of scope nature of the comment, the staff notes that, in reality there is no sharp distinction low and high burnup. It is a continuum. That means the difference between fuel burned to 40 GWd/MTU and 41 or 42 GWd/MTU can be very small. The NRC has historically established differences between low burnup and high burnup SNF to assist in the safety reviews of DSSs and transportation packages.

As stated in Section 3.2.2 of NUREG-2224, HBU SNF fuel has different characteristics than low burnup SNF with respect to cladding oxide thickness, hydride content, radionuclide inventory and distribution, heat load, fuel pellet grain size, fuel pellet fragmentation, fuel pellet expansion, and fission gas release to the rod plenum. These characteristics have been considered in NUREG-2224, as they may affect the mechanisms by which the fuel cladding may breach, and the amount of fuel that can be released through these breaches. Additional details may be found in Appendix C.5 to NUREG/CR-7203, “A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages,” issued September 2015 (NRC, 2015) (ADAMS Accession No. ML15266A413). This reference is provided in NUREG-2224.

The staff notes that the conclusions in NUREG-2224 on the effects of hydride reorientation on HBU SNF (i.e., burnups greater than 45 GWd/MTU) adequately bound any potential effects of hydride reorientation on SNF with burnups between 40 and 45 GWd/MTU. This is because, as discussed in Chapter 1 of NUREG-2224, the extent of hydride reorientation is dependent on the cladding hydrogen content and HBU SNF between 40 and 45 has a lower hydrogen content than HBU SNF with burnups greater than 45 GWd/MTU.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.28: *A commenter stated: "In this 2013 DOE A Project Concept for Nuclear Fuels Storage and Transportation report it states: "Experimental data over the last twenty years suggest that fuel utilizations as low as 30,000 MWd/t [30 GWd/MTU] can present performance issues including cladding embrittlement under accident conditions as well as normal operations. The NRC is actively seeking rulemaking to address cladding performance for loss of coolant accidents and reactivity insertion accidents. These cladding performance issues need to be addressed before extended fuel utilization fuel can be loaded into dry casks and transportation systems. Section 9.1 discusses needed R&D". Why is the above referenced experimental data being discounted?"*

NRC Response: The comment is outside the scope of NUREG-2224.

NUREG-2224 does not address SNF cladding performance for loss of coolant accidents and reactivity insertion accidents during power plant operation, but instead expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation. NUREG-2224 also does not address the rulemaking referenced in the comment.

Notwithstanding the out of scope nature of the comments, the staff notes that NUREG-2224 is informed on the condition of HBU SNF when removed from the reactor core, as discussed in Sections 1.2 through 1.4 of NUREG-2224.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.29: *A commenter stated that there is experimental evidence to show embrittlement of HBU SNF cladding and cited a report on ring compression testing by Argonne National Laboratory. The commenter asserted that the NRC does not require damaged fuel to be put in individual damaged fuel cans, even though the report by Argonne National Laboratory shows that HBU SNF can become damaged during dry storage. The commenter questioned how HBU SNF in dry storage will be monitored to ensure it does not become damaged.*

NRC Response: The staff disagrees with the comment.

The NRC disagrees that the cladding of HBU SNF rods behaves in a brittle manner, per the engineering assessment of experimental results on the mechanical performance of HBU SNF, as discussed in Chapter 2 of NUREG-2224. The NRC notes that the report referenced by the commenter was considered in the development of NUREG-2224. The commenter is referred to Sections 1.5.4 and 1.5.5 for the staff's review of those results.

The commenter is also referred to response to Comment 3.1.7 on the technical basis for dry storage of HBU SNF and ongoing activities to obtain experimental and field-gathered evidence on the performance of HBU SNF in dry storage.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.30: *A commenter stated that the solubility of hydrogen in the cladding increases with higher cladding temperatures. The commenter suggested that, accordingly, there is a need to lower the maximum temperature for HBU SNF during dry storage and transport operations.*

NRC Response: The staff disagrees, in part, with the comment.

The staff agrees that the solubility of hydrogen in zirconium-based alloys is dependent of the cladding temperatures, and that the cladding hydrogen content is an important parameter for the extent of hydride reorientation (see Section 1.5.1 of NUREG-2224). However, NUREG-2224 does not revise the current safety review guidance on peak (maximum) SNF cladding temperatures during dry storage and transport operations, as defined in Interim Staff Guidance - 11, Revision 3 (ISG-11, Rev. 3) (ADAMS Accession No. ML033230335). ISG-11, Rev. 3 defined cladding temperature limits that provide reasonable assurance that creep and hydride reorientation will not compromise spent nuclear fuel cladding integrity during transportation and dry storage. Per this guidance, for all fuel burnups (low and high), the maximum calculated fuel cladding temperature should not exceed 400 °C (752 °F) for normal conditions of storage and short-term loading operations (e.g., drying, backfilling with inert gas, and transfer of the cask to the storage pad). In Section 1.2 of NUREG-2224, the staff has recognized that research conducted since issuance of ISG-11, Rev. 3, suggests that hydride reorientation could not be prevented in HBU SNF cladding. Therefore, both the NRC and the DOE have conducted extensive testing to understand the effects of hydride reorientation on the mechanical performance of HBU SNF. The results of this extensive testing program are addressed in NUREG-2224, improving the existing technical basis (e.g., as provided in ISG-11, Rev. 3) supporting the staff's conclusion that radial hydride reorientation will not compromise HBU SNF cladding integrity during transportation and dry storage up to 60 years.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.31: *A commenter stated that HBU SNF gives off substantially more heat than low burnup fuel, and that increasing the number of HBU SNF assemblies in dry storage systems increases the heat load that ventilation must supply. The commenter then stated that the Increased heat also increases the pressure in the fuel rods and in the DSS canister. The commenter expressed concern that some components, such as neutron absorbers and polymer seals, degrade at temperatures well below 400 °C (752 °F).*

NRC Response: The staff disagrees, in part, with the comment.

The staff agrees that HBU SNF is generally thermally hotter that than low burnup SNF. The staff also agrees that the increased temperature results in higher pressures inside the SNF rods. The staff has evaluated the effects of higher pressures on the extent of hydride reorientation – see Section 1.5.3 of NUREG-2224.

The staff disagrees with the assertion that an increase in rod internal pressure due to higher rod internal gas pressures will result in higher DSS canister pressures. The assertion ignores the presence of the SNF cladding, which serves as a defense-in-depth barrier preventing the release of rod internal gases to the the inside of the DSS canister.. Further, the staff takes a conservative approach for the evaluation of potential releases of rod internal gases to the inside of the DSS canister. In the safety review of a DSS canister, applicants conservatively assume the entire failure of all SNF rods in the DSS canister, which provides reasonable assurance that the DSS confinement boundary is maintained during normal, off-normal, and accident conditions of storage (even if there were to be releases of gases from the SNF rods to the inside of the DSS canister).

The staff, however, notes that each DSS design is reviewed against the guidance in current Standard Review Plans to ensure adequate performance of the all DSS structures, systems and components important to safety per the 10 CFR Part 72 regulatory requirements. The staff's safety review includes an assessment of the performance of the spent fuel, subcomponents supporting heat removal, neutron absorbers, and any seals for use in the DSS design. The staff review considers the temperatures expected during the entire period of dry storage operation for the performance of structures, systems and components important to safety. The material composition and respective thermal properties—such as thermal conductivity, thermal expansion, specific heat, and heat capacity as a function of the temperature over the range in which the components are to operate—are verified during the review. The neutron absorber material must be demonstrated to be adequately durable during the period of operation (consistent with the regulatory requirement in 10 CFR 72.124(b)). The materials should have excellent physical and chemical stability, including a high resistance to radiation and corrosion. Further, these materials should experience no reduction in effectiveness under normal, off-normal, and accident conditions. Applicants for DSS designs with metallic seals generally rely on data from the seal manufacturer to determine the maximum service temperatures for seals. Because of the importance of seal integrity, the staff ensures during its review that safety analysis reports include laboratory test results using qualified procedures or data sheets that reference such test results.

For additional information on the NRC's staff review guidance, the commenters are referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities," (ADAMS Accession No. ML003686776) and NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," (ADAMS Accession No. ML101040620).
No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.32: *A commenter stated that transportation of low burnup should occur before considering transportation of HBU SNF in order to gain experience and develop knowledge.*

NRC Response: The staff disagrees with the comment.

To be certified by the NRC, transportation packages must meet the pertinent regulatory requirements under 10 CFR Part 71. The NRC approves the designs only after a rigorous safety review that considers the allowed contents for the transportation packages. Based on these safety reviews, the NRC has certified transportation packages for HBU SNF. Because low burnup spent fuel has been around longer and there is more of it, there are more transportation packages for low burnup fuel than for HBU SNF. However, as more data has become available on the performance of HBU SNF, the NRC has certified transportation packages with HBU SNF contents.

Both domestic and international operating experience since dry storage began in 1986 as well as short-term tests show that both low and HBU SNF can be stored and transported safely. No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.33: *A commenter stated: "Canister Integrity is essential to radiation protection. The seals must be intact and not degraded or leaking. The welds must be intact and not leaking. The stainless steel must not be corroding or cracking. The canister is the second radiation barrier after the very thin cladding of the fuel rods."*

NRC Response: The staff agrees with the comment.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.34: *Various commenters expressed concern that the inert environment of a DSS canister should be verified prior to transport to a consolidated storage facility, such as those with applications currently being reviewed by the NRC.*

NRC Response: The comment is outside the scope of NUREG-2224. NUREG-2224 does not discuss specific licensing and certification actions, such as those for consolidated storage facilities pending before the NRC, but instead expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation. Further, NUREG-2224 does not address guidance nor approval criteria for helium monitoring of the DSS confinement during dry storage operation or prior to transport.

Notwithstanding the out of scope nature of the comment, the staff notes that the safety review (including of any consolidated interim storage facility) is conducted per current Standard Review

Plans for ISFSIs, DSS designs, and transportation packages, as well as any other pertinent Interim Staff Guidance or guidance/technical reports for aging management.

For additional information, the commenter is referred to NUREG-1567, “Standard Review Plan for Spent Fuel Storage Facilities” (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility” (ADAMS Accession No. ML101040620), NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear fuel” (ADAMS Accession No. ML003696262), NUREG-1927, Revision 1, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel” (ADAMS Accession No. ML16179A148), and NUREG-2214, “Managing Aging Processes in Storage Report, Draft Report for Comment” (ADAMS Accession No. ML17289A237).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.35: *A commenter expressed concern that the static bending test results discussed in NUREG-2224 (obtained by testing intact fuel rods) may not be applicable to HBU SNF with pinholes and hairline cracks.*

NRC Response: The staff disagrees with the comment. As stated in Section 2.3.4, the cladding strains that control the static response of an intact fuel rod are the high tensile strains at the face of the crack at the pellet-pellet interface. If a pinhole or hairline crack were to be present at this location, it could influence the static test results because of the strain concentrations they may create. However, the staff considers the probability that a pinhole or hairline crack is at the pellet-pellet crack face simultaneously longitudinally and circumferentially to be low. Therefore, it is reasonable that the CIRFT static test results for intact fuel rods can also be applied to undamaged fuel with pinholes or hairline cracks.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.36: *A commenter expressed that the conclusions in NUREG-2224 required additional detail.*

NRC Response: The staff disagrees with the comment. The staff considers that the conclusions in NUREG-2224 are well defined. The staff provides an adequate assessment and pertinent conclusions regarding the effects of hydride reorientation on the performance of HBU SNF during dry storage and transportation. Sections 2.3.4 and 2.4.3 of NUREG-2224 provide NRC’s conclusions on the whether the discussed results (as obtained from Zircaloy-4-clad HBU SNF segments) bound the performance of other modern cladding alloys. These conclusions provided the basis for the example licensing and certification approaches discussed in Chapter 3 (for dry storage of HBU SNF up to 60 years) and Chapter 4 (for transportation of HBU SNF, directly-loaded or previously dry-stored up to 60 years).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.37: *Various commenters stated that only Zircaloy-4-clad HBU SNF was tested in the NRC-sponsored test program, as discussed in NUREG-2224. The commenters stated that fewer studies have been conducted to assess the mechanical performance of ZIRLO™-clad or M5®-clad HBU SNF, which a commenter concluded are more susceptible to hydride reorientation in the cladding. A commenter concluded that extreme levels of hydrogen in Zircaloy-4 cladding in HBU SNF result in cladding embrittlement. The commenters also stated that the NRC should not make conclusions on the mechanical performance of ZIRLO™-clad or M5®-clad HBU SNF based on the test results of Zircaloy-4-clad HBU SNF.*

NRC Response: The NRC disagrees, in part, with the comment.

The NRC agrees that only Zircaloy-4-clad HBU SNF was tested in the NRC-sponsored test program, as discussed in NUREG-2224, and that static and dynamic bending of ZIRLO™-clad and M5®-clad HBU SNF following a hydride reorientation treatment is currently being tested under the DOE's Sister Rod Program. Additional information on this program may be found at https://www.energy.gov/ne/downloads/high-burnup-spent-fuel-data-project-sister-rod-test-plan-overview.

The staff notes that the extent of hydride reorientation of HBU SNF is a function of multiple parameters (e.g., hydrogen content, cladding microstructure, end-of-life rod internal pressures) and not just the cladding composition. The staff has provided discussions on these topics in Section 1.5 of NUREG-2224. Further, the NRC disagrees that the cladding of HBU SNF rods behaves in a brittle manner due to the presence of hydrides, per the engineering assessment of experimental results on the mechanical performance of HBU SNF, as discussed in Chapter 2 of NUREG-2224.

With respect to the applicability of static bending results for Zircaloy-4-clad HBU SNF to other claddings used with HBU SNF, as discussed in Section 2.3.4 of NUREG-2224, the NRC has determined that:

1. the rod-average burnup of the tested hydride-reoriented Zircaloy-4-clad HBU SNF segments is conservative per the HBU SNF irradiated in commercial reactors in the United States (i.e., burnups ranged between 63.8 and 66.8 GWd/MTU, well exceeding 45 GWd/MTU)), and
2. the average hydrogen content of the tested hydride-reoriented Zircaloy-4-clad HBU SNF segments is bounding to other M5®-clad HBU SNF irradiated in commercial reactors in the United States, and conservative to the average hydrogen content of other Zircaloy-2, Zircaloy-4 and ZIRLO™-clad HBU SNF irradiated in commercial reactors in the United States (as determined by metallographic characterization of the cladding in the samples).

Therefore, the NRC considers that the Zircaloy-4 HBU SNF tested (irradiated at the H.B Robinson Steam Electric Plant) with the Cyclic Integrated Reversible Fatigue Tester (CIRFT)

provides representative results for other HBU SNF generated from commercial reactor operations in the United States.

Comment 3.4.38: *The NRC staff received one comment which was interpreted as a derogatory comment regarding the NRC staff.*

NRC Response: The comment is out of the scope of NUREG-2224.

NUREG-2214 expands the technical basis in support of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.39: *A commenter requested clarification on NRC's expectations of CoC holders and licensees once NUREG-2224 is finalized with regards to implementation for HBU SNF currently residing in operating DSSs, future CoC amendments, and CoC renewal applications.*

NRC Response: The staff notes the comment. NUREG-2224 presents example approaches for licensing and certification of HBU SNF for dry storage (under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste") and transportation (under 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"). Staff expects these examples approaches, when followed by applicants, to minimize or eliminate the need for requests for additional information during the staff's safety review of applications for HBU SNF dry storage and transportation. The staff anticipates that future revisions of the Standard Review Plans for dry storage systems and transportation packages will reference the licensing and certification approaches delineated in NUREG-2224.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.40: *A commenter questioned how the conclusions in NUREG-2224 will be used by the NRC to "develop better peak cladding temperatures".*

NRC Response: The staff disagree with the comment. The purpose of NUREG-2224 does not include revising the maximum (peak) cladding temperature limits defined as acceptance criteria for adequate fuel cladding performance per existing review guidance in Interim Staff Guidance 11, Revision 3 (ADAMS Accession No. ML033230335). Any future revision to that acceptance criteria would be addressed in a separate effort and would be incorporated into a revised Standard Review Plan for ISFSIs and DSSs, and a revised Standard Review Plan for transportation packages after a public comment period.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.41: *A commenter stated that significant portions of NUREG-2224 are repeated in other sections. The commenter stated that a more effective arrangement would be to refer to the original section when the information is presented versus repeating all of the information.*

NRC Response: The staff notes the comment. Although NUREG-2224 repeats some sections in Chapter 3 and Chapter 4, the staff intentionally repeated these sections so the reader would only have to refer to the respective chapter for information on either dry storage (10 CFR Part 72) or transportation (10 CFR Part 71).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.42: *Various commenters stated that the NRC does not adequately consider the presence of the hydride rim in HBU SNF cladding.*

NRC Response: The staff disagrees with the comment. The concern expressed in the comment is addressed in the discussion in Section 1.4, in which the staff cites data generated by Argonne National Laboratory (ANL), which have shown that, for the full range of gas pressures anticipated during drying and storage, the hydride rim remains intact following cooling under conditions of decreasing pressure. Based on these results, the staff has concluded that the hydride rim has load bearing capacity and, therefore, it is appropriate to include the hydride rim in the effective cladding thickness calculation. Therefore, the staff considers the inclusion of the hydride rim thickness in the calculation of the effective cladding thickness acceptable when mechanical test data referenced in the structural evaluation have adequately accounted for its presence.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.43: *A commenter stated that, instead of addressing the existing high burnup fuel problems, the nuclear industry and Department of Energy are pushing for even higher burnup fuels, with the misnomer name of "Accident Tolerant Fuels".*

NRC Response: The comment is outside the scope of NUREG-2224. NUREG-2224 does not address the licensing and certification of Accident Tolerant Fuel, but instead expands the technical basis of safety review guidance regarding hydride reorientation in HBU SNF cladding and provides an engineering assessment of the results of research on the mechanical performance of HBU SNF following hydride reorientation

Notwithstanding the out of scope nature of the comment, for additional details on NRC's preparatory activities related to Accident Tolerant Fuel, the commenter is referred to the "Project

Plan to Prepare the U.S. Nuclear Regulatory Commission for Efficient and Effective Licensing of Accident Tolerant Fuel, Version 1.0" (ADAMS Accession No. ML18236A507).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.44: *A commenter questioned whether the discussion in Section 2.3.2 implies that an SNF rod structurally behaves like reinforced concrete beams.*

NRC Response: The staff notes the comment.

