RESPONSES TO PUBLIC COMMENTS ON DRAFT NUREG–2214, "MANAGING AGING PROCESSES IN STORAGE (MAPS) REPORT"

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Washington, DC 20555-0001

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CHAPTER 1: INTRODUCTION

On October 24, 2017, the U.S. Nuclear Regulatory Commission (NRC) staff published for comment Draft NUREG–2214, "Managing Aging Processes in Storage (MAPS) Report" (82 FR 49233). The staff developed NUREG–2214 to improve the effectiveness and efficiency of safety reviews of renewal applications for specific licenses of independent spent fuel storage installations (ISFSIs) and certificates of compliance (CoCs) for dry storage systems (DSSs). Through the development of the draft technical guidance document, the staff considered stakeholder input received at public meetings on renewal topics, including a public meeting on December 7, 2017 specifically aimed at soliciting stakeholder input on the proposed Draft NUREG–2214. The meeting summary is available on the Agencywide Documents Access and Management System (ADAMS) at Accession No. ML17356A028.

The staff received 7 comment letters, consisting of 322 individual comments. The commenters and their affiliations are shown in the table below, along with the ADAMS Accession Number of the individual comment letters. The staff considered these public comments when finalizing the guidance in NUREG–2214.

	Name	Affiliation	ADAMS Accession No.	Submittal Date
1	Donna Gilmore	SanOnofreSafety.org	ML17363A207	12/26/2017
2	Donna Gilmore	SanOnofreSafety.org	ML17363A209	12/26/2017
3	George Carver	NAC International	ML18002A350	12/20/2017
4	Benjamin Holtzman	Nuclear Energy Institute	ML18002A351	12/21/2017
5	Raymond Lutz	Citizens Oversight, Inc.	ML18046A071	01/02/2018
6	Marvin Lewis	Public	ML18046A070	12/07/2017
7	Kimberly Manzione	Holtec International	ML18046A067	12/21/2017

The staff responses to the comments are presented in this document. In reading the comments and responses, note the following:

- The comments have been grouped by topical area.
- The comment numbers are added by NRC staff for convenience and to simplify cross-referencing.
- The content of the comments is taken directly as provided by the commenters. In a few cases, minor editorial changes have been made by NRC staff only for readability.
- The formatting in original comment letters has in some cases been altered to fit this present document. NRC staff has tried to retain formatting that is pertinent to the comments.

CHAPTER 2: RESPONSES TO PUBLIC COMMENTS ON GENERAL ISSUES

The comments on General Issues either: (1) reference the purpose and scope of NUREG–2214 or (2) are not specific or related to the guidance.

2.1 Comments from Donna Gilmore/SanOnofreSafety.org

Comment 2.1.1

Comment: *Overall, I find a lack of evidence for conclusions in NUREG–2214. NUREG–2214 makes statements the NRC has lack of evidence that there are various problems and within certain timeframes. However, the fact the NRC is approving use of inferior thin-wall canisters that are welded shut and that cannot and have not been inspected (inside or out) for cracks or other degradation, and cannot be monitored, repaired or maintained to PREVENT leaks is the root of the problem. Until you address this root problem, you are putting our communities at risk for major radioactive releases, with no adequate plans in place to prevent or stop these releases. Cherry picking data and making assumptions due to inability to inspect canisters and their contents should be unacceptable to anyone who that cares about the future of this country. The problem is now. These canisters are reaching the age of no return. Please include information in this presentation as part of my comments. These issues have not been adequately addressed, but are critical for managing aging processes in storage.*

Coast to Coast Spent Fuel Dry Storage Problems and Recommendations, Erica Gray, NRC REG CON 2015, November 18, 2015

https://www.nrc.gov/public-involve/conference-symposia/dsfm/2015/dsfm-2015-erica-gray.pdf

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The NRC requires licensees and CoC holders to evaluate potential aging effects for DSSs, and where necessary, include aging management programs (AMPs) to assure that the important to safety structures, systems, and components continue to maintain their intended functions. If a specific inspection or examination methodology is under development at the time of a renewal review, the NRC may condition a license or CoC renewal to require development of the methodology within a certain timeframe. If the licensee or CoC holder identifies that proposed inspections cannot be performed in accordance with conditions of the license or CoC renewal, then the licensee or CoC holder is out of compliance with the renewed license and would be required to propose an alternative approach that demonstrate that the safety functions of the structure, system, or component will be maintained for the period of extended operation.

Nondestructive examination (NDE) methods that can be used to inspect canisters already exist and have been used by the nuclear industry for decades to examine structures, systems, and components that are constructed from welded stainless steels and welded austenitic nickel-based alloys. Methods to apply existing NDE techniques to welded austenitic stainless steel canisters have been developed and are being tested by both the Electrical Power Research Institute (EPRI) and DSS manufacturers. The NRC and the nuclear industry, including DSS manufacturers and users of these systems, are active participants in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Task Group to

develop a code case for the examination, acceptance criteria, and corrosion/degradation assessment methodologies for dry cask storage systems.

Comment 2.1.2

Comment: *Regarding your criticality response, you agree that your assumption of no criticality in dry storage assumes no water enters the canister, as you noted below with the statement "the criticality safety control during storage does rely on the exclusion of water from the canister."*

I highly recommend you make this clarification in your public outreach and on your website as one of your safety assumptions regarding risk of criticality in dry storage. I'm sure the majority of the public and decision makers do not know this.

NRC Response: This comment is not specific to NUREG–2214; therefore, no changes were made in response to the comment.

The staff considers the NRC standard review plan (SRP) for dry storage of spent fuel to be the most appropriate means of communicating the recommended practice for evaluating criticality with respect to canister flooding.

In the recent issuance of the draft SRP for dry storage (NRC, 2017), the staff describes the conditions under which canister flooding must be considered in a criticality analysis. The draft SRP states:

> The DSS [dry storage system] or DSF [dry storage facility] SSCs must be designed so that at least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, must occur before an accidental criticality is possible ("Double Contingency," as stated in American National Standards Institute (ANSI)/American Nuclear Society (ANS) 8.1, "Nuclear Criticality Safety in Operations with Fissionable Materials Outside Reactors"; see 10 CFR 72.124(a)…. For DSS or DSF storage container designs, these criteria are typically met by demonstrating a low likelihood of storage container failure and a low likelihood of flooding of the storage container to sufficient depth to cause criticality (i.e., to the height of the active fuel) in the container's dry storage configuration.

The staff notes that the comment is associated with a separate discussion between the commenter and NRC staff about the storage system at a particular site. That discussion is documented in ADAMS Accession No. ML17356A068.

<u>Reference</u>

NRC. NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities, Draft Report for Comment," ADAMS Accession No. ML17310A693. Washington, DC: U.S. Nuclear Regulatory Commission. November 2017.

Comment 2.1.3

Comment: *Regarding risks of breach of canister (through-wall cracks), your statements that this cannot happen in 20 years and would possibly take decades is based on false assumptions. I have read your responses to my comments you referenced in ISG-2, Revision 2 Response to Stakeholder Comments (ADAMS Accession No. ML16117A082). Link:* https://www.nrc.gov/docs/ML1611/ML16117A082.pdf

NRC Response: The staff disagrees with this comment; therefore, no changes were made in response to the comment.

The staff has been actively involved in addressing the potential for stress corrosion cracking (SCC) of welded stainless steel canisters for more than 10 years. The conditions under which SCC may occur and the rates at which it may progress have been evaluated by EPRI (Chu, 2014). The staff also evaluated conditions for cracking of dry storage canisters, as discussed in the April 21, 2015, public meeting with the Nuclear Energy Institute on the Chloride Induced Stress Corrosion Cracking Regulatory Issue Resolution Protocol. The meeting summary is available at ADAMS Accession No. ML15146A090, and the presentation materials are available at ADAMS Accession No. ML15146A115.

Although there are differences between the EPRI and the NRC analyses, such as the parameters used to determine crack growth rates, both the EPRI and NRC analyses reached similar conclusions on the range of possible crack growth rates and the important influence of site-specific conditions. The staff finds there is no basis to support the commenter's assertion that canisters may have through-wall cracks in as little as 20 years.

The staff will continue to collect and review relevant operational experience on atmospheric exposure and SCC of austenitic stainless steels. If, at some time in the future, the NRC were to identify a concern with the safe storage of spent fuel, the NRC would evaluate the issue and take actions necessary to protect public health and safety.