The discussion in Section 2.3.2 on the concrete slab and steel I-beam was included to explain the behavior of a composite system where the centers of gravity of each of the two components (i.e., concrete slab and steel I-beam) are not coincident. As later discussed in the same section, for composite fuel rods, the centers of gravity of the two components (cladding, fuel pellet) are coincident. Therefore, NUREG-2224 does not imply that an SNF rod structurally behaves like a reinforced concrete beam.

The staff has also addressed the effects of the fuel rod's end-of-life rod internal pressure and associated cladding hoop stresses on hydride reorientation – see Section 1.5.3 of the report.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.45: *A commenter stated that the NRC's conclusions regarding the structural strength of the fuel pellet are inadequate.*

NRC Response: The staff disagrees with the comment. As discussed in Chapter 2, the pellet does impart flexural rigidity to the fuel rod per the experimental results provided in NUREG/CR-7198, Revision 1 (ADAMS Accession No. ML17292B057). Therefore, the structural analysis of the fuel rod can account for the pellet's bending resistance, if adequate empirical data has been obtained.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.46: *A commenter expressed concern that the NRC is allowing for the SNF cladding to be cracked up to 50 percent.*

NRC Response: The staff disagrees with the comment. NUREG-2224 does not make any conclusions on the acceptability of cracks in the cladding. The staff's current guidance on the classification of SNF for dry storage and transportation (i.e., intact, undamaged, and damaged fuel) can be found Interim Staff Guidance 1, Revision 2 (ADAMS Accession No. ML071420268).

The staff notes that Section 1.5.4 of NUREG-2224 discusses the criterion used by Argonne National Laboratory to define an effective ductility based on ring compression testing. The criterion depends on whether the tested sample exhibited more than 2 percent offset strain before significant cracking occurred (i.e., crack extension exceeding 50% of the cladding thickness). Based on this criterion, Argonne National Laboratory was confident that the samples had adequate effective ductility.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.47: *A commenter stated that length-wise hydrides, parallel to the length of the fuel rod (perpendicular to the circumference) are more likely to fail due to hoop stress, which is always the strongest direction of stress. Nonetheless, the hydride cracking appears somewhat random, so hoop stress and splitting open of fuel rods should be assumed.*

NRC Response: The staff disagrees with the comment. As discussed in Section 1.4, hydrides preferentially precipitate in the circumferential-axial orientation during reactor operation. The mechanical properties of cladding with hydrides in the circumferential-axial direction have been well characterized for use in the structural evaluations of dry storage systems and transportation packages. Further, per the technical basis in Section 1.5 of NUREG-2224, the staff has concluded that the different orientations of hydrides in HBU SNF (i.e. circumferential-axial and radial-axial) have been adequately considered and the orientation does not have a meaningful impact to the mechanical properties of the cladding.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.48: *A commenter stated: "NAC International Inc. ("NAC") is writing to share its comments regarding draft NUREG- 2224. NAC has extensive experience in the transportation of spent nuclear fuel. This includes the transportation of low burnup and high burnup fuel both domestically and abroad. NAC has obtained a Certificate of Compliance (CoC) from the NRC authorizing the transportation of high burnup PWR fuel without having to use damaged fuel cans or demonstrating moderator exclusion. In other words, NAC was the first to be authorized for shipment of bare high burn up PWR fuel assemblies using a method that protected and demonstrated the adequacy of the HBU fuel cladding. Obtaining this approval resulted in many first of kind questions, evaluations, and analyses. After reviewing draft NUREG-2224, we noticed that one of the significant questions raised during our endeavor to obtain approval for the bare shipment of high burnup PWR fuel was not included. After a transportation cask is loaded, the remaining water must be purged from the cask and in our CoCs, the cask must be vacuum dried and backfilled with helium. These evolutions must be performed within an analytically based time limit. Failure to meet these time requirements results in an action for the cask to be re-flooded with water to re-establish an acceptable cooling method for the fuel assemblies.*

The NRC rejected NAC's request to ship bare high burnup PWR assemblies that have been subjected to a cask re-flood. ISG-11, "Cladding Considerations for the Transportation and Storage of Spent Fuel," Revision 3 discusses this type of repeated cycling. It also establishes a limit of 10 cycles and cladding temperature variations less than 65°C {117°F) for both low burnup and high burnup fuel. However, ISG-11 is written in reference to dry cask storage only and not transportation. NAC encountered this exact issue when obtaining its CoC for the transportation of bare high burn up PWR fuel assemblies.

The NRC should revise draft NUREG-2224 to include a discussion of this issue. It should provide the basis for why they have a concern with cycling high burnup fuel at least once in transportation and not dry cask storage, and specifically with respect to any limitations on reflooding a previously dried cask. It would also benefit the industry if the NRC would include guidance on addressing their concerns.

NRC Response: The staff agrees, in part, with the comment. The staff disagrees that the acceptance criteria for cladding conditions, as discussed in Interim Staff Guidance (ISG)-11, Revision 3 (ADAMS Accession No. ML033230335), applies only to dry storage. The acceptance criteria apply to both transportation and dry storage of SNF.

The staff agrees that the thermal cycling criterion in ISG-11, Revision 3, has been a pressing technical concern for industry, as it limits the operational options for a licensee if there is a need for a reflooding of a bare fuel cask. The results discussed in NUREG/CR-7198, Revision 1 (ADAMS Accession No. ML17292B057), and evaluated in NUREG-2224, provide reasonable assurance that intact spent nuclear fuel can be subjected to at least one thermal cycle exceeding 65 °C (117°F) (e.g., during reflooding) without compromising the safety analyses for the transportation package or dry storage system. This conclusion applies to cladding that has been demonstrated to be free of hairline cracks and pinholes, as well as larger defects. An applicant should provide a justification, on a case-by-case basis, for the effects of reflooding on potential oxidation of the fuel pellet during reflooding operations if the cladding is not demonstrated to be intact (e.g., undamaged cladding with hairline cracks and pinholes). The staff revised the report to clarify this new position regarding the acceptance criteria in ISG-11, Revision 3 (ADAMS Accession No. ML033230335).

Comment 3.4.49: *A commenter stated: "While the technical content of the report is sound, it solely focuses on macroscopic metrics to evaluate the performance of HBU spent fuel (fuel's temperature, rod internal pressure, and the environment (storage or transport operations). I was a bit surprised it did not include any discussion/characterization or at least some acknowledgements about some of the underlying "mesoscale" mechanisms such at interfacial intermixing, interface chemistry etc. occurring during the aging process of spent fuel that may ultimately impact the fuel rod load-carrying capacity (shifted from the pellets to the cladding) and pellet-clad debonding. This will not only depend on the type of fuel but also on the type of cladding used."*

NRC Response: The NRC notes the comment. NUREG-2224 focuses on the macroscopic behavior of HBU SNF, consistent with historical practice per the current Standard Review Plans for dry storage and transportation. The NRC has approved the use of empirical models based on the macroscopic behavior of the cladding to evaluate its integrity under normal conditions of transport and hypothetical accident conditions (transportation). Based on the engineering assessment discussed in Chapter 2 of NUREG-2224, the NRC has provided applicants with a new methodology derived from Cyclic Integrated Reversible Fatigue Tester results to evaluate cladding integrity using less, but still appropriate, conservatism. This new methodology accounts for the increased flexural rigidity (bending stiffness) provided by the pellet to the fuel rod.

For additional information on the NRC's staff review guidance, the commenter is referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No. ML003696262).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.50: *One commented stated: "Inclusion of laboratory-directed research and development work on pellet-cladding interaction regarding the additional complexity introduced at the pellet-cladding interface which might give an additional degree of complexity to high burn-up fuel science and technology."*

NRC Response: The NRC disagrees with the comment. NUREG-2224 focuses on the macroscopic behavior of HBU SNF, consistent with historical practice per the current Standard Review Plans for dry storage and transportation (see response to Comment 3.4.49 in Section 3.4).

The inclusion of laboratory-directed research and development work on pellet-cladding interaction does not serve the intent of NUREG-2224.

For additional information on the NRC's staff review guidance, the commenter is referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No. ML003696262).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.51: *A commenter stated: "Work on atomistic models showing the differences in elastic and mechanical properties of Zry-2, Zry-4, ZIRLO, and M5 should be included/discussed. See, Weck, P.F., Kim E, Tikare V. Mitchell J.A., "Mechanical Properties of zirconium alloys and zirconium hydrides predicted from density functional perturbation theory," Dalton Trans. 44, 18769 (2015) This work could readily be extended to irradiated zirconium alloys."*

NRC Response: The staff notes the comment.

NUREG-2224 focuses on the macroscopic behavior of HBU SNF, consistent with historical practice per the current Standard Review Plans for dry storage and transportation (see response to Comment 3.4.48). NUREG-2224 discusses experimental testing on the macroscopic structural performance of HBU SNF, which is different to the work on atomistic models cited by the commenter.

For additional information on the NRC's staff review guidance, the commenter is referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No. ML003696262).

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.52: *Various commenters stated that shocks during normal conditions of transport were not adequately addressed in NUREG-2224.*

NRC Response: The NRC disagrees with the comment. Per Section 2.4.2 of the final NUREG-2224, an applicant would be expected to identify the strains anticipated for the transportation package being certified, which may include, for example, shocks associated to rail-car-to-rail-car coupling during transport. The staff notes that shock events are an integral part of normal conditions of transport and recognizes that these loads will produce higher cladding strains. However, due to their low frequency, these shocks will not contribute significantly to the cumulative damage assessed per Section 2.4.2 of NUREG-2224.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.53: *A commenter stated the results discussed in Section 2 of NUREG-2224 indicate that the hydride-reoriented HBU SNF segments failed at bending moments exceeding 90 Newton·meter. The commenter concluded that the NRC should have tested at lower force levels.*

NRC Response: The NRC notes the comment. The NRC tested HBU SNF rod segments over a range of bending moments between 0 (i.e., no load) to 90 Newton-meters (N·m). Therefore, all load levels up to 90 N·m were adequately tested. The HBU SNF segments did not fail at these bending loads even after hydride reorientation.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.54: *A commenter stated that the comparison of flexural rigidity between the tested HBU SNF segments and the cladding-only response was not provided to actual testing of cladding without fuel, but by calculation. The commenter further questioned why the NRC did not use actual testing of cladding only.*

NRC Response: The NRC disagrees with the comment. As discussed in Section 2.3.3 of NUREG-2224, the flexural rigidity is the product of the elastic modulus, E, and the moment of inertia of the cross section, I. The moment of inertia is a constant and depends only on the cladding diameter and thickness. The elastic modulus is based on experimental cladding-only tensile test data. Therefore, the flexural rigidity is indeed based on experimental cladding-only test data since the moment of inertia is a constant.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.55: *A commenter stated that the NRC should not make conclusions regarding the safety of HBU SNF during design-basis drop accidents based on a single static bending test of Zircaloy-4-clad HBU SNF. The commenter also expressed concern that the testing may have not been performed on HBU SNF with a sufficiently high cladding hydrogen content and oxide layer.*

NRC Response: The NRC disagrees with the comment.

The NRC recognizes that only one CIRFT static bending test was conducted under the NRC-sponsored test program discussed in NUREG-2224. However, to compensate for the limited data, the staff chose a conservative testing approach (radial hydride treatment) to maximize the fraction of cladding radial hydrides precipitated in the HBU SNF test segments. Section 2.3.4 also states that thermal cycling was repeated for five cycles to further induce a higher fraction of radial hydrides. This conservative approach allowed the staff to make generalizations on hydride reorientation's effects on modern cladding alloys, as discussed later in Section 2.3.4 of the report.

In addition, the staff notes that, during a 30-foot side drop, the maximum bending moment in the rod is approximately 34 N-m. Figure 2-7 shows that at bending moments during loading that are less than 35 N-m, the flexural rigidities of the four as-irradiated rods—which have only circumferential hydrides—and the hydride-reoriented segment (denoted as HR2 in Figure 2-7)—

which has both circumferential and radial hydrides—are essentially the same. This result supports the pretest expectation that, because the bending tensile stress in the cladding is parallel to the plane of both the radial and circumferential hydrides, the presence of radial hydrides would not significantly alter the flexural response from the case where only circumferential hydrides are present.

DOE is continuing the research the NRC began by testing other cladding types, testing HBU SNF with different hydrogen content, oxide thicknesses, etc. Additional information on DOE's research program may be found at https://www.energy.gov/ne/downloads/high-burnup-spent-fuel-data-project-sister-rod-test-plan-overview. NUREG-2224 reflects what was learned from the testing sponsored by the NRC.

See also response to Comment 3.4.37 on NRC's conclusions relative to other HBU SNF generated from reactor operations in the United States.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.56: [Page 1-24, Lines 43-45 and Page 1-25, Lines 1-2 in the Draft Report for Comment]

A commenter expressed concerns regarding the structural analysis supporting NUREG-2224. The commenter expressed concern that the structural analysis does not consider pinch loads that would occur in a drop accident, with the fuel rods above providing a force on top of the rod being analyzed.

NRC Response: The staff notes the comment. The structural analyses of fuel rods to determine fuel rod displacements and cladding longitudinal bending stresses under hypothetical accident conditions (transportation) (HAC) and normal conditions of transport have always used cladding-only mechanical properties. Section 2.3.3 determines a factor, based on the results in NUREG/CR-7198, Rev 1 (ADAMS Accession No. ML17292B057), by which the cladding-only mechanical properties can be multiplied to account for the fact that the fuel pellets increase the flexural rigidity of the rod, which controls the rod's displacement and longitudinal bending stress response to lateral loads.

The commenter states "The concern is that the structural analysis only considers lateral inertia loads on the fuel rod supported from the bottom and does not consider the pinch load that would occur in a drop accident from rods above providing a force on top of the rod being analyzed." The pinch loading caused by rod-to-rod contact during HAC was not part of the test program in NUREG/CR-7198, Rev 1. The pinch mode is discussed in Section 1.5.5 of NUREG-2224. The purpose of the test program was to evaluate the effect of fuel pellets on the flexural rigidity of the rod and cladding longitudinal bending stresses.

No revisions were made to NUREG-2224 in response to the comment.

Comment 3.4.57: *One commenter noted that there is no discussion about a lower temperature bound that could impact cladding integrity during vacuum drying and transportation operations. The commenter also stated that the report focuses a lot on the higher temperature bounds, but, if a plant decides to go into safe store decommissioning, is there a lower temperature bounds that plants should be aware of.*

NRC Response: The NRC notes the comment.

The comment pertains to long periods of storage as the fuel cools down. NUREG-2224 focuses on the phenomena of hydride reorientation and its impacts dry storage and transportation. The reader is referred to NUREG-2214 for generic discussions on potential degradation phenomena occurring in dry storage systems during periods of operation up to 60 years.

Comment 3.4.58: *One commenter noted that the labels in Figure 2-8 are not very clear at first glance for EI1, A, EI2, and B. NRC should include lines between the labels and the points on the graph referred to for clarity for Points A and B.*

NRC Response: The NRC notes the comment. The figure is a reproduction from NUREG/CR-7198, Revision 1. The reader is referred to that document for additional clarification.

4. Specific Technical Comments

The NRC received several comments specific to the technical content of NUREG-2224. The NRC's responses and, where necessary, resulting edits to NUREG-2224 are documented below. The identified section, page, line and, if applicable, figure or table, pertain to the Draft Report for Comment.

4.1 Abstract

Comment 4.1.1: [Page iii, Line 3 in the Draft Report for Comment]

A commenter requested that the word "performance" in the cited sentence be changed to "properties". The commenter stated that, "overall, we are interested in the performance of the cladding, but it is the time dependent change of the properties that affect the performance."

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised per the comment.

4.2 Chapter 1

Comment 4.2.1: [Section 1.4, Page 1-6, Figure 1-1 in the Draft Report for Comment]

A commenter requested clarification on why it was important for Figure 1-1 to define differences between the inner and outer hydrogen concentrations. The commenter also requested that the typical hydride rim of the cladding segments shown in Figure 1-1 be defined in that figure and in Section 1.4.

NRC Response: The staff agrees with the comment.

The staff has modified the discussion in Section 1.4 to clarify that the distribution of hydrides varies across the thickness of the cladding, as shown in Figure 1-1. The staff considers that Section 1.4 provides sufficient explanation on the hydride rim thickness and density. The hydride rim and density are a function of the hydrogen in the supersaturated condition and the heat flux experienced during reactor operations. Therefore, providing a typical depth of the hydride rim would not provide additional value to the discussion. The staff has instead provided a reference to the reader in Section 1.4 for additional information on the conditions leading to hydride rim formation – see Adamson, R., et al., "Corrosion Mechanisms in Zirconium Alloys". IZNA7 Special Topic Report Corrosion Mechanisms in Zirconium Alloys 2007, Advanced Nuclear Technology International, Skultuna, Sweden, 2007. The staff considers the reference citation to be sufficiently adequate to address the comment.

Comment 4.2.2: [Section 1.4, Page 1-6, Figure 1-1 in the Draft Report for Comment]

Various commenters requested the NRC define the hydrogen content in different radial cross sections of the cladding images shown in Figure 1-1. A commenter requested that a marker identify the different radial cross sections in the images shown in Figure 1-1.

NRC Response: The NRC disagrees with the comment.

The staff has already clarified in Section 1.4 that the distribution of hydrides varies across the thickness of the cladding (in response to Section 4.2.1). In response to Comment 4.2.2. and to be consistent with this revised discussion per Comment 4.2.1, the staff revised Figure 1-1 to identify the average hydrogen content of the samples, which the staff considers adequate for demonstrating the differences between the hydride content in the different cladding alloys. Since Figure 1-1 no longer defines differences in hydrogen content between the outer 1/3 and inner 2/3 of the cladding, the suggested "dashed-line" marker is no longer needed.

No additional changes were made to NUREG-2224 in response to the comment.

Comment 4.2.3: [Section 1.4, Page 1-6, Figure 1-1 in the Draft Report for Comment]

A commenter requested to add a note and leader indicating the waterside and the oxide layer to Figure 101.

NRC Response: The staff agrees with the comment.

The staff has added notes in Figure 1-1 to identify the water side and the oxide layer.

Comment 4.2.4: [Section 1.4, Page 1-6, Figure 1-1 in the Draft Report for Comment]

A commenter requested to add a note and leader indicating the hydride rim for the Zircaloy-4 and ZIRLO™ cladding cross-section images in Figure 1-1.

NRC Response: The staff agrees with the comment.

The staff has added notes in Figure 1-1 to identify the hydride rim for the Zircaloy-4 and ZIRLO™ cladding cross-section images.

Comment 4.2.5: [Section 1.4, Page 1-6, Figure 1-1 in the Draft Report for Comment]

A commenter stated that Figure 1-1 should define the linear heat generation for the HBU SNF cladding shown.

NRC Response: The staff disagrees with the comment. The staff notes Section 1.4 provides perspective on the formation of a hydride rim at high burnups. However, the information on linear heat during operation of the cladding segments is not used elsewhere in the document. Therefore, the staff considers that adding that information to Figure 1-1 does not provide additional value to the report.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.6: [Section 1.5 in the Draft Report for Comment]

A commenter stated that Section 1.5 discusses many forms of information, including the results of post-irradiation examinations, physical testing of irradiated cladding samples, interpretations of metallographic images of irradiated cladding samples, rod performance code predictions, and finite element calculations. The commenter requested that the NRC clarify, whenever possible, the types of information and data being discussed. The commenter provided examples of where clarification may be appropriate.

NRC Response: The staff agrees with the comment. The staff has added clarification of the types of data used in the evaluation throughout the NUREG-2224, as appropriate. The staff notes that revisions have been made in response to other comments. When the specific sentences cited by the commenter were not removed from the text in response to other comments, the staff implemented the proposed individual revisions to the text.

Comment 4.2.7: [Section 1.5.3, Page 1-11, Lines 5 to 9 in the Draft Report for Comment]

A commenter stated that Section 1.5.3 implies that there is a close correlation between burnup and the release of pellet-sequestered fission gases to the rod free volume. The commenter asserted that the literature does not appear to support this general conclusion. To avoid the implication that there is a strong correlation, the commenter proposes that the discussion be prefaced with "observations of commercial power fuel indicate that...". The commenter also stated that cited discussion may be more appropriate in Section 1.5.3.1, instead.

NRC Response: The staff agrees with the comment. The staff has prefaced the cited discussion with the proposed text. Further, the staff has consolidated the discussions in Section 1.5.3 and Section 1.5.3.1 in response to both this comment and Comment 4.2.9.

Comment 4.2.8: [Section 1.5.3, Page 1-11, Line 12 in the Draft Report for Comment]

A commenter requested to replace the word "release" with "generation" in the cited sentence.

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised per the comment.

Comment 4.2.9: [Section 1.5.3.1, Page 1-14, Lines 12 to 17 in the Draft Report for Comment]

Various commenters discussed the discrepancies on end-of-life rod internal pressures calculated by different studies conducted by Oak Ridge National Laboratory and Pacific Northwest National Laboratory, respectively.

NRC Response: The staff agrees with the comment. The staff recognized in the NUREG-2224, Draft Report for Public Comment, that calculations on end-of-life rod internal pressures using the FRAPCON fuel performance code were performed by both Oak Ridge National Laboratory (Bratton, et al., "Rod Internal Pressure Quantification and Distribution Analysis Using FRAPCON," FCRD-UFD-2015-000636, ORNL/TM-2015/557, 2015) and Pacific Northwest National Laboratory (Richmond and Geelhood, "FRAPCON Analysis of Cladding Performance during Dry Storage Operations." Pacific Northwest National Laboratory, PNNL--27418, 2018). In Section 1.3.5 of NUREG-2224, Draft Report for Public Comment, the staff noted the discrepancies between the FRAPCON code predictions of these two studies and committed to evaluate the merits of both approaches used in analyses. As a result of these comments, the staff has completed that evaluation and has revised Section 1.5.3 to address the results of that evaluation.

Comment 4.2.10: [Section 1.5.3.1, Page 1-14, Lines 12 to 17 in the Draft Report for Comment]

A commenter suggested clarifying that publicly-available empirical data is not available to verify the code predictions; however, the FRAPCON code is well-validated for standard PWR rod predictions.

NRC Response: The staff agrees with the comment. The revised discussion in Section 1.5.3 (in response to Comment 4.2.9) clarifies that the FRAPCON fuel performance code is well-validated for standard boiling water reactor and pressurized water reactor (PWR) rod predictions, as well as for integral fuel burnable absorber (IFBA) PWR rod predictions.

Comment 4.2.11: [Section 1.5.3.1, Page 1-14, Lines 12 to 17 in the Draft Report for Comment]

A commenter questioned whether FRAPCON code modifications were required by the study of Pacific Northwest National Laboratory on the end-of-life pressures of integral fuel burnable absorber rods.

NRC Response: The staff agrees with the comment. The discussion in Section 1.5.3 (as revised in response to Comment 4.2.9) was further revised to reference Section 2 of

"FRAPCON Analysis of Cladding Performance during Dry Storage Operations" (Richmond, D. J. and K. J. Geelhood, Pacific Northwest National Laboratory, PNNL-27418, April 2018) for additional details on the FRAPCON model and assumptions used by Pacific Northwest National Laboratory. The staff considers the reference citation sufficiently adequate to address the comment.

Comment 4.2.12: [Section 1.5.3.1, Page 1-14, Lines 12 to 17 in the Draft Report for Comment]

A commenter requested that clarification be provided that, in the absence of empirical data on the end-of-life pressures of integral fuel burnable absorber (IFBA) rods and with the evidence provided by the code-predicted values, that the pressures in both standard and IFBA rods is well below the 90 MPa cladding hoop stress that has been shown to be capable of producing hydride reorientation in ZIRLO™ fuel rod cladding.

NRC Response: The staff agrees with the comment.

The discussion in Section 1.5.3 (as revised in response to Comment 4.2.9) has been further revised to clarify that, in the absence of publicly-available empirical data on end-of-life (EOL) rod internal pressures (RIPs) for integral fuel burnable absorber (IFBA) rods and per the evidence provided by the FRAPCON code-predicted values (validated by non-publicly available empirical data), the staff concludes that the EOL RIPs in both standard and IFBA rods result in cladding hoop stresses below the 90 MPa (1.3×10^4 psia) level that has been shown to be capable of producing hydride reorientation in ZIRLO™ fuel rod cladding.

Comment 4.2.13: [Section 1.5.3.1, Page 1-13, Figure 1-4 in the Draft Report for Comment]

A commenter suggested modifying the caption of Figure 1-4 to "code-predicted" as opposed to "calculated" rod internal pressures.

NRC Response: The staff notes the comment.

The revised discussion in Section 1.5.3 (in response to Comment Section 4.2.9) has removed the Figure 1-4 previously included in the Draft Report for Comment. Therefore, the revision to the caption of that figure, as requested by the commenter, is no longer needed.

No additional revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.14: [Section 1.5.3.1, Page 1-14, Figure 1-5 in the Draft Report for Comment]

A commenter suggested revising the caption of Figure 1-5 to "Aggregated empirical and code-predicted rod internal pressure for PWR fuel rods, evaluated at 25 C".

NRC Response: The staff notes the comment.

The revised discussion in Section 1.5.3 (in response to Comment 4.2.9) has removed Figure 1-5 previously included in the Draft Report for Comment. Therefore, the revision to the caption of that figure, as requested by the commenter, is no longer needed.

No additional revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.15: [Section 1.5.3.2, Page 1-15, Figure 1-6 in the Draft Report for Comment]

A commenter suggested revising the caption of Figure 1-6 to "code-predicted maximum rod internal pressure (average + 3 sigma) [Bratton et al] evaluated for a range of rod average fill gas temperatures."

NRC Response: The staff notes the comment.

The revised discussion in Section 1.5.3 (in response to Comment 4.2.9) has removed Figure 1-6 previously included in the Draft Report for Comment. Therefore, the revision to the caption of that figure, as requested by the commenter, is no longer needed.

No additional revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.16: [Section 1.5.3.3, Page 1-18, Figure 1-9 in the Draft Report for Comment]

A commenter suggested revising the caption of Figure 1-9 to "calculated PWR cladding hoop stress based on the maximum code-predicted rod internal pressure, evaluated for a range of rod average fill gas temperatures."

NRC Response: The staff notes the comment.

The revised discussion in Section 1.5.3 (in response to Comment 4.2.9) has removed Figure 1-9 previously included in the Draft Report for Comment. Therefore, the revision of the caption of that figure, as requested by the commenter, is no longer needed.

No additional revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.17: [Section 1.5.4, Page 1-19, Line 17 in the Draft Report for Comment]

A commenter stated that NUREG-2224 should differentiate the ductility (as measured by ring compression testing) from the material property elongation (i.e., the classical ductility typically

tabulated in the literature). The commenter requested that NUREG-2224 use the term "effective ductility" when discussing results from ring compression testing.

NRC Response: The staff agrees with the comment.

The staff has made revisions throughout NUREG-2224 to use the term "effective ductility" to differentiate the ring compression testing-measured ductility from the material property elongation (i.e., the classically-defined ductility typically tabulated in the technical literature).

Comment 4.2.18: [Section 1.5.3.1, Page 1-12, Lines 1 to 3 in the Draft Report for Comment]

A commenter stated that the non-proprietary data on the sister rods has initial helium fill pressures as low as 240 psia (1.65 MPa) for one vendor and 290 psia (2.00 MPa) for another; with the much older assembly designs at 365 psia (2.52 MPa).

NRC Response: The staff agrees with the comment.

The staff revised Section 1.5.3 to clarify that some of the older legacy fuel designs have initial helium fill pressures as high as 2.52 megapascals (MPa) (365 pounds per square inch absolute (psia)). The staff notes that the rest of the discussion in Section 1.5.3 adequately addresses helium fill pressures in the range of 1.65 MPa (240 psia) and 2.00 MPa (290 psia).

Comment 4.2.19: [Section 1.5.3.1, Page 1-12, Line 7 in the Draft Report for Comment]

A commenter requested that the cited sentence start with "Data are also not publicly available…"

NRC Response: The staff agrees with the comment.

NUREG-2224 has been revised per the comment. In addition, the staff has clarified in the cited sentence that the type of data being discussed is, more specifically, empirical EOL RIP data (consistent with the response to Comment 4.2.6.).

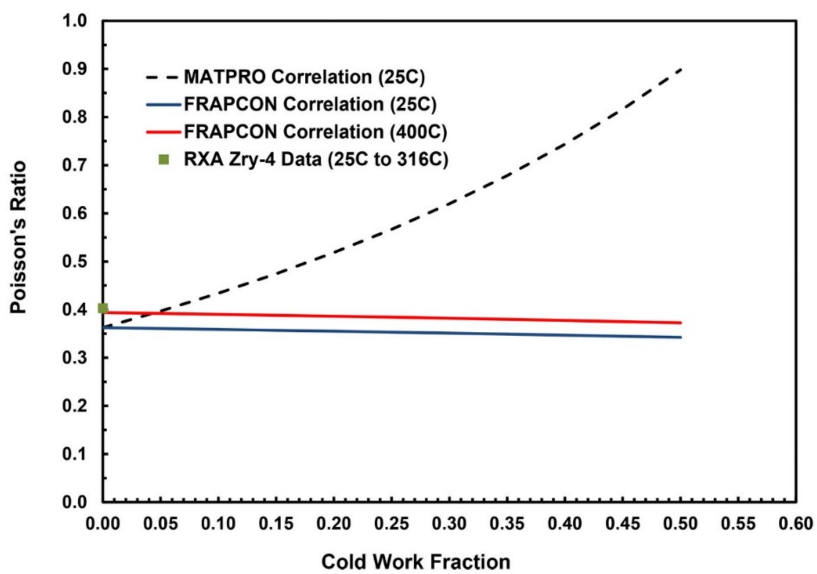
Comment 4.2.20: [Section 1.4, Page 1-7, Lines 4 to 9 in the Draft Report for Comment]

A commenter provided an evaluation and comments on Section 1.4 of the Draft Report for Comment.

The commenter stated: "Mechanical Properties Data for Zr-based Cladding Alloys: The two Geelhood et al. references provide adequate guidance to the applicant for the mechanical properties of Zircaloy-2 (Zry-2), Zircaloy-4 (Zry-4) and to some extent ZIRLO. Both references

contain the same correlation for Young's modulus (E), which is a function of cold-work fraction, temperature, fast fluence, and excess oxygen (not relevant).

Only the 2nd reference (Geelhood et al, 2013 – please correct to 2014 in text and reference list) contains correlations for a much needed second elastic constant, which in this case is the shear modulus (G). For elastic analyses of isotropic materials, most stress-strain relationships are written in terms of E and Poisson's ratio (v), which is generally in the range of 0.2 to 0.4 for metals and always <0.5. Reviewers of license applications should be aware that E and G correlations used in FRAPCON-3.5, FRAPCON-1.5, and MATPRO are compared in Geelhood et al, 2014. Although not pointed out by the authors, the MATPRO correlation for G has an incorrect dependence on cold-work fraction (CW) that leads to unrealistic values of v = E/(2G) – 1. The FRAPCON correlation for G appears to be correct. The figure below shows the comparison among MATPRO-calculated v, FRAPCON-calculated v, and v-data for recrystallized-annealed (RXA, CW = 0) Zry-4 in the 24°C−316°C temperature range [1]. The FRAPCON correlations for E and G give v = 0.37±0.02 in the 25°−400°C temperature range with no significant variation with CW, temperature, and fast fluence.



[1] Scwenk, E.B., K.R. Wheeler, G.D. Shearer, "POISSON'S RATIO IN ZIRCALOY-4 BETWEEN 24° AND 316°C," J. Nucl. Mater. 73 (1978) 129-131."

NRC Response: The staff agrees, in part, with the comment.

The staff corrected the publication date for the Geelhood, et al, reference from 2013 to 2014. The staff, however, considers the proposed clarification regarding the mechanical property correlations more pertinent to future revisions of the Standard Review Plans for dry storage systems and transportation packages. The staff has noted the comment and will make appropriate revisions to those documents in the future.

No additional changes were made to NUREG-2224 in response to the comment.

Comment 4.2.21: [Section 1.2, Page 1-3, Line 8 in the Draft Report for Comment]

A commenter suggest removing the word "other" in the cited sentence, since it implies that Zircaloy cladding is an advanced alloy.

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised per the comment.

Comment 4.2.22: [Sections 1.5.3, 1.5.4, and 1.5.5 in the Draft Report for Comment]

A commenter stated that, since the ideal gas law is known to be applicable over the range of use, the word "extrapolated" is inaccurate. The commenter suggested the use of the word "evaluated", instead.

NRC Response: The staff agrees with the comment.

The discussion in Section 1.5.3 (in response to Comment 4.2.9) has been revised per the comment. Sections 1.5.4 and 1.5.5 do not use the word "extrapolated," therefore no revisions are warranted in those discussions.

Comment 4.2.23: [Section 1.3, Page 1-4, Lines 8-10, and Section 1.5.3, Page 1-11, Line 12 in the Draft Report for Comment]

Various commenters requested clarification that the fission and decay gases generated by the SNF have a lower contribution to the SNF rod internal pressure relative to the initial gas fill at fabrication.

NRC Response: The staff agrees with the comment.

The staff revised the discussion in Section 1.3 of NUREG-2224 to clarify that the main driving force for cladding creep at a given temperature is the hoop stress caused by internal rod pressure. The internal rod pressure results from the initial fill gas pressure condition and, to a smaller extent, from fission and decay gases released to the gap between the fuel and cladding during dry storage operations.