Reference

Chu, S. "Flaw Growth and Flaw Tolerance Assessment for Dry Cask Storage Canisters." EPRI-3002002785. Palo Alto, California: Electric Power Research Institute. August 2014.

Comment 2.1.4

Comment: *Regarding the 2-year old Diablo Canyon canister that has all the conditions for cracking, you admit there are salts and a temperature low enough for moisture to stay on the canister and dissolve salts. However, you claim there would be insufficient humidity at the Pacific Coast locations for moisture to dissolve (deliquesce) the salts on the canister, which we both agree can trigger the initiation of cracking. You need to reevaluate your source data regarding California Pacific Coast weather. As stated in this California Climate Zones document, frequently daily fog is common along the San Diego and San Luis Obispo coastline. https://www.pge.com/includes/docs/pdfs/about/edusafety/training/pec/toolbox/arch/climate/california_climate_zones_01-16.pdf*

I raised this issue with Darrell Dunn in one of the NRC meetings, asking him where the location was of his NOAA weather data. He said he used Vandenberg Air Force Base. That is not relevant to San Onofre. I sent him the above link, but did not receive a response. I also sent him some photos of evening and morning fog along the coast. I live 5 miles from San Onofre and have a view of the coastline from my backyard. Maybe pictures would help you understand this better.

Photo of San Onofre in morning fog.

https://sanonofresafety.files.wordpress.com/2017/12/la-174296-me-0919-surf-19-jpg-20130506fog.jpg

Photo taken about 5 miles northeast of San Onofre. Frequent fog is common. It's so thick here you cannot see the ocean.

https://sanonofresafety.files.wordpress.com/2017/12/20171125_164412coastalfognearsanonofre.jpg?w=640

Photo taken about 5 miles northeast of San Onofre at a similar location on a clear evening. You can see Catalina Island in the distance and Dana Point Harbor to the right.

https://sanonofresafety.files.wordpress.com/2017/12/20171123_164900_sunsetnearsanonofre.jpg?w=640

Your August 5, 2014 meeting summary admits once cracks starts they can go through the wall in 16 years. I participated in that meeting. In that meeting you also said canisters would not have a low enough temperature for moisture to stay on the canister for 30 years. However, the 2-year old Diablo Canyon canister information proves that is not true.

NRC Response: The staff disagrees with this comment; therefore, no changes were made in response to the comment.

The staff disagrees with the assertion that the NRC has not considered the most relevant climate data in its evaluation of chloride induced stress corrosion cracking (CISCC). The staff has previously responded to assessments made by stakeholders regarding the Diablo Canyon canisters in the response to public comments for 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. ML16125A534) and Interim Staff Guidance (ISG)-2, Revision 2, "Fuel Retrievability in Spent Fuel Storage Applications" (ADAMS Accession No. ML16117A082).

The information presented in the California climate zones summary is not hourly weather data that is necessary to evaluate the potential for CISCC. The staff compared the California climate zones summary to the National Oceanic and Atmospheric Administration (NOAA) data used in the staff presentation on the subject at the April 21, 2015, public meeting with the Nuclear Energy Institute on the Chloride Induced Stress Corrosion Cracking Regulatory Issue Resolution Protocol (meeting summary: ADAMS Accession No. ML15146A090; presentation materials: ADAMS Accession No. ML15146A115). The maximum dew points were calculated using the maximum daytime temperature and the corresponding afternoon (4:00PM) relative

humidity for all 16 California climate zones. The maximum dew points ranged from 62 to 68°F [16.7 to 20.8°C]. The 2014 data from the Vandenberg NOAA station was examined and the maximum dew points for the months of July, August, and September ranged from 61 to 66°F [16.1 to 18.9°C]. The differences are minor. Nevertheless, the NRC will continue to evaluate additional information to update the CISCC assessment as data becomes available.

Comment 2.1.5

Comment: *The EPRI report you referenced cherry picked data to reach their conclusion of many decades before through-wall cracks. Here is a link to the evidence of that. Please let me know if you disagree with these points and why. Otherwise, please quit using that EPRI report as justification.* https://sanonofresafety.files.wordpress.com/2013/06/epri-critiqueandkoebergplant2015-05-17.pdf

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The staff notes that the comment refers to operational experience from a refueling water storage tank at the Koeberg Nuclear Power Station in South Africa. The Koeberg tank is not exposed to the same environmental conditions as those expected for DSSs. When the operational conditions for DSSs are considered, the conditions under which SCC can initiate and propagate are limited. The NRC presented an analysis of crack growth at the April 21, 2015, public meeting with the Nuclear Energy Institute on the Chloride Induced Stress Corrosion Cracking Regulatory Issue Resolution Protocol (meeting summary: ADAMS Accession No. ML15146A090; presentation materials: ADAMS Accession No. ML15146A115).