Comment 4.2.24: [Section 1.4, Page 1-5, Line 16 in the Draft Report for Comment]

A commenter requested removing the words "migration and" in the cited sentence to avoid confusion.

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised per the comment.

Comment 4.2.25: [Section 1.4, Page 1-6, Figure 1-1 in the Draft Report for Comment]

Two commenters stated that there was an error on the hydrogen content specified for the inner 2/3 cross-sectional area of the Zircaloy-4 image.

NRC Response: The staff notes the comment.

Since Figure 1-1 no longer defines differences in hydrogen content between the outer 1/3 and inner 2/3 of the cladding (per the revisions made in response to Comment Section 4.2.2), the specific changes requested in the comment are no longer necessary.

No additional changes were made to NUREG-2224 in response to the comment.

Comment 4.2.26: [Section 1.5.5, Pages 1-24 to 1-26 in the Draft Report for Comment]

A commenter provided an evaluation and comments on the discussion in Section 1.5.5 regarding the NRC's assessment of ring compression test results.

The commenter stated:

"The ring compression tests with defueled cladding were conducted as ductility screening tests. Proper interpretation of the results are as follows: (a) if the samples exhibit ≥2% offset strain before significant cracking occurs (>50% of wall), we are confident that the samples have ductility; (b) for temperatures at which the offset strains drop below 2%, the ductility is too low to be measured with confidence by the RCT. The 2% offset-strain criterion is conservative for M5, which has a very thin oxide layer. For M5, we have confidence in ductility at offset strains as low as 1.5%. However, such a change in criterion would have little impact on the DDT. The staff has presented interesting arguments and analyses concluding that the RCT loading generates more displacement than would be experienced by accelerating cladding impacting a grid-spacer spring. In particular, for equivalent support loads, the displacement (change in outer diameter in loading direction) would be about 50% less for a distributed dynamic load. The argument can still be made that RCT loading, which induces bending in the hoop direction, is more prototypic than pressurized-tube loading or ring-stretch loading. Also, NRC's conclusion that RCT loading results in higher hoop bending stresses than pinch loading works well with

what we expect to demonstrate: that defueled cladding has adequate ductility for cladding subjected to relevant ranges of peak internal gas pressure, peak drying-transfer-storage cladding temperature (PCT), and slow cooling under pressure/hoop-stress."

NRC Response: The staff agrees with the comment. The staff revised the description of Argonne National Laboratory's ring compression testing criterion for assessing effective ductility of the cladding (as discussed in Section 1.5.4 of NUREG-2224) per the comment.

Comment 4.2.27: [Section 1.4, Page 1-7, Lines 4 to 9 in the Draft Report for Comment]

A commenter expressed concerns about some of the results on the mechanical properties of M5® cladding, as discussed in some of the references cited in NUREG-2224.

The commenter stated: "The three references for M5 mechanical properties present data and correlations for M5 based mostly on non-ASTM ring-stretch tests and the data/correlations are inadequate for analysis of spent nuclear fuel rods during drying-transfer-storage-transport. For example, correlations are given for the ultimate tensile stress (UTS) and the uniform elongation (UE). Because of sample geometry and loading method, the ring-stretch sample has no deformation regime for which the stresses and strains are uniform within the gauge section. Also, the total elongation (TE) is highly dependent on sample geometry and loading details. The Bouffioux reference does contain room-temperature results of ASTM-type tube burst tests for irradiated M5. However, axial tensile data are needed, along with data from elevated temperature tube-burst tests to determine axial and circumferential plastic stress-strain relationships for irradiated M5. Determination of M5 mechanical properties is a high priority in Sister Rod testing."

NRC Response: The staff agrees with the comment. The staff revised the discussion in Section 1.4 to clarify that an applicant for a dry storage system or transportation package should adequately justify the use of any mechanical properties and the associated experimental methods for the relevant fuel designs. Further, the staff has noted the comment for incorporation into a future revision of the Standard Review Plans for dry storage systems and transportation packages where the change would be more appropriate.

Comment 4.2.28: [Section 1.1, Page 1-1, Line 25 in the Draft Report for Comment]

Various commenters stated that, per Interim Staff Guidance 2, Revision 2 (ADAMS Accession No. ML16117A080), the options for demonstrating compliance with the ready-retrieval requirement (per 10 CFR 72.122(l)) were expanded to also include ready-retrieval on a cask or canister basis. The commenters stated that this information should be reflected in NUREG-2224 to aid the reader in determining when the integrity of HBU SNF cladding is relevant to compliance with the ready-retrieval requirement, per 10 CFR 72.122(l).

NRC Response: The staff agrees with the comment. The staff revised Section 1.1 to clarify that the requirement for ready retrieval may be demonstrated by either (A) removing individual or canned spent nuclear fuel (SNF) assemblies from wet or dry storage; (B) removing a canister loaded with SNF assemblies from a dry storage system (DSS) cask or overpack; or (C) removing a DSS cask loaded with SNF assemblies from its storage location. The staff has further clarified that the ready retrieval requirement is defined by the approved design bases for the DSS design or ISFSI's specific license.

Comment 4.2.29: [Section 1.2, Page 1-3, Line 18 in the Draft Report for Comment]

A commenter stated that EPRI Report NP-4524 states that breached PWR fuel rods will not split open from fuel oxidation for 100 years if the rod is not exposed to air until temperature drops below 230°C. The commenter stated that the Draft Report for Comment indicates a lower temperature but provided no supporting reference or information.

NRC Response: The staff disagrees with the comment.

Section 1.2 in NUREG-2224 does not state that fuel rods will split open if the rod is exposed to air at temperatures below 230 °C (446 °F). However, to avoid unnecessary confusion, the staff has clarified that research has shown that the uranium dioxide (UO_2) in the fuel pellet may oxidize (to U_4O_9) at temperatures less than 230 °C (446 °F). The staff has also provided reference citations in support of this statement (see McEachern, R.J. and P. Taylor, "A review of the oxidation of uranium dioxide at temperatures below 400°C," Journal of Nuclear Materials, Journal of Nuclear Materials, Volume 254, Issues 2–3, Pages 87-121, 1998; Jung, H., P. Shukla, T. Ahn, L. Tipton, K. Das, X. He, and D. Basu, "Extended Storage and Transportation: Evaluation of Drying Adequacy," San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses, 2013, ADAMS Accession No. ML13169A039).

Comment 4.2.30: [Section 1.5, Page 1-7, Line 18 in the Draft Report for Comment]

A commenter stated that the use of the word "Infiltrates" in the cited sentence is not an appropriate or objective word. The commenter proposed to replace with "is picked up by" or another word with a more neutral connotation.

NRC Response: The staff agrees with the comment.

NUREG-2224 has been revised per the comment.

Comment 4.2.31: [Section 1.5.3.1 in the Draft Report for Comment]

A commenter suggested that the NRC include the latest set of rod puncture test data from Oak Ridge National Laboratory and Pacific Northwest National Laboratory in NUREG-2224. The commenter stated that the inclusion would add about another 10 data points to the discussion in Section 1.5.3.1.

NRC Response: The staff agrees, in part, with the comment.

The staff revised Section 1.5.3 of NUREG-2224 to clarify that additional empirical data has been obtained at both ORNL and PNNL on end-of-life rod internal pressures since the issuance of the Draft Report for Comment. However, these laboratories have not yet publicly-issued their final reports on these data, and therefore the data was not incorporated into NUREG-2224. The staff will update NUREG-2224, as appropriate, as new information is generated.

Comment 4.2.32: [Section 1.5.4, Page 1-18, Line 20 in the Draft Report for Comment]

A commenter stated that the cooling rate used for radial hydride treatment (i.e., 5 °C per hour), as discussed in Section 1.5.4, should not be considered "slow cooling." The commenter stated that this cooling rate is orders of magnitude faster than in dry storage. The commenter stated that annealing of the cladding occurs at elevated temperatures in zirconium alloys, but keeping temperatures at 400 °C (752 °F) for 1 to 24 hours with subsequent rapid cooling (5 °C per hour) does not allow for sufficient time at temperature for appreciable annealing to occur. The commenter stated that annealing of the cladding would occur if actual fuel rods in dry storage reached these temperatures.

NRC Response: The staff agrees with the comment.

The staff revised the discussion in Section 1.5.4 to remove the "slow" qualifier and to clarify that the chosen cooling rate does not allow for sufficient time at temperature for appreciable annealing of irradiation hardening to occur, thus allowing a separate assessment of the effects of hydride reorientation.

Comment 4.2.33: [Section 1.2, Page 1-2, Line 26 in the Draft Report for Comment]

A commenter suggested adding rod void (plenum) volume to the list of factors affecting rod internal pressure.

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised per the comment.

Comment 4.2.34: [Section 1.2, Page 1-2, Line 29 in the Draft Report for Comment]

A commenter requested that the word "it" be revised in the cited sentence. The commenter was unclear if the NRC meant that it is critical to control the peak cladding temperature to within values that preserve cladding integrity, or if it is critical to control the average gas temperature within the rods.

NRC Response: The staff agrees with the comment.

The staff revised the discussion in Section 1.2 to clarify that an important consideration for demonstrating adequate cladding performance is to control the peak cladding temperature of the fuel rods during vacuum drying and storage/transport operations to temperatures demonstrated to preserve cladding integrity.

Comment 4.2.35: [Section 1.2, Page 1-3, Line 30 in the Draft Report for Comment]

A commenter questioned whether Interim Staff Guidance 11, Revision 3 (ADAMS Accession No. ML033230335), applies to transportation of SNF.

NRC Response: The staff disagrees with the comment. The staff notes that Interim Staff Guidance 11, Revision 3 (ADAMS Accession No. ML033230335), applies to both dry storage and transportation.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.36: [Section 1.3, Page 1-4, Lines 24-25 in the Draft Report for Comment]

A commenter stated that the cited sentence implies that the gas temperature is increasing because the volume increased. The commenter stated that the volume change would not change the temperature, but the instead, the pressure.

NRC Response: The staff agrees with the comment.

The staff revised the discussion in Section 1.3 to clarify that the design and materials used for fabrication of fuel rods are such that the creep of the cladding is self-limited. As the average gas temperature of the fuel rod increases during drying-transfer and storage/transport operations, the gas pressure within the fuel column increases (with a corresponding increase in cladding hoop stresses). If the increase of gas pressure is sufficient to result in cladding creep, the internal volume of the rod will increase, which will, in turn, reduce the gas pressure within the fuel column (with a corresponding decrease in cladding hoop stresses). The net effect is a slow decrease in pressure and hoop stress with increasing creep strain. The stress also

decreases with increasing storage or transport time due to the decrease in rod internal pressure with decreasing temperature.

Comment 4.2.37: [Section 1.5, Page 1-7, Lines 36 to 37 in the Draft Report for Comment]

A commenter stated that, while determining a threshold temperature and hoop stress for hydride reorientation may be difficult, research is ongoing to understand hydride reorientation. The commenter further stated that, from this research, a better metric for ensuring hydride reorientation will not impact cladding performance can be determined. The commenter concluded that ongoing research may indicate that, for higher peak cladding temperatures (i.e., higher than 400 °C (752 °F)), hydride reorientation may not occur.

NRC Response: The staff notes the comment.

See responses to Comment 3.1.8 and Comment 3.4.40.

No changes were made to NUREG-2224 in response to the comment.

Comment 4.2.38: [Section 1.5.2, Page 1-10, Lines 37 to 40 in the Draft Report for Comment]

A commenter stated: that recrystallized-annealed (RXA) claddings are more susceptible to hydride reorientation due to a larger fraction of grain boundaries in the radial direction. The commenter further stated that RXA claddings have lower hydrogen content making hydride reorientation less impactful.

NRC Response: The staff agrees, in part, with the comment.

The staff revised the discussion in Section 1.5.2 to clarify that recrystallized annealed (RXA) claddings have lower hydrogen uptakes during reactor operation than cold-worked stress relief annealed (CWSRA) cladding alloys. The staff has added a reference citation for this statement. However, the discussion in Section 1.5 is sufficiently adequate in addressing the parameters (e.g., cladding hydrogen content) affecting the degree of hydride reorientation.

Comment 4.2.39: [Section 1.5.3.1, Page 1-12, Figure 1-3 in the Draft Report for Comment]

A commenter made an observation that the Oconee Unit 1 data in Figure 1-3 gives a wide range of rod internal pressures for a very small range of burnup. The commenter stated that these data are not very conducive to developing a correlation between burnup and rod internal pressure.

NRC Response: The staff disagrees with the comment. The staff notes that Figure 1-3 includes data from multiple rods from various power units. The fact that the Oconee-1 data may not span a large range of burnups does not impact the general trend of all the data in Figure 1-3.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.40: [Section 1.5.5, Page 1-25, Line 17 in the Draft Report for Comment]

A commenter stated that there would be some pellet swelling which would reduce the original pellet-cladding interface gap, leaving minimal deflection of the cladding in the pinch loading mode before the pellet began to provide significant resistance.

NRC Response: The staff agrees with the comment.

The staff revised the discussion in Section 1.5.5 of NUREG-2224 to clarify that the gap at the pellet-cladding interface is generally closed at rod segments irradiated to high burnup due to pellet expansion during irradiation. The closed gap is expected to limit the deflection of the cladding before encountering any resistance by the pellet.

Comment 4.2.41: [Section 1.3, Page 1-4, Lines 24 to 25 in the Draft Report for Comment]

A commenter requested that the sentence on Line 24 and continuing on Line 25 be deleted. The commenter further stated that, given the low amount of energy released by decay heat, the average gas temperature will increase by an infinitesimal amount.

NRC Response: The staff disagrees with the comment.

The staff notes that the cited statement does not quantify the increase in temperature, and therefore a revision is not warranted.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.42: [Section 1.5 in the Draft Report for Comment]

A commenter stated that the discussion in Section 1.5 is not very robust and a bit too simple. The commenter then stated that the discussion does not address the distinction between cold work stress relieved and recrystallized-annealed cladding microstructures.

NRC Response: The staff disagrees with the comment.

Section 1.5.2 discusses the differences between the microstructures of cold-worked stress relieved annealed (CWSRA) and recrystallized annealed (RXA) cladding alloys and provides appropriate references to the reader for additional information.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.43: [Section 1.5, Pages 1-7 and 1-8 in the Draft Report for Comment]

A commenter stated that Section 1.5 does not mention the impact of some fuel designs, which is a major topic for boiling water reactor (BWR) claddings (but not for pressurized water reactor claddings, at least in the United States).

NRC Response: The staff notes the comment.

The staff notes that the comment is not clear as to what specific fuel designs were not considered in Section 1.5 of NUREG-2224. However, to improve the discussion, the staff has added clarification that some BWR fuel designs incorporate zirconium liners in the Zircaloy-2 cladding at the inner diameter surface, which occupies about 10% of the cladding thickness. The liners are metallurgically bonded to the Zircaloy-2 tube and consist of zirconium alloyed with varying amounts of iron. The addition of iron improves corrosion resistance in the case of fuel failure. In Zircaloy-2 cladding with a zirconium liner, hydrogen is observed to diffuse preferentially to the liner as cooling rates decrease. Such preferential diffusion results from the lower solubility of hydrogen in pure zirconium relative to the solubility in Zircaloy-2.

Comment 4.2.44: [Section 1.5.3, Page 1-11, Lines 2-14 in the Draft Report for Comment]

A commenter stated that Section 1.5.3 is weak since there is no mention of the impact of an inner liner, which is present in most of the BWR rods. The commenter stated that this should be a part of the discussion in the first paragraph of Section 1.5.3. The commenter cited a presentation given at an Extended Storage and Collaborative Program meeting on May 4, 2015).

NRC Response: The staff agrees with the comment.

The staff revised the discussion in Section 1.5.3 (in response to Comment 4.2.49) to address the relevance of the inner liner in BWR rods. The staff considers those revisions to be sufficiently adequate to address this comment.

No additional revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.45: [Section 1.5.4, Page 1-21 in the Draft Report for Comment]

A commenter stated that data on M5® cladding, as obtained at Argonne National Laboratory, are not really applicable to hypothetical dry storage conditions starting from 400 °C (752 °F). The applicant stated that this was addressed in a presentation on the Electric Power Research Institute's Nuclear Fuel Industry Research Program. The applicant stated that the NRC did not participate in this meeting.

NRC Response: The staff notes the comment.

The staff notes that the comment is not clear on the exception raised, and the staff did not receive the referenced supporting presentation in the comment submission.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.46: [Section 1.5.5, Page 1-26, Line 3 in the Draft Report for Comment]

A commenter stated that the Electric Power Research Institute (EPRI) had previously concluded that the fuel pellet imparts structural support to the SNF rod. The commenter cited a reference (EPRI Report 1009929) in support of the assertion, and requested that this reference be cited in support of the assertions in NUREG-2224."

NRC Response: The staff agrees with the comment.

The staff revised NUREG-2224 to clarify that the contribution of the fuel pellet to the flexural rigidity of the rod has been historically ignored because of the lack of experimental bending test data, although it has been previously evaluated by finite element analysis per the reference cited by the commenter. The revised discussion includes the reference citation proposed by the commenter.

Comment 4.2.47: [Section 1.5.5, Page 1-25, Line 29 and Section 2.3.3, Page 2-13, Line 27 in the Draft Report for Comment]

A commenter stated that a number of sections in NUREG-2224 refer to NRC's expectations on the results of future HBU SNF testing to be performed under DOE-sponsorship. The commenter, further stated that, with the testing currently underway, it seems that it would be better to hold off final publication of NUREG-2224 until this testing is completed.

NRC Response: The staff disagrees with the comment.

The assessment from the NRC-sponsored research in NUREG-2224 allows for an adequate evaluation of the effects of hydride reorientation during postulated design-basis drop accidents in dry storage, normal conditions of transport (NCT), or hypothetical accident conditions (HAC) during transportation (see Sections 2.3.4 and 2.4.3 of NUREG-2224 for the NRC's conclusions

based on the engineering assessment of Cyclic Integrated Reversible Fatigue Tester (CIRFT) results in NUREG/CR-7198, Revision 1 (ADAMS Accession No. ML17292B057).

Section 2.3.4 of NUREG-2224 concludes that additional CIRFT static bending conducted under DOE sponsorship will provide additional confirmation of the conclusions in NUREG-2224. As discussed in Section 2.3.3, the NRC considers that the orientation of the hydrides is not a critical consideration when evaluating the adequacy of cladding-only mechanical properties. The properties necessary to implement this alternative are derived from cladding-only uniaxial tensile tests and include modulus of elasticity, yield stress, ultimate tensile strength and uniform strain, and the strain at failure (i.e., the elongation strain). Additional considerations for acceptable cladding-only mechanical properties (i.e., alloy type, burnup, and temperature) may be found in either of the current Standard Review Plans (SRPs) for dry storage of SNF (NUREG-1536, Revision 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility,” (NRC, 2010) (ADAMS Accession No. ML101040620) for the review of applications for Certificates of Compliance under 10 CFR Part 72; and NUREG-1567, “Standard Review Plan for Spent Fuel Storage Facilities,” (NRC, 2000a) (ADAMS Accession No. ML003686776) for the review of applications for specific licenses under 10 CFR Part 72)) or transportation (NUREG-1617, “Standard Review Plan for Transportation Packages for Spent Nuclear Fuel,” (NRC, 2000b) (ADAMS Accession No. ML003696262)).

The staff recognizes that additional data on CIRFT dynamic bending response for modern cladding fuel rods will be obtained under DOE sponsorship. The staff expects this data will be needed for the evaluation of vibration normally incident to transport (per the requirement in 10 CFR 71.71(c)(5). That said, the staff has described and implemented a fatigue cumulative damage model per the CIRFT dynamic bending results in NUREG/CR-7198, Revision 1 for Zircaloy-4-clad HBU SNF. The NRC has noted that the same model may be used by applicants once new CIRFT dynamic bending data are available for HBU SNF with other cladding alloys. Therefore, the absence of this data should not prevent the issuance of NUREG-2224.

The staff also recognizes that additional data on the performance of HBU SNF will be obtained over time. The NRC will continue to periodically evaluate that data and determine whether revisions to NUREG-2224, the current Standard Review Plans, or other guidance documents are appropriate.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.48: [Section 1.3, Page 1-4, Line 22 in the Draft Report for Comment]

A commenter requested to add the words “storage and transport” before “operations” in the cited sentence.

NRC Response: The staff notes the comment.

The staff notes the comment is unclear. The cited paragraph does not contain the word "operations," therefore the proposed revision by the commenter is not warranted.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.49: [Section 1.5.4, Page 1-19, Lines 19 and 24 in the Draft Report for Comment]

A commenter requested clarification on the meaning of the term "figure of merit", which the NRC used in Section 1.5.4.

NRC Response: The staff agrees with the comment.

The staff recognizes that there is not a clear definition for the term. Therefore, the staff has removed the statement from the report.

Comment 4.2.50: [Section 1.5.4, Page 1-20, Line 21 in the Draft Report for Comment]

A commenter questioned if there is a need to limit the number of thermal cycles experienced by HBU SNF during operations (i.e., the maximum of 10 cycles per Interim Staff Guidance 11, Revision 3) if the results in NUREG-2224 suggest no effect on radial hydrides from multiple cycles.

NRC Response: The staff disagrees with the comment.

The discussion in Section 1.2 addresses the technical basis for the safety review guidance in Interim Staff Guidance 11, Revision 3 (ADAMS Accession No. ML033230335), including a discussion on the effects of thermal cycling on hydride reorientation.

See also response to Comment 3.4.48 for how the NRC has used the results in NUREG-2224 to provide additional flexibility for thermal cycling during operations.

Comment 4.2.51: [Section 1.5.5, Page 1-24, Lines 25-28 in the Draft Report for Comment]

A commenter requested clarification on the criteria that must be assumed in an analysis showing hypothetical reconfiguration of HBU SNF.

NRC Response: The staff notes the comment. The purpose of the cited statement is to provide perspective of NRC's prior position regarding the licensing and certification of dry storage systems and transportation packages for high burnup spent nuclear fuel. The staff directs the reader to the discussions in Sections 3.2.4.2 and 4.2.4.2 of NUREG-2224 on the revised NRC

position on the use of supplemental analyses relying on HBU SNF reconfiguration and the associated criteria.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.52: [Section 1.3, Page 1-5, Line 1 in the Draft Report for Comment]

A commenter questioned whether the cited statement specifically pertains to the HBU SNF demonstration program being sponsored by the DOE (and contracted to the Electric Power Research Institute).

NRC Response: The staff notes the comment. The discussion in Section 1.3 is not specific to the DOE's Research Cask Program on HBU SNF. The discussion does not preclude other surrogate surveillance and monitoring programs from being used to confirm the conclusions of the accelerated short-term testing. The discussion properly references the guidance in NUREG-1927, Revision 1 (ADAMS Accession No. ML16179A148), which defines staff's expectations for the use of a surrogate surveillance and monitoring program.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.2.53: [Section 1.5.3.1, Page 1-12, Lines 1 to 3 in the Draft Report for Comment]

A commenter stated that the cited paragraph could easily be misread by a member of the public and recommended it be changed/clarified so the last sentence is eventually referring to Figure 1-5 and not the discussion M5® data. The commenter stated that this is mainly an issue of the order of presenting information.

NRC Response: The staff agrees with the comment.

The staff revised the discussion in Section 1.5.3 (which consolidates the discussion on Section 1.5.3.1 previously on the Draft Report for Comment – see response to Comment 4.2.9) to provide a footnote stating that publicly-available empirical EOL RIP data are available for ZIRLO™-clad SNF rods but not for M5®-clad SNF rods. The revision allows for a continuous flow of the discussion regarding the data in Figure 1-5.

Comment 4.2.54: [Section 1.2, Page 1-3, Line 18 in the Draft Report for Comment]

A commenter stated that a reference is needed for the term "later research" in the cited sentence.

NRC Response: The staff agrees with the comment. The staff revised Section 1.2, to clarify that research has shown that the uranium dioxide (UO_2) in the fuel pellet may oxidize (U_4O_9) at temperatures less than 230 °C (446 °F), and has provided supporting reference citations (see McEachern, R.J. and P. Taylor, “A review of the oxidation of uranium dioxide at temperatures below 400°C,” Journal of Nuclear Materials, Journal of Nuclear Materials, Volume 254, Issues 2–3, Pages 87-121, 1998; Jung, H., P. Shukla, T. Ahn, L. Tipton, K. Das, X. He, and D. Basu, “Extended Storage and Transportation: Evaluation of Drying Adequacy,” San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses, 2013, ADAMS Accession No. ML13169A039).

Comment 4.2.61: [Section 1.3, Page 1-4; Lines 9 to 10 in the Draft Report for Comment]

A commenter stated that the term "which" be deleted on Line 9 and Line 10 entirely.

NRC Response: The staff agrees with the comment.

NUREG-2224 has been revised per the comment.

4.3 Chapter 2

Comment 4.3.1: [Section 2.1, Page 2-1, Lines 4 to 6 in the Draft Report for Comment]

A commenter noted that the first sentence implies that the canister is relied upon for confinement in transportation, but that the transportation package provides a containment function. The commenter suggested rewording the sentence to clarify intent of this statement.

NRC Response: The staff agrees with the comment. The staff has revised Section 2.1 to clarify that either the sealed canister, cask cavity, or overpack generally serves as the primary barrier in a dry storage system or transportation package for protecting against the release of radioactive solid particles or gases from the loaded spent nuclear fuel to the environment.

Comment 4.3.2: [Section 2 in the Draft Report for Comment]

A commenter noted that “HBR” does not appear to be defined in the text or appear in the acronyms.

NRC Response: The staff disagrees with the comment.

The acronym was defined in the “Abbreviations and Acronyms” introductory section of NUREG-2224, as well as the first time it was used in Section 2.2 of the report.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.3: [Section 2 in the Draft Report for Comment]

A commenter stated that the extent of the cladding types tested using the Cyclic Integrated Reversible Fatigue Tester does not appear to be discussed.

NRC Response: The staff disagrees with the comment.

Section 2.2 of the report describes the HBU SNF rod segments tested under the NRC's test program as having intact Zircaloy-4 cladding irradiated at the H.B. Robinson Steam Electric Plant.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.4: [Section 2.1, Page 2-1, Line 24 in the Draft Report for Comment]

A commenter indicated that Section 2 could benefit from a sentence at the end of the introduction stating the purpose of each subsection. The commenter proposed the following text, as an example: "Section 2.2 provides a discussion of the available fuel rod composite static and dynamic performance empirical data and its acquisition, Section 2.3 describes the application of the empirical data in transportation applications and the development of a composite rod analytical model, and Section 2.4 discusses the application of the empirical data for storage and transportation fatigue evaluations."

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised in response to the comment.

Comment 4.3.5: [Section 2.3.4, Page 2-15, Figure 2-9 in the Draft Report for Comment]

A commenter provided specific comments on Figure 2-9 in the Draft Report for Comment. The commenter stated: "Figure 2-9 references: the radial-hydride continuity factor (RHCF) used by Argonne and referred to in this NUREG requires continuity or radial-circumferential hydrides. Figure 2-9 is not high enough in magnification or quality to assess continuity. However, it is clear enough that relatively few radial hydrides are present in the outer 1/3 of the cladding wall for RHCF to be 100%. The following two images represent enlargement (1st) and enlargement-enhancement (2nd) of the Figure 2-9 image. Based on both of them, it would be more accurate to say that the RHCF is >50% in this image. Also, Argonne takes about 30 images at 100X to cover the whole cross section of a metallographic sample. The maximum length of continuous radial-circumferential hydrides (projected onto the radius) is determine for each image. Argonne reports the average and one-sigma variation of the data from the 30 images. Assuming that Figure 2-9 represents the longest radial hydrides observed, the RHCF defined by Argonne may be below 50%."

NRC Response: The staff agrees with the comment. Section 2.3.4 has been revised to indicate that the conservative conditions of the radial hydride treatment induced a radial hydride continuity factor (RHCF) exceeding 50 percent in parts of the cladding thickness, as shown in Figure 2-9.

Comment 4.3.6: [Section 2.2, Page 2-3, Line 5 in NURGE-2224 In the Draft Report for Comment]

A commenter noted that it was mentioned in the public meeting on the Draft Report for Comment (September 6, 2018) that the burnups of the HBU SNF rods tested under the NRC-sponsored research program was in the 60 GWd/MTU range and recommended including this information in the final NUREG-2224.

NRC Response: The staff agrees with the comment.

The introductory material of Section 2.2 was revised to clarify the rod-average burnup of the HBU SNF rods tested in the NRC-sponsored research program. Table 2-1 was also added to identify the burnups for each test segment.

Comment 4.3.7: [Section 2.4.3, Page 2-23, Line 32 in the Draft Report for Comment]

Various commenters noted that the 100 MPa number will need to be corrected based on the improved accuracy on rod internal pressure information.

NRC Response: The staff agrees with the comment.

The staff revised Section 2.4.3 to clarify that in an actual spent fuel rod there is internal gas pressure, which creates a hoop stress on the order of 90 megapascals (MPa) (1.3×10^4 pounds per square inch absolute (psia)) or less. The discussion refers the reader to Section 1.5.3 (revised in response to Comment 4.2.9), which discusses the technical basis for the 90 MPa (1.3×10^4 psia) value.

Comment 4.3.8: [Section 2.3.4, Page 2-17, Lines 9 to 11 in the Draft Report for Comment]

A commenter stated that the intent of cited paragraph seems to be that a bounding hoop stress of 140 MPa is a good bounding number. The commenter further stated that the paragraph also states that future testing of HBU SNF under DOE-sponsorship is expected to confirm this for ZIRLO™-clad and M5®-clad HBU SNF. The commenter stated that the determination of 140 MPa as a bounding hoop stress is based on erroneous data on rod internal pressures for integral fuel burnable absorber rods, and therefore, the NRC should lower this 140 MPa bounding number based on correct rod internal pressure data from actual fuel rods."

NRC Response: The staff notes the comment.

The staff notes that Section 1.5.3 on end-of-life rod internal pressures and cladding hoop stresses was revised in response to Comment 4.2.9. The revised discussion addresses maximum cladding hoop stresses for pressurized water reactor (PWR) integral fuel rod burnable absorber (IFBA) rods, which are expected to remain below 90 megapascals (MPa) (1.3×10^4 pounds per square inch absolute (psia)) at a peak cladding temperature of 400 °C (752 °F). Based on this revised discussion, the staff agrees that 140 MPa (2.0×10^4 psia) is a bounding cladding hoop stress relative to the maximum cladding hoop stresses for PWR IFBA rods (i.e., the aforementioned 90 MPa (1.3×10^4 psia) value).

The NRC recognizes that the bounding hoop stress of 140 MPa (2.0×10^4 psia) used for the testing of the HBU SNF segments in the NRC-sponsored program is based on Oak Ridge National Laboratory (ORNL) analyses, which did not properly implement the FRAPCON fuel performance code (see Bratton, et al., "Rod Internal Pressure Quantification and Distribution Analysis Using FRAPCON," FCRD-UFD-2015-000636, ORNL/TM-2015/557, 2015). However, the fact that a higher cladding hoop stress (140 MPa (2.0×10^4 psia)) was chosen for the NRC - sponsored testing provides additional confidence for the staff of the adequacy of the conclusions in Section 2.3.4. This is because the higher cladding hoop stress used for testing is expected to result in a higher extent of hydride reorientation. Therefore, with respect to the technical basis in NUREG-2224, the fact that ORNL's analyses may have yielded higher cladding hoop stresses served to provide increased confidence to the staff that hydride reorientation is not an important consideration for the analyses of design-basis drop accidents in dry storage and transportation.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.9: [Section 2.4.3, Page 2-23, Line 29 in the Draft Report for Comment]

A commenter noted that, while this sentence indicates that the fatigue test results cannot be applied to thermal fatigue in storage, thermal fatigue is not an issue for storage. The commenter recommended adding a clarifying statement to that effect.

NRC Response: The staff agrees with the comment.

The staff revised Section 2.4.3 to provide a reference to NUREG-2214 (ADAMS Accession No. ML17289A237), which addresses thermal fatigue of SNF cladding during dry storage. The discussion in Section 2.4.3 already addresses the staff's conclusions regarding the applicability of the CIRFT data to thermal fatigue in dry storage.

Comment 4.3.10: [Section 2.3.4, Page 2-14, Line 35 in the Draft Report for Comment]

A commenter questioned the NRC's decision to apply 5 thermal cycles to the HBU SNF tested under the NRC-sponsored research program discussed in NUREG-2224. The commenter stated that real HBU SNF would not experience these number of cycles.

NRC Response: The staff disagrees with the comment.

As stated in the introductory paragraph of Section 2.3.4, the staff chose a conservative testing approach (radial hydride treatment) to maximize the fraction of cladding radial hydrides precipitated in the HBU SNF test segments. Section 2.3.4 also states that thermal cycling was repeated for five cycles to further induce a higher fraction of radial hydrides. This conservative approach allowed the staff to make generalizations on hydride reorientation's effects on modern cladding alloys, as discussed later in Section 2.3.4 of the report.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.11: [Section 2.4.3, Page 2-23, Line 23 in the Draft Report for Comment]

A commenter stated that daily, and especially seasonal, fluctuations in temperature will be very slow and should not be considered a fluctuating or "cyclic" load.

NRC Response: The staff notes the comment.

The discussion on Section 2.4.3 provides a comprehensive discussion of cyclic loads expected during dry storage and transportation, including those induced by temperature variations. Temperature fluctuations during storage will result in cyclic hoop stresses in the cladding due to internal pressure fluctuations. The purpose of the discussion in Section 2.4.3 is to explain that the fatigue test results discussed in Section 2.4 are not applicable to loading conditions that produce fluctuations in hoop stress. Therefore, the fatigue test results cannot be applied to thermal fatigue during dry storage. The staff notes that Section 2.4.3 was revised (in response to Comment 4.3.9) to provide a reference to NUREG-2214 (ADAMS Accession No. ML17289A237), which provides additional information on thermal fatigue of SNF cladding during dry storage.

No additional revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.12: [Section 2.3.2, Pages 2-6 and 2-7 and Figure 2-6 in the Draft Report for Comment]

A commenter stated that Figure 2-6 seem to contradict the behavior model discussed in pages 2-6 and 2-7 of Section 2.3.2 in the Draft Report for Comment. The commenter noted that, if

stress and strain go up in the cladding at the fuel pellet interfaces, the model of the fuel rod as a homogenous solid is not appropriate. The commenter further recommended that some discussion be added about the resolution of the differences between the model discussed on pages 2-6 and 2-7 and real discrete pellet effects in the structural performance of a HBU SNF rod.