When the NRC first looked at operational experience of other plant components, the information available was both inconsistent and incomplete. Because it is important to understand operational experience, a more detailed investigation of the operational experiences for nuclear plant components has been conducted. A summary of the operational experience with CISCC and an assessment of the operational experience with the EPRI model for CISCC propagation are included in EPRI Report, "Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless Steel Canisters" (Chu, 2017). The operational experience considered includes CISCC events at Koeberg (South Africa), St. Lucie (Florida), San Onofre (California) and multiple initiation test studies carried out in Japan and in the U.S.

Reference

Chu, S. "Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless Steel Canisters." EPRI-3002008193. Palo Alto, California: Electric Power Research Institute. March 2017.

Comment 2.1.6

Comment: *The Sandia Lab report you referenced found salt particles on the Diablo canister. Since there was limited access (only through the outlet air vent in the cask) it was only a partial search of dust particles, yet it was enough to find salts. The report admits they may have found more salt if they had access through the inlet air vent, since this is where salt would most likely enter the cask.*

SANDIA REPORT SAND2014-16383, "Analysis of Dust Samples Collected from Spent Nuclear Fuel Interim Storage Containers at Hope Creek, Delaware, and Diablo Canyon, California," Charles R. Bryan and David G. Enos, July 2014 http://prod.sandia.gov/techlib/access-control.cgi/2014/1416383.pdf

NRC Response: This comment is not specific to NUREG–2214; therefore, no changes were made in response to the comment.

See the response to Comment 2.1.4.

Comment 2.1.7

Comment: *Regarding Aging Management, you do not consider aging management in the initial 20-year license. You relicensed Calvert Cliffs dry storage installation where similar thin-wall canisters are used. However, you are not requiring them to inspect for cracks. Instead only requiring a visual inspection, which, as you know, cannot adequately find cracks or measure depth of cracks. And you're only requiring they perform the visual inspection on one canister at the facility.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

Although the regulations do not explicitly require aging management programs (AMPs) in the initial storage term, the staff's reviews of DSS applications include an evaluation of materials performance over the length of the initial license, considering corrosion and other degradation processes. In addition, the staff evaluates if periodic inspections are needed to monitor material conditions or performance (NRC, 2010).

Regarding the capability to identify cracking with visual inspections, the example AMP for chloride stress corrosion cracking of canisters in NUREG-2214 Table 6-2 recognizes the limitations of visual inspections to directly identify cracking. However, pitting and crevice corrosion can be detected by visual examinations, and these corrosion phenomena are considered to be precursors to stress corrosion cracking. As a result, the example AMP recommends visual inspections to identify corrosion and, if corrosion is discovered, volumetric examinations to characterize the extent and severity of corrosion and stress corrosion cracking. As the commenter stated, the example AMP recommends that at least one canister at each site be inspected. By choosing the canister considered to be most susceptible to corrosion and

cracking, the inspection is expected to be a leading indicator for canister degradation at the site. If indications of degradation are found, the AMP recommends the inspection of additional canisters to determine the extent of condition.

<u>Reference</u>

NRC. NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," ADAMS Accession No. ML101040620, Washington, DC: U.S. Nuclear Regulatory Commission. July 2010.

<u>Comment 2.1.8</u>

Comment: *The word "inspection" is being used loosely by the nuclear industry, including the NRC. You know existing facilities have no idea how many of their canisters have cracks or how deep the cracks may be. Also, even if they eventually figure out how to find cracks and depth of cracks, you have no proven method to repair these cracks once the canisters are loaded with nuclear fuel waste.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the responses to Comments 2.1.7 and 2.1.9 for the staff's discussion of inspections and corrective actions.