NRC Response: The staff notes the comment.

The discussion in the cited pages of Section 2.3.2 provide a general discussion of composite systems and specifically a fuel rod as a composite system by assuming the fuel is a homogenous uncracked solid. This discussion only serves to illustrate that, if the centers of gravity of the fuel and cladding are coincident, then the flexural rigidity of the composite section is equal to the sum of the flexural rigidities of the individual components regardless of whether the components are bonded or unbonded.

In order to address the potential confusion, the discussion in Section 2.3.2 of the Draft Report for Comment was separated in the final NUREG-2224. The new section (i.e., Section 2.3.3 of the final NUREG-2224) delineates a separate discussion on the calculation of cladding strain from CIRFT static test data, which recognizes that the actual behavior of the fuel rod where the fuel is no longer a homogenous solid as previously discussed in Section 2.3.2 (i.e., the fuel pellets separate at their interface during bending). The staff notes that the new Section 2.3.3 addresses the real discreet pellet effects in the structural performance of a HBU SNF rod. Further, the staff notes that Figure 2-6 pertains to the discussion of the new Section 2.3.3 in NUREG-2224, and not to the pages in Section 2.3.2 cited by the commenter.

Comment 4.3.13: [Section 2.3.3, Page 2-13, Line 16 in the Draft Report for Comment]

A commenter pointed out that "… maximum equivalent tensile stress…" in the cited statement appears to be referencing axial stress and not the von Mises stress. The commenter stated that the word "equivalent" might cause confusion because "equivalent stress" can mean von mises stress depending on the context.

NRC Response: The staff agrees with the comment. The staff has revised Section 2.3.3 to remove the word "equivalent".

Comment 4.3.14: [Section 2.3.3, Page 2-13, Lines 19 to 28 in the Draft Report for Comment]

A commenter stated that, although the methodology laid out in NUREG-2224 is reasonable for calculating nominal stresses and providing a reasonable assurance of safety, it doesn't address pellet gaps and the effect they can have on cladding stress. The commenter noted that this might need to be addressed based on results from future DOE-sponsored research on HBU SNF.

NRC Response: The staff disagrees, in part, with the comment.

The staff notes that Section 2.3.3 of the final NUREG-2224 already addresses the performance of an SNF rod as a non-homogeneous solid (i.e., assuming pellet to pellet cracks during bending).

The staff notes that results obtained under the DOE-sponsored research on HBU SNF will be monitored, including any additional data generated to understand the effects of pellet gaps on the structural performance of HBU SNF.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.15: [Section 2.3.4.2, Page 2-19, Line 22 in the Draft Report for Comment]

A commenter requested the NRC clarify the basis for the 80 N m in the cited calculation. Although the commenter thought that the NRC meant that acceptance limits based on bending moments can be used from CIRFT test data if they are available, the commenter thought that the NRC was showing an example calculation, and not a general conclusion for all HBU SNF dry storage systems and transportation packages. If the NRC's intent is different, the commenter recommended clarifying this.

NRC Response: The staff agrees with the comment.

In Section 2.3.5.2 of the NUREG-2224 (previously Section 2.3.4.2 of the Draft Report for Comment), the staff has clarified that lower-bound maximum bending N m (as also stated in Section 2.3.4 of the final NUREG-2224). Further, the staff has added clarification to Section 2.3.5.2, that the cited calculation is an example on how to define the safety margin based on the lower-bound maximum bending moment achieved in CIRFT tests. The staff implemented the calculation based on the CIRFT tests in the NRC-sponsored research program (i.e., the staff implemented the calculation using the lower-bound maximum bending moment of 80 N m achieved during testing of Zircaloy-4-clad HBU SNF segments).

The commenter is also referred to the Section 2.3.5.1 in the NUREG-2224, which states that, using 80 N·m provides a conservative basis for calculating safety margin. However, to quantify the safety margin, it is necessary to know the bending moment in the fuel rod as a function of the g-load acting on the rods due to a side drop event. The g-loads acting on the rods are dependent on each dry storage system (DSS) or transportation package. Therefore, neither Section 2.3.5.1 and Section 2.3.5.2 of the final NUREG-2224 are meant to provide generic conclusions but rather to provide example calculations that applicants may replicate for their specific DSS designs or transportation packages.

Comment 4.3.16: [Section 2.4.1, Page 2-21, Table 2.5 in the Draft Report for Comment]

A commenter recommended a clarification as to whether the NRC considers the 0.060% strain amplitude defined in Table 2.5 of the Draft Report for Comment to be the fatigue limit, or just the practical extent of CIRFT data obtained from the testing of Zircaloy-4-clad HBU SNF segments. The commenter noted that, while Page 2-4, Line 24 in the Draft Report for Comment, mentions the likely existence of a fatigue limit, the section reads as if NRC is not endorsing the idea.

NRC Response: The staff agrees with the comment.

The staff has provided clarification in Section 2.4.1 of NUREG-2224. The staff recognizes that some materials, like steel, have fatigue endurance limits. However, other materials, like aluminum, do not have fatigue endurance limits. The staff does not consider that there is sufficient CIRFT dynamic bending test data to determine whether the various irradiated zirconium-based alloys (i.e., Zircaloy-2, Zircaloy-4, ZIRLO™, M5®) used for cladding materials have fatigue endurance limits. Therefore, the staff does not identify a fatigue endurance limit in NUREG-2224.

Comment 4.3.17: [Section 2.3.2, Page 2-6, Line 2 in the Draft Report for Comment]:

A commenter questioned the intent of Section 2.3.2 and suggested the section be modified to clarify the intent.

NRC Response: The staff agrees with the comment.

The staff has revised Section 2.3.2 to clarify that the intent is to explain the structural behavior of composite systems, in general, followed by a discussion to specifically address the spent fuel rod composite system assuming the fuel is a homogenous, uncracked solid.

Comment 4.3.18: [Section 2.3.2, Page 2-6, Line 2 in the Draft Report for Comment]:

A commenter suggested the term "normal explanation" in the cited statement be clarified.

NRC Response: The staff agrees with the comment.

Section 2.3.2 was revised to remove the cited statement and, instead, state that "A spent fuel rod is considered to be a composite system consisting of cladding and fuel. The structural response of the fuel-rod composite is usually explained as follows."

Comment 4.3.19: [Section 2.3.2, Page 2-6, Line 2 in the Draft Report for Comment]:

A commenter indicated that Section 2.3.2 is unclear on whether the discussion assumes that the SNF rods behave like a linear, continuous, composite beam, with homogenous solid fuel.

NRC Response: The staff agrees with the comment.

The cited sentence in Section 2.3.2 was revised to clarify that the fuel is assumed to be a homogenous, uncracked solid.

Comment 4.3.20: [Section 2.3.2, Page 2-6, Line 6, Equation 2-1 in the Draft Report for Comment]

A commenter stated that the fuel has been modeled as homogenous elastic beam elements in structural dynamic applications per the equation: $EI = E_c I_c + xE_p I_p$, where x is a variable parameter with a value between 0 and 1. The commenter stated that the contribution of the stiffness of the fuel pellet has been and unknown, and therefore the value of x has also been an unknown. The commenter suggested that a value of x = 0.5 may be a good rule of thumb. However, the commenter stated that the staff's discussion in Section 2.3.2 suggests that x = 1 may be more appropriate, although the commenter stated that values between 0 and 1 would still demonstrate adequate performance of the cladding during normal conditions of transport and design-basis drop accidents.

NRC Response: The staff notes the comment. The staff notes that, if the fuel is assumed to be a homogenous solid (per the discussion in Section 2.3.2), the value of x in the equation cited by the commenter is always 1.0. However, the staff has chosen a different approach from that described the comment. Rather than describing the fuel rod flexural rigidity as $EI = E_c I_c + xE_p I_p$ (per the comment), the staff chose to describe the fuel rod flexural rigidity by an equation of the form $EI = xE_c I_c$, where, x, accounts for the increase in cladding flexural rigidity due to the presence of the fuel pellet (see Equation 2-4 in NUREG-2224). The staff chose this approach because, when evaluating cladding stresses during normal conditions of transport and design-basis accidents, applicants generally use cladding-only mechanical properties and add the mass of the fuel to the cladding. Multiplying the cladding properties by a factor (x) to account for the presence of the fuel pellet simplifies the calculation and removes the need to know the value of E_p. The commenter is referred to Section 2.3.4 of the final NUREG-2224, which provides the basis for the value of x.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.21: [Section 2.3.2, Page 2-7, Lines 22 to 23 in the Draft Report for Comment]

A commenter expressed disagreement with the statement "Thus, for a spent fuel rod, where the fuel is a homogenous solid, the flexural rigidity is given by Equation 2-1, regardless of whether the fuel is bonded to the cladding." The commenter stated: "I don't think the test data agrees with this statement. You can test this analytically by using Equation 2-1 to calculate an EI, using a closed-form solution for beam bending, and plotting the response in Figures 2-7 or 2-8. The linear relationship should have the straight line shape of the first segment of the PNNL curve in Figure 2-7. If you do this, I would expect one EI, not the three EI regions discussed in the test data."

NRC Response: The NRC notes the comment.

In response to Comment 4.3.12, the discussion in Section 2.3.2 of the Draft Report for Comment was separated in the final NUREG-2224. The new section (i.e., Section 2.3.3 of the final NUREG-2224) delineates a separate discussion on the calculation of cladding strain from CIRFT static test data, which recognizes that the actual behavior of the fuel rod where the fuel is no longer a homogenous solid as previously discussed in Section 2.3.2 (i.e., the fuel pellets separate at their interface during bending). The discussion in the new Section 2.3.3 and later sections recognize that Equation 2-1 is not an appropriate approach to evaluate the flexural rigidity of a rod, since the fuel does not behave as a homogenous solid during bending. The commenter is referred to Section 2.3.4, which discusses a conservative methodology for calculating cladding stress and strain and fuel rod flexural rigidity.

See also response to Comment 4.3.20.

No revisions were made to NUREG-2224 in response to this comment.

Comment 4.3.22: [Section 2.3.2, Page 2-7, Lines 22 to 23 in the Draft Report for Comment]

A commenter asked if the sentence, "Thus, for a spent fuel rod, where the fuel is a homogenous solid, the flexural rigidity is given by Equation 2-1, regardless of whether the fuel is bonded to the cladding.", was intended to convey that that fuel is a discontinuous solid instead of a continuous solid, so real fuel does not behave according to Equation 2-1.

NRC Response: The staff notes the comment.

The experimental CIRFT results discussed in Sections 2.2 and 2.3.3 demonstrate that SNF rods do not behave according to Equation 2-1. The cited sentence in Section 2.3.2 is making a statement about the behavior of a SNF rod assuming the fuel is a homogenous solid.

See also response to Comment 4.3.12.

No revisions were made to NUREG-2224 in response to this comment.

Comment 4.3.23: [Section 2.3.2, Page 2-7, Lines 22 to 23 in the Draft Report for Comment]

A commenter stated, "If my memory is correct, FEA of the stiffness contribution of pellets inside cladding was studied by INL around 2013, and the conclusion was that when pellets were constrained (had nowhere to move or slide relative to the cladding) that Equation 2-1 would be correct. In that special case, this statement in NUREG-2224 is correct. However, if the fuel pellets have room to move, such as into plenum space or into pellet-pellet gaps, then effective EI would be less than Equation 2-1. With ORNL finding a 5 mm pellet-pellet gap in a sister rod, I think pellet gap effects needs to be addressed in discussions of fuel rod EI and stress calculations."

NRC Response: The agrees with the comment.

The staff notes that discussion in Sections 2.3.2 through 2.3.4 are consistent with the comment.

See response to Comment 4.3.12 regarding revisions to NUREG-2224 to clarify the homogeneous vs. non-homogeneous structural performance of the fuel pellet in HBU SNF rods.

See response to Comment 4.3.26 on the applicability of Equation 2-1.

See response to Comment 4.3.14 on pellet-to-pellet gaps.

No revisions were made to NUREG-2224 in response to this comment.

Comment 4.3.24: [Section 2.2, Page 2-3, Line 33 and Page 2-4, Lines 21 and 29 in the Draft Report for Comment]

A commenter asked if the cited statements should be clarified to refer to HBU SNF rod segments irradiated at the H. B. Robinson Steam Electric Plant Unit 2.

NRC Response: The staff notes the comment.

The staff notes that, as stated in Section 2.2, all the rods tested under the NRC-sponsored research program were of high burnup (see Table 2-1) and irradiated at the H. B. Robinson (HBR) Steam Electric Plant Unit 2.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.25: [Section 2.2, Page 2-4, Line 3 in the Draft Report for Comment]

The commenter asked whether the cited statement should be revised to state: "The test methodology and its corresponding method to measure and calculate cladding stress and strain is applicable to current commercial power fuel rod types".

NRC Response: The staff agrees with the comment.

The sentence was revised to read: "The CIRFT test methodology and the methodology developed in NUREG-2224 for calculating cladding stress and strain are applicable to all current commercial power fuel rod types, and the use of cladding-only properties to calculate cladding stress and strain is always conservative."

Comment 4.3.26: [Section 2.3.2, Page 2-6, Lines 10 and 14 in the Draft Report for Comment]

A commenter suggested that the NRC consider starting a new paragraph on Line 10 just before "on the other hand…." Also, the commenter recommended, at Line 14, consider adding a subsection "Example of the use of a composite section in a non-nuclear application." Finally, the commenter suggested moving the example to an appendix.

NRC Response: The staff agrees in part with the comment.

NUREG-2224 was revised in response to the comment. The staff considers that moving the example to an appendix would interrupt the flow of the discussion.

Comment 4.3.27: [Section 2.3.4.2, Page 2-19, Line 22 in the Draft Report for Comment]

A commenter noted that the discussions on safety margin in Section 2.3.4.2 of the Draft Report for Comment be clarified as the margin to rod failure in bending.

NRC Response: The staff agrees with the comment.

The cited sentence was revised from "fuel rod failure" to "fuel rod bending failure."

Comment 4.3.28: [Section 2 in the Draft Report for Comment]

The commenter stated that "...there is discussion on the bonding between the pellet and the cladding. There are two types of bonding that can be described: a chemical bond between the pellet and the cladding, and a mechanical bond resulting from residual compression of the cladding on the pellets. While the end result of either bonding is the composite behavior

described in the text, the chemical bonding is likely irreversible, while the mechanical bonding may change over time."

NRC Response: The staff agrees with the comment.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.29: [Section 2 in the Draft Report for Comment]

In the sections deriving dynamic and seismic response of a rod in transportation, likely the section headers should include the word "transportation."

NRC Response: The NRC disagrees with the comment.

The staff notes that Section 2.4.3, which addresses how the CIRFT dynamic bending test data can be implemented by applicants, provides discussions on the applicability of these data to transportation and dry storage.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.30: [Section 2 in the Draft Report for Comment]

The commenter stated that Section 2 would be more understandable if all of the solid mechanics work were collected into a single subsection (i.e., showing how the composite section is calculated, showing the difference between cladding only and composite rods, the net effect of hydride reorientation, and how impact conditions are addressed mechanically) and then discussing the various applications (transportation fatigue, dry storage accidents, transportation accidents). The commenter further stated that the discussion on the results of the fatigue testing should be in the discussion of the test, i.e., section 2.2.

NRC Response: The NRC disagrees with the comment.

The current flow of information in Section 2 of NUREG-2224 is adequate since no other commenters expressed concern.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.31: [Section 2 in the Draft Report for Comment]

A commenter stated that the use of the discussion on the opening of the pellet-pellet interfaces is confusing because the word "cracks" is used, suggesting that a pellet crack has occurred.

The commenter expressed belief that the discussion is not about the development of additional cracks in the pellet body, but about the development of openings between pellet ends as bending is increased. The commenter suggested that the discussion be modified to remove the mention of cracking.

NRC Response: The NRC disagrees with the comment.

The use of the word cracks is appropriate because the pellets are bonded at the top and bottom of the pellet-pellet interface. That interface must crack to open. Changing “cracks” to “the development of openings" implies that there is no bond at the top and bottom of the interface.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.32:

[Section 2.3.2, Page 2-7, Line 13 in the Draft Report for Comment]

A commenter stated that the example provided in the cited discussion is technically correct, but not applicable to the current situation since previous tests ignored the presence of the fuel (hence it would be like removing one of the two boards in the NRC example, not changing the arrangement or bonding between them). The commenter stated that this does have a very significant effect on the overall flexural rigidity, so the NRC is presenting a correct argument here, but in such a way that ignores the history of performing these calculations as if the fuel did not exist or provided zero structural support. The commenter further stated that, without this background information, this section does not seem to add value to the report. The commenter suggested that the section be clarified to explain the impetus for performing such calculations in the context of the history of performing cladding-only material property testing.

NRC Response: The staff agrees with the comment.

The staff has added an introductory paragraph to Section 2.3.2 to provide background as suggested in the comment.

Comment 4.3.33: [Section 2.3.3, Page 2-12, Table 2-1 in the Draft Report for Comment]

A commenter stated that Table 2-1 in the Draft Report for Comment should include the information referred to in the previous paragraph (i.e. comparison of flexural rigidity) and include the two comparisons with extra rows (or transpose the table and add columns). The commenter explained that this would make the paragraph and table more clear and useful. Similarly, the commenter also noted that the paragraph compares the EI2 cases with the numbers from the EI1 case, and recommended that, if this is applicable, the bottom row of the table should span the EI1 and EI2 region (not just one with a dash in the EI2 column insinuating no data).

NRC Response: The staff agrees with the comment.

NUREG-2224 has been revised per the comment. Further, the staff has revised Table 2-2 of the final report so that the entry in the "EI2" column for "Cladding-Only (Validated PNNL Models)" reads "26.933".

Comment 4.3.34: [Section 2.3.3, Page 2-13, Lines 24-26 in the Draft Report for Comment]

A commenter stated: "a safety factor of 1.4 is developed to account for hydride reorientation. This factor is further reduced to 1.25 to account for additional uncertainty. Then this is effectively reduced to a factor of 1.0 by suggesting to use cladding only properties and not allow any credit for the flexural rigidity provided by the cladding-fuel system which was clearly demonstrated by the CIRFT testing. I understand some reduction is prudent due to the limited data and the lack of data on all cladding types at this time, but to completely ignore any rigidity provided by the pellet is extremely conservative, unnecessary and not risk informed."

NRC Response: The NRC disagrees the comment.

The NRC defined the factors 1.4 and 1.25 for Zircaloy-4-clad HBU SNF rods. Until CIRFT results are available for HBU SNF with other cladding types, cladding-only properties should be used.

See also response to Comment 4.3.42.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.35: [Section 2.3.1, Page 2-14, Lines 12 to 13 in the Draft Report for Comment]

A commenter stated that the option to use cladding-only properties with a factor to account for the flexural rigidity should be the primary option. The commenter explained that "the first alternative to use cladding only properties (page 2-13, Line 32) is extremely conservative as it totally ignores the rigidity provided by the fuel. The 1.25 factor suggested for Zr-4 seems reasonable – a roughly 10% reduction in the value determined to allow for other uncertainties. Also, the factors that will be developed for Zirlo and M5 from the HBU sister rod testing should employ a similar method to determine a factor for these materials."

NRC Response: The NRC disagrees with the comment.

The staff considers both alternatives appropriate for evaluating the structural performance of the cladding.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.36: [Section 2.3.4.1, Page 2-18, Line 10 in the Draft Report for Comment]

A commenter noted that the assumed fuel density in the evaluation is low, and questioned whether this is because this has been reduced to account for dish and chamfer. The commenter questioned whether the assumed fuel density was appropriate.

NRC Response: The staff notes the comment.

The fuel density is calculated from data for a rod from a boiling water reactor 7 × 7 assembly, as found in Table C.1 of NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Storage System at a Nuclear Power Plant." (ADAMS Accession No. ML071340012).

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.37: [Section 2.3.4.1, Page 2-18, Line 20 in the Draft Report for Comment]

A commenter questioned whether BWR 7 × 7 rods is a typo, and whether it should instead be 15 × 15 PWR for the H.B. Robinson HBU SNF rods.

NRC Response: The NRC disagrees with the comment.

Section 2.3.4 of the final report explains that two fuel rods are considered in the cited evaluation: (1) a pressurized water reactor 15 × 15 and (2) an H. B. Robinson fuel rod (HBR fuel rod). The boiling water reactor (BWR) 7 × 7 rod from in Table C.1 of NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Storage System at a Nuclear Power Plant." (ADAMS Accession No. ML071340012) is used to determine fuel and cladding density to be able to characterize the HBR fuel rod as explained on Section 2.3.4 of the final report.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.38: [Section 2.4.1, Page 2-21, Lines 6 to 8 in the Draft Report for Comment]

A commenter noted that applying a conservative factor to account for a higher temperature is not necessary, if the results at room temperature are conservative.

NRC Response: The staff disagrees with the comment.

NUREG-2224 does not conclude that the results at room temperature are conservative. Therefore, the commenter’s assertion is unsubstantiated. The 90% reduction in equivalent strain is to account for all uncertainty, including the uncertainty associated with higher test temperatures.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.39: [Section 2.4.3, Page 2-24, Lines 11 to 12 in the Draft Report for Comment]

A commenter noted that this section concludes that seismic events aren't expected to compromise the fuel. However, the commenter questioned, in the linear damage rule for cumulative damage, whether any impact from seismic should be included in the cumulative damage, or if seismic damage is so small it is negligible compared to other damage. The commenter suggested a statement could be added to clarify that point.

NRC Response: The staff agrees with the comment.

The staff has provided an evaluation in Section 2.4.3 to explain that seismic impacts are negligible in the structural performance of a HBU SNF rod. The staff has further revised Section 2.4.3 to clarify that seismic strains do not need to be included in a cumulative damage evaluation conducted per Section 2.4.2 of NUREG-2224.

Comment 4.3.40: [Section 2.3.4.1, Page 2-17, Table 2-3 in the Draft Report for Comment]

A commenter recommended that, based on its content, the caption of Table 2-3 in the Draft Report for Comment should read "Relevant properties of 15x15 fuel assembly and a 15x15 fuel rod".

NRC Response: The staff agrees, in part, with the comment.

The staff has revised the caption of Table 2-4 of the final NUREG-2224 (previously Table 2-3 in the Draft Report for Comment) to "PWR 15 × 15 SNF Assembly Parameters," which better describes the contents of the table.

Comment 4.3.41: [Section 2.3.4.1, Page 2-19, Line 1 in the Draft Report for Comment]

A commenter recommended that, to provide consistency in units, the cited sentence should read "…2.9 to 4.6 g…".

NRC Response: The staff agrees with the comment. The staff has revised the cited sentence per the comment.

Comment 4.3.42: [Section 2.2, Page 2-4 in the Draft Report for Comment]

A commenter stated that, although this page contains useful data, it would be more useful to recreate the graphs that are referred to, such that the reader can better understand that referenced data and conclusions.

NRC Response: The NRC disagrees with the comment.

Reproducing all pertinent figures from NUREG/CR-7198, Revision 1 (ADAMS Accession No. ML17292B057), is not necessary for understanding NUREG-2224. The conclusions from those figures are adequately defined in Section 2.2, which is sufficient for an understanding of NUREG-2224. The reader can readily access NUREG/CR-7198, Revision 1, which is available from the NRC's website.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.3.42: [Section 2.3.3, Page 2-12, Lines 15 to 22 and Page 2-13, Lines 1 to 7 in the Draft Report for Comment]

One commenter stated that the methodology described in NUREG-2224 is not at all risk-informed. The commenter stated that, instead, it continues the dated practice of building conservatism upon conservatism to the point of the actual data having little to no relationship to the regulatory numbers used.

NRC Response: The staff disagrees with the comment.

To the extent practicable, NUREG-2224 was developed with a risk-informed mindset per the available test results at the time of publishing. For instance, as discussed in Section 2.3.4, the staff has made conclusions regarding the performance of modern cladding alloys during design-basis drop accidents per the static bending of Zircaloy-4-clad fuel rods. More specifically, the staff chose a risk-informed approach for the evaluation of these accidents when considering the limited set of data on static bend testing of Zircaloy-4-clad fuel rods. The staff also recognizes that a robust risk-informed regulatory framework requires continued engagement with industry and other stakeholders, and that process will require multiple iterations and improvements.

4.4 Chapter 3

Comment 4.4.1: [Section 3.1, Page 3-4; Lines 19 to 20 in the Draft Report for Comment]

A commenter stated that the cited statement is a good qualifier about the design basis and allows use of the Interim Staff Guidance 2, Revision 2 (ADAMS Accession No. ML16117A080).

NRC Response: The staff notes the comment.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.2: [Section 3.2.4.2.2, Page 3-13, Lines 25 to 30 in the Draft Report for Comment]

A commenter stated that the cited paragraph suggests that renewals are not possible until data from a HBU SNF demonstration program is available. The commenter stated that several licenses were renewed before the HBU SNF demonstration program started since allowed renewals rely on learning aging management programs based on the fact the data would be available when needed.

NRC Response: The staff agrees with the comment. The staff revised Section 3.2.4.2.2 to have consistent language with Sections 3.2.4.2.4 and 3.2.4.2.5.

Comment 4.4.3: [Section 3.2, Page 3-5, Line 5 and Section 4.2, Page 4-6, Lines 5-6 in the Draft Report for Comment]

A commenter stated that the two alternatives for the evaluation of drop accidents proposed in NUREG-2224 appear to more of a series (i.e., must go through cladding-only alternative (first approach) before the pellet-contribution alternative (second approach)). Another commenter suggested clarification that the second approach may be used in lieu of the first approach.

A commenter also stated that, based on the discussion in Section 1.5.5 on ring compression testing, since the cladding-only properties are overly conservative and not representative of actual fuel stress conditions during transportation and drop accidents, it is not clear why the first approach based on cladding-only properties is even needed.

NRC Response: The staff notes the comment.

Figure 3-1 (included in Section 3.2) is very clear that the second approach (pellet-contribution alternative) may be used by an applicant if the acceptance criteria of the first alternative (cladding-only alternative) is not met. However, to provide additional clarity, the NRC has revised the discussion in Section 3.2 to state that the second alternative would be necessary only if the structural evaluation using cladding-only mechanical properties is unsatisfactory. The NRC has also clarified in the discussion that an applicant may choose to implement the second alternative even if the first alternative were to yield satisfactory results. The staff notes that the test data necessary for the implementation of the second alternative is not currently available for all cladding alloys in use for HBU SNF. Hence, until that data is available, the first alternative may be implemented by an applicant.

Comment 4.4.4: [Section 3.2.4; Page 3-11; Line 39 in the Draft Report for Comment]

A commenter requested clarification on NRC's expectations on industry actions per the following statement in the Draft Report for Comment: "The staff considers it prudent to gather and review evidence that HBU fuel in dry storage beyond 20 years has maintained its analyzed

condition." The commenter was unclear on how this would be accomplished and at what schedule. The commenter also stated that the cited sentence redundantly states "gathered and reviewed".

NRC Response: The staff disagrees with the comment. Section 3.2.4.1 defines the NRC's expectations concerning the use of a confirmatory demonstration program for obtaining evidence that HBU SNF in dry storage beyond 20 years has maintained its analyzed condition. Subsections 3.2.4.1.1 and 3.2.4.1.2 state that an applicant may describe the activities used to obtain (gather) and evaluate (review) confirmatory data from the demonstration program under the aegis of a maintenance plan or an aging management program (AMP). The maintenance plan or AMP would be implemented after the initial 20 years of dry storage. These subsections refer the applicants to Appendices B and D of NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" (ADAMS Accession No. ML16179A148), when developing the description of activities to assess data from the confirmatory demonstration. As discussed in this response, the terms "gather" and "reviewed" are not synonymous.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.5: [Section 3.2.4.2, Page 3-12; Line 18 in the Draft Report for Comment]

A commenter requested clarification on whether supplemental safety analyses would be required to be performed until the completion of a HBU SNF demonstration program.

NRC Response: The staff notes the comment. Section 3.2.4.2 states that the supplemental safety analyses are an alternative approach to the confirmatory demonstration for HBU SNF discussed in Section 3.2.4.1. Section 3.2.4.1 states that an applicant may describe the activities to obtain and evaluate confirmatory data to be performed under the aegis of either a maintenance plan or aging management program for dry storage periods exceeding 20 years. Therefore, a license or Certificate of Compliance may be approved for dry storage periods beyond 20 years even when all activities of the referenced demonstration program are not completed, consistent with the guidance in Appendix D of NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel" (ADAMS Accession No. ML16179A148). The staff notes that this consistent with license and CoC renewals approved to date.

No revisions to NUREG-2224 were made in response to the comment.

Comment 4.4.6: [Section 3.3; Page 3-19; Line 25 in the Draft Report for Comment]

A commenter stated that, during the public meeting on the Draft Report for Comment (on September 6, 2018), there was clearly confusion with regards for canning of fuel. The

commenter stated that NUREG-2224 should clearly specify that canning would be performed during the transfer from a spent fuel pool to dry storage, based on the fuel condition known at that time. The commenter further stated that, it is unlikely that once loaded and placed into operation that these systems will need to be opened and fuel assemblies residing in the dry storage systems will need to be inspected and subsequently canned.

NRC Response: The staff disagrees with the comment.

The staff considers Section 3.3 of the final report sufficiently clear. Section 3.3 of the final report states that the staff will follow the guidance in the current Standard Review Plans for dry storage applications in its review of an application for a dry storage system with damaged HBU SNF contents. NUREG-2224 references these SRPs (i.e., NUREG-1567, “Standard Review Plan for Spent Fuel Storage Facilities,” (ADAMS Accession No. ML003686776) and NUREG-1536, Revision 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility,” (ADAMS Accession No. ML101040620)).

No revisions to NUREG-2224 were made in response to the comment.

Comment 4.4.7: [Section 3.1, Page 3-1, Lines 30 to 35 in the Draft Report for Comment],

Various commenters questioned the statements from Interim Staff Guidance 1, Revision 2 (ADAMS Accession No. ML071420268), which were reproduced in NUREG-2224.

NRC Response: The staff considers the comment to be out of the scope of NUREG-2224.

The discussion cited by the commenters pertains to the staff’s review guidance in Interim Staff Guidance (ISG)-1, Revision 2 (ADAMS Accession No. ML071420268). Because NUREG-2224 does not revise the staff’s position in ISG-1, Revision 2, the staff considers the comment to be out of the scope of NUREG-2224.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.8: [Section 3.2, Page 3-6, Lines 7 to 9 and Section 4.2, Page 4-6; Line 3 in the Draft Report for Comment]

A commenter suggested revising the cited statement from “is obtained” to “can be obtained”. The commenter stated that the latter language provides flexibility for using another method to determine the numerical factor for use in the approach defined in Figure 3-3. The commenter stated that it is understood that the other method for obtaining the numerical factor would need to be qualified.

NRC Response: The staff agrees with the comment. Section 3.2 of NUREG-2224 has been revised per the comment.

Comment 4.4.9: [Sections 3.2.1 and 3.2.2, Page 3-7, Lines 6, 8, and 23 in the Draft Report for Comment]

Various commenters questioned whether NUREG-2224 should cite ANSI N14.5 for defining leaktight requirements for dry storage systems (DSSs) and transportation packages. The commenters also questioned whether all currently-approved DSSs and transportation packages are designed to ANSI N14.5.

NRC Response: The NRC staff disagrees, in part, with the comment.

NUREG-2224 does not conclude that the design basis for all currently-approved leaktight DSSs and transportation packages is based on compliance with ANSI N14.5. However, the current Standard Review Plans (SRPs), for DSS designs, ISFSI and transportation packages define leaktight consistent with ANSI N14.5. NUREG-2224 is consistent with the definition in the current SRPs. Further, the staff notes that ANSI N14.5, which contains guidance for both leaktight and non-leaktight systems, is an NRC-accepted approach to verify adequate performance of confinement and containment systems, whether the design is considered leaktight or not. To provide additional clarification, the staff has added a definition of leaktight to the glossary of NUREG-2224.

For additional information on the NRC's staff review guidance, the commenters are referred to NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear fuel" (ADAMS Accession No. ML003696262).

Comment 4.4.10: [Page 3-7 in the Draft Report for Comment]

A commenter stated: "Section 3.2.1 discusses leaktight for storage. For storage, most welded canister-based systems are designed and tested to be leaktight, and NUREG-2224 suggests that dose calculations for release are not needed if the storage system remains leaktight. That is rather straightforward for a storage canister inside an overpack."

NRC Response: The NRC staff notes the comment.

The comment is consistent with NRC's staff review guidance – see NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776) and

NUREG-1536, Revision 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility” (ADAMS Accession No. ML101040620).

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.11: [Section 3.2.4.2.2, Page 3-13, Line 31 in the Draft Report for Comment]

A commenter stated that the cited paragraph is not very clear. The commenters stated that the intent of the cited paragraph should be that if the confinement boundary is maintained, then supplemental safety analyses for reconfiguration are not necessary.

NRC Response: The NRC agrees with this comment.

The ability to maintain confinement in a dry storage system is demonstrated by the thermal, structural, and material analyses, together with adequate aging management activities for the DSS subcomponents supporting confinement. If these provide reasonable assurance that the integrity of the confinement boundary is maintained even after hypothetical reconfiguration of the fuel under normal, off-normal, and accident-level conditions, then supplemental safety analysis for the confinement performance of the DSS design are not expected. Section 3.2.4.2.2 in NUREG-2224 has been revised, accordingly.

Comment 4.4.12: [Section 3.2.4.2.3, Page 3-14, Lines 5 to 6 in the Draft Report for Comment]

A commenter stated that release of fission gases into the canister will impact the heat transfer. The commenter considered that, given the relatively small volume in comparison, this would be negligible, especially since much of the rod fill gas is also helium. The commenter stated that, for lower pressure systems not relying on convection, it does not take much helium to get the full effect of heat removal. The commenter also asserted that, for convection-based systems, the amount of fission gas would be very small compared to the overpressure of helium in these systems.

NRC Response: The NRC staff disagrees with the comment.

The release of fission gas products may impact a horizontally-oriented canister-based DSS; therefore, the staff concludes it should be considered.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.13: [Section 3.2.4.2.3, Page 3-14, Lines 22 to 23 in the Draft Report for Comment]

A commenter stated that the cited statement sounds like the rod fill gas release will cause a rise in temperature. The commenter suggested to clarify whether the staff means that a release of fission gas inside the canister may influence heat transfer, which could change the peak component temperatures.

Response: The NRC staff agrees with the comment.