<u>Comment 2.1.9</u>

Comment: *To make matters worse, the NRC allows decommissioning nuclear facilities to destroy their spent fuel pools, yet neither the NRC or Edison have explained how you could possibly repair canisters (with or without a pool) or deal with leaking canisters. Both the NRC and Edison have avoided giving a straight answer to what they will do with through wall cracks. Show me the approved plan in writing, not some vague statement it's up to the license to figure this out as needed. How will these canisters ever be transported with even partial cracks?*

I have read the other documents you referenced. They do not provide evidence for your claims. We can go into that further, but I don't need to waste more of my time if the above facts are going to be ignored by the NRC.

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

While NUREG-2214 recommends an approach to inspect canisters, it does not pre-approve or prescribe specific correction actions. The staff recommends that corrective actions be determined on a case-specific basis, given the wide range of potential inspection findings,

storage system designs, and available repair and recovery options. The NRC will evaluate whether the licensees' corrective actions are effective and adequate to maintain the intended functions of the important-to-safety structures, systems, and components, and whether the licensee remains compliant with the requirements in 10 CFR Part 72.

__**Comment 2.1.10**__

Comment: *Do you realize the consequences if San Onofre canisters start leaking and potentially explode from gases building up from the high burnup fuel or from a criticality of water entering through the cracks?*

- *Do you realize we're talking about major Southern California evacuations, many likely permanent?*

- *Do you realize the economic impact to California, the nation and even the world?*

Now is not the time to play bureaucratic nuclear roulette with our country. Please step up and do the right thing.

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

Regarding the generic issue of potential cracking, the NRC requires aging management programs and recommends that those programs include periodic inspections of canisters to provide for the early detection of cracks well before they can grow through a canister wall. In addition, in all phases of dry storage (initial and renewed licenses), 10 CFR 50.47 and 72.32 require sites have an emergency plan that includes the capability to detect and mitigate accidents and promptly notify offsite response organizations.

Additionally, the NRC has evaluated the radiological impact of spent fuel canister breaches, although the studies are only partially applicable to the cracking condition postulated in the comment. In 1988, the NRC analyzed a postulated accident involving the removal of the lid of a storage cask in which all the fuel rods have been damaged (NUREG–1140). The analysis considered the release of krypton and iodine gases and found that doses were below the EPA's protective action guides for taking protective action after an accident. In 2007 (NUREG–1864) and 2014 (NUREG–2125), the NRC calculated the radiological risks of spent fuel storage and transportation, respectively. Those analyses considered the post-accident release of gases and particulates from breached storage canisters that contain damaged fuel assemblies. In those studies, a large portion of radionuclides within the canister was considered able to pass through the breach, as the sizes of canister breaches following an impact accident were either calculated to be relatively large (NUREG–2125) or simply assumed to be so (NUREG–1864). Those studies concluded that the risks of storage and transportation are low.

References

NRC, NUREG–1140, "A Regulatory Analysis on Emergency Preparedness for Fuel Cycle and Other Radioactive Material Licensees," ADAMS Accession No. ML062020791, Washington, DC, January 1998.

NRC, NUREG–1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant," ADAMS Accession No. ML071340012, Washington, DC, March 2007.

NRC, NUREG–2125, "Spent Fuel Transportation Risk Assessment," ADAMS Accession No. ML14031A323, Washington, DC, January 2014.

2.2 Comment from Nuclear Energy Institute

Comment 2.2.1

Comment: *Since Interim Staff Guidance (ISGs) will be incorporated into the new, combined NUREG–2215 currently issued in draft and sunset as documents, how will this NUREG keep from being out of date with its many references to ISGs?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The ISG documents will remain available to the public at the ADAMS locations referenced in NUREG-2214, and they will continue to provide a useful historical reference to support the technical basis of the aging management recommendations in NUREG–2214.

2.3 Comment from Raymond Lutz/Citizens Oversight, Inc.

Comment 2.3.1

Comment: *Although we view NUREG–2214 as a large step in the right direction as it contains a wealth of valuable information on aging processes and expectations, we have a fundamental disagreement with this document. The abstract says "The MAPS Report evaluates known aging degradation mechanisms to determine if they could affect the ability of dry storage system components to fulfill their safety functions in the 20- to 60-year period of extended operation." We view this time scale as to be insufficient, as we have outlined. Simply stated, 20 to 60 years does not acknowledge the clear reality of the likely situation, which we believe is 300 to 1,000 years, and that only deals