The staff has revised Section 3.2.4.2.3 to clarify that, for Scenario 1(a) in Category 1 (as described in Section 3.2.4.2), the fuel rods are assumed to breach in such a manner that the cladding remains in its nominal geometry (no fuel reconfiguration). However, the discussion recognizes that, depending on the canister orientation (horizontal or vertical), the release of fuel rod fill gas and fission product gases may influence heat transfer which can cause a change to maximum component temperatures.

Comment 4.4.14: [Section 3.2.4.2.4, Page 3-15, Line 39 in the Draft Report for Comment]

A commenter stated that the cited statement appears to be incomplete. The commenter proposed to revise the cited statement with the following text: "…shows that reactivity increases for longer decay times … and the application would need to use …".

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised in response to the comment.

Comment 4.4.15: [Section 3.2.4.2.5, Page 3-17, Line 19 in the Draft Report for Comment]

A commenter suggest adding the following text to the end of the cited statement: "…using insights from NUREG/CR-7203 for reconfigured geometry".

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised in response to the comment.

Comment 4.4.16: [Section 3.2.4.2.4, Page 3-16, Line 6 and Section 3.2.4.2.5, Page 3-17; Line 41 in the Draft Report for Comment]

Various commenters noted that the Draft Report for Comment cited a fuel failure rate of 3 percent for normal conditions of storage, instead of 1 percent.

NRC Response: The staff agrees with the comment.

The fuel failure rate of 3 percent cited in the Draft Report for Comment was incorrect. NUREG-2224 has been revised in response to the comments.

Comment 4.4.17: [Section 3.2.4.2.5, Page 3-18; Line 9 in the Draft Report for Comment]

A commenter suggested revising the cited statement for additional clarity with the following text: "dose far away from the cask and therefore …".

NRC Response: The staff agrees with the comment. NUREG-2224 has been revised in response to the comment.

Comment 4.4.18: [Section 3.2.4.2.5 in the Draft Report for Comment]

A commenter stated that scenario 2 is no cladding failure, so it does not seem to belong in the cited discussion.

NRC Response: The NRC disagrees with the comment.

For scenario 2, Oak Ridge National Laboratory analyzed the fuel as homogenized and collapsed it against the side of the basket plates. For certain scenarios, this may be an appropriate model for damaged fuel for facilities with horizontal storage or for a tip-over event. This is further clarified in Section 3.2.4.2.5.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.19: [Section 3.2.2, Page 3-8, Line 26 in the Draft Report for Comment]

A commenter requested clarification on whether the bounding 15 percent release fraction for fission gases is specifically for HBU SNF at the end of reactor operations. The commenter suggested clarification that the cited release fraction value applies to transportation applications.

NRC Response: The NRC staff notes with the comment.

Section 3.2.2 identifies the bounding release fraction for all dry storage applications of HBU SNF, irrespective of when the fuel is loaded into a dry storage system.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.20: [Section 3.2.2, Page 3-9, Lines 42 to 43 in the Draft Report for Comment]

A commenter stated that the release fraction of 3 × 10^{-5} was developed for transportation accident scenarios. The commenter stated the opinion that this value may be appropriate for accident conditions of storage (e.g., tipover accident), but stated that a smaller value is justified for fire and off-normal conditions, and even less for normal conditions. The commenter stated that it does not make sense to use the same release fraction developed from a transportation drop accident as for normal conditions of storage.

NRC Response: The NRC disagrees with the comment.

As stated in Section 3.2.2, if the bounding release fractions provided in NUREG-2224 are not used in an application, other release fractions may be used if the applicant properly justifies the basis for their usage. The justification of the proposed release fractions of the source terms should consider an adequate description of burnup for the test specimen, number of tests, collection method for quantification of respirable release fractions, test specimen pressure at the time of fracture, and source collection system. Since the commenter did not provide a technical basis to support a different value of a bounding release fraction, the values in Section 3.2.2 are adequate and provide sufficient flexibility to applicants.

No revisions to NUREG-2224 were made in response to the comment.

Comment 4.4.21: [Section 3.2.2, Page 3-10; Line 32 and Section 4.2.2, Page 4-11, Line 17 in the Draft Report for Comment]

Some commenters stated that international work on release fractions is publicly available.

NRC Response: The NRC staff notes the comment.

The staff used the information cited by the commenters in its assessments and appropriate references are provided in NUREG-2224.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.22: [Section 3.2.2, Page 3-8, Line 2 in the Draft Report for Comment]

A commenter stated that the language above both Tables 3-1 and Table 4-1 of NUREG-2224 appears to define additional expectations for the justification of alternative release fractions for non-leaktight designs. The commenter requested that the language be made more consistent with that in NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620).

NRC Response: The staff agrees with the comment.

To improve consistency with NUREG-1536, Revision 1 (ADAMS Accession No. ML101040620) while still considering how the characteristics of HBU SNF differ from low burnup fuel, the language above Table 3-1 and Table 4-1 of NUREG-2224 was revised to read: "If the release fractions are not used, other release fractions may be used in the analysis provided the applicant properly justifies the basis for their usage. Justification of the proposed release fractions of the source terms should consider an adequate description of burnup for the test segment, number of tests, collection method for quantification of respirable release fractions, test segment pressure at the time of fracture, and source collection system."

Comment 4.4.23: [Section 3.2.2, Page 3-7, Line 26 in the Draft Report for Comment]

A commenter stated that, with regards to non-leaktight confinement, the expectation in NUREG-2224 is to use fixed fuel failure rates (and bounding release fractions in Table 3-1) in the event an applicant cannot provide and justify other values (normal, off-normal, accident). The commenter questioned whether the fixed/bounding values in Table 3-1 take into account the better cladding properties described in the prior sections of NUREG-2224.

NRC Response: The staff notes the comment.

The failure rates for HBU SNF (i.e., the fraction of fuel rods that develop cladding breaches) during normal, off-normal, and accident conditions of storage, as defined in Table 3-1 of NUREG-2224, are consistent with those defined in the current Standard Review Plans (SRPs) for the evaluation of dry storage systems with a non-leaktight confinement for low burnup fuel (see NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities," (ADAMS Accession No. ML003686776) and NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," (ADAMS Accession No. ML101040620)). Therefore, the failure rates are not informed by the test results discussed in NUREG/CR-7198, Revision 1 (ADAMS Accession No. ML17292B057). The bounding release fractions consider the aforementioned failure rates, and the different characteristics of HBU SNF relative to low burnup fuel, as discussed in Section 3.2.2.

Consistent with the guidance in those SRPs, an applicant may propose alternate approaches (e.g., alternate fuel failure rates) for the evaluation of dry storage systems with a non-leaktight confinement for HBU SNF.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.4.24: [Chapters 3 and 4 in the Draft Report for Comment]

A commenter stated that the examples in Chapter 3 and 4 make extensive reference to the approaches developed in NUREG/CR-7203 (ADAMS Accession No. ML15266A413). The

commenter stated that NUREG/CR-7203 clearly and repeatedly states that the configurations considered are studies beyond the design basis, many of those excessively conservative, if not physically impossible, and that the purpose of that report is to show the sensitivity to those assumed conditions. However, the commenter stated that the way they are referenced in NUREG-2224 does not appear to fully recognize this.

The commenter recognized that NUREG-2224 states in numerous places that other approaches may be acceptable, but concluded that statements in NUREG-2224 (such as "In an approach acceptable to the staff …"), for all practical purposes, essentially elevate those approaches or conditions to design basis conditions. The commenter recommended adding an introductory section to those chapters that clearly clarifies the nature of the studies performed in NUREG/CR-7203 and its relevance for NUREG-2224. The commenter considered the clarification valuable for both applicants and NRC reviewers.

NRC Response: The NRC disagrees, in part, with the comment.

The NRC recognizes that some of the configurations evaluated in NUREG/CR-7203 (ADAMS Accession No. ML15266A413) and considered in NUREG-2224 are detailed, conservative, and intended for research purposes. A wide range of fuel reconfiguration scenarios are analyzed. However, NUREG/CR-7203 provides adequate reference approaches acceptable to the staff for reviewing the safety consequences of HBU SNF reconfiguration in transportation packages and dry storage systems. Further, the NRC has recognized in NUREG-2224 that the results in NUREG/CR-7203 should not be considered generically applicable. Therefore, NUREG-2224 is sufficiently clear on the nature of the studies in NUREG/CR-7203. In addition, the NRC disagrees with the assertion made by the commenter that statements in NUREG-2224 (e.g., "... in an approach acceptable to staff ...") imply that these configurations are design-basis conditions. Both NUREG/CR-7203 and NUREG-2224 are technical reports and are not regulatory requirements.

No revisions were made to NUREG-2224 in response to the comment.

4.5 Chapter 4

Comment 4.5.1: [Section 4.2.4.2.2, Page 4-14, Line 2 in the Draft Report for Comment]

A commenter requested clarification on the data needed from a HBU SNF demonstration program, which would be needed before shipment of fuel dry stored beyond 20 years. The commenter stated that initial data from the loading of a HBU SNF demonstration program will be available, but the data after opening the cask will not be available until about 2027. Another commenter stated that the HBU SNF demonstration program will only last 10 years.

NRC Response: The staff disagrees with the comment.

The discussion in Section 4.2.4.2.2 properly references the review guidance in NUREG-1927, Revision 1, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel” (ADAMS Accession No. ML16179A148), which discusses the staff’s expectations for data to be obtained under a surrogate surveillance and monitoring program. No revisions were made to NUREG-2224 in response to the comment.

Comment 4.5.2: [Section 4.2.4.2.2, Page 4-14, Line 4 in the Draft Report for Comment]

A commenter requested the staff revise the word “confinement” to “containment” in the cited statement.

NRC Response: The staff disagrees with the comment.

The discussion pertains to aging management activities for the dry storage system subcomponents supporting the primary confinement boundary during prior dry storage.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.5.3: [Section 4.2.4.2, Page 4-12, Line 44 in the Draft Report for Comment]

A commenter stated that the statement “hypothetical reconfiguration…into justified geometric forms” in the cited statement is too open-ended.

NRC Response: The staff disagrees with the comment.

The staff directs the commenter to Section 4.2.4.2 and its subsections for discussions on supplemental analyses based on reference reconfiguration scenarios per NUREG/CR-7203 (ADAMS Accession No. ML15266A413). These supplemental analyses apply if an applicant chooses not to rely on a surrogate demonstration program to confirm the results from separate-effects testing that have provided the technical bases for dry storage of HBU SNF beyond 20 years. The staff did not attempt to generically define the geometric rearrangement of the fuel to be analyzed, since these analyses may be dependent on both the safety function being evaluated and the specific design of the transportation package.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.5.4: [Section 4.2.1, Page 4-7 in the Draft Report for Comment]

A commenter stated that Section 4.2.1 discusses leaktight containment for transportation. The commenter further stated that, for transportation, you typically have a transport overpack with a

bolted lid that will not be leaktight, but inside the non-leaktight transport overpack, you have a leaktight canister. The commenter suggested that Section 4.2.1 be expanded to include a discussion of a leaktight canister inside a non-leaktight transport overpack. The commenter stated that, similar to the storage condition, if the canister inside the transport overpack remains leaktight, one should not need to perform dose calculations for releases, even though the transport overpack may not be leaktight. The commenter stated that, if the canister remained leaktight, there would be no material available for release.

NRC Response: The NRC staff disagrees, in part, with the comment as it relates to the expansion of Section 4.2.1. No revisions will be made to Section 4.2.1. This is because the words "containment boundary" cover all cases, including the case that the canister is leaktight and the overpack is not leaktight.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.5.5: [Section 4.2.4.2, Page 4-13, Line 6 in the Draft Report for Comment]

A commenter stated that it is not apparent how the postulated fuel reconfigure scenarios are applicable to HBU SNF.

NRC Response: The NRC staff notes the comment.

The scenarios the staff evaluated in NUREG-2224 serve as reference points for the effects of HBU SNF reconfiguration on the shielding, criticality, containment/confinement, and thermal analyses. As discussed in Section 4.2.4.2, absent data affirmatively confirming that the HBU SNF has remained in its analyzed configuration, an applicant could conduct supplemental analyses similar to those evaluated in NUREG/CR-7203 (ADAMS Accession No. ML15266A413). The supplemental analyses would serve to demonstrate that, even if reconfiguration occurred, the transportation package would still meet the pertinent 10 CFR Part 71 regulatory requirements in the areas of shielding, criticality, containment/confinement, and thermal performance.

No revisions were made to NUREG-2224 in response to the comment.

Comment 4.5.6: [Section 4.2.2, Page 4-8, Line 27 in the Draft Report for Comment]

A commenter suggested clarifying the cited statement with the following text: "If the release fractions in Table 4-1 are not used …". The commenter stated that a justification would not be needed if using the values in Table 4-1.

NRC Response: The staff agrees with the comment.

NUREG-2224 was revised to clarify that, if these release fractions are not used, other release fractions may be used in the analysis provided the applicant properly justifies the basis for their usage.

Comment 4.5.7: [Section 3.2.2, Page 3-8, Line 5, and Section 4.2.2, Page 4-9, Line 27 in the Draft Report for Comment]

A commenter stated that the cited statements are the only places where the term "respirable fraction" is mentioned. The commenter stated that NUREG-1536, Revision 1 (ADAMS Accession No. ML101040620), does not discuss respirable fraction, which implies that particle size and respirable fraction may not be accounted for in the release fractions provided. The commenter requested that NUREG-2224 be clarified to define whether it is acceptable for an applicant to apply a respirable fraction (with appropriate justification) to the release fractions in Table 5-2 of NUREG-1536, Revision 1, and/or the release fractions in Tables 3-1 and Table 4-1 of NUREG-2224 for analyses of inhalation dose. The commenter notes that, in NUREG/CR-6672 (ADAMS Accession No. ML003698324) there are much lower values presented for a parameter identified as a "rod to cask release fraction for respirable fuel fines".

NRC Response: The staff agrees with the comment. The cited statements in NUREG-2224 were revised to remove the qualifier "respirable".

Comment 4.5.8: [Section 4.2.2, Page 4-8, Line 24 in the Draft Report for Comment]

A commenter stated that the release fractions for non-leaktight containments in Tables 3-1 and 4-1 may be too conservative relative to the noted cladding properties covered in prior sections of NUREG-2224. Another commenter expressed that these release fractions were not sufficiently conservative.

NRC Response: The NRC disagree with the comment.

The assessment on release fractions discussed in NUREG-2224 was based on available data from HBU SNF experimental testing. NUREG-2224 provides adequate references in support of these release fractions. Further, the failure rates for HBU SNF (i.e., the fraction of fuel rods that develop cladding breaches) during normal, off-normal, and accident conditions of storage, as defined in Tables 3-1 and 4-1 of NUREG-2224, are equivalent to those defined in the staff review guidance in the current Standard Review Plans (SRPs) for the evaluation of non-leaktight dry storage systems and transportation packaged for low burnup fuel (see NUREG-1567, "Standard Review Plan for Spent Fuel Storage Facilities" (ADAMS Accession No. ML003686776), NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" (ADAMS Accession No. ML101040620), and NUREG-1617, "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" (ADAMS Accession No. ML003696262)). Consistent with the guidance in those SRPs, an applicant may

propose alternate approaches (e.g., alternate fuel failure rates) for the evaluation of non-leaktight dry storage systems and transportation packages for HBU SNF.

No revisions were made to NUREG-2224 as a result of the comment.

4.6 Chapter 5

Comment 4.6.1: [Chapter 5, Page 5-1 in the Draft Report for Comment]

A commenter stated that the conclusions section is somewhat vague in nature and could use some work to more clearly state the conclusions. The commenter further stated that it is possible to catch the conclusions after reading the entire report and conclusions sections carefully, but absent that, the conclusions section is too vague and doesn't provide a clear path forward on how to use the information in this report.

Response: The staff disagrees with the comment.

The conclusions in Chapter 5 of NUREG-2224 clearly define the purpose of the technical report and the conclusions on the performance of HBU SNF during postulated design-basis drop accidents and vibration normally incident to transport, per the engineering assessment in Chapter 2. The commenter is referred to Chapters 3 and 4 of NUREG-2224 for additional details on the staff-accepted licensing and certification approaches for dry storage systems and transportation packages with HBU SNF contents.

No revisions were made to NUREG-2224 in response of the comment.

Comment 4.6.2: [Chapter 5, Page 5-1, Line 36 in the Draft Report for Comment]

A commenter requested the NRC include tipover in addition to drop accident scenarios in the cited statement.

Response: The staff agrees with the comment. NUREG-2224 has been revised in response to comment.

Comment 4.6.3: [Chapter 5, Page 5-1, Lines 40 to 42 in the Draft Report for Comment]

A commenter stated that the conclusion is okay based on what we know today and expect from most of a HBU SNF test program under DOE-sponsorship. The commenter further stated that CIRFT tests on samples with pellet-pellet gaps or under grid spacers have not been completed. The commenter also stated that knowledge about Optimized ZIRLO, which has a complicated and non-uniform microstructure, is limited.

NRC Response: The staff notes the comment.

The conclusions regarding the performance of HBU SNF during design-basis drop accidents and vibration normally incident to transport are adequately supported by the engineering assessment in Chapter 2 of NUREG-2224. The NRC recognizes that additional testing is being conducted under the sponsorship of the DOE. The NRC continues to actively engage with the DOE to ensure that our regulatory framework is informed by new results on the performance of HBU SNF in dry storage and transportation.

See also response to Comment 3.4.37 and Comment 4.3.14.

No revisions were made to NUREG-2224 in response to the comment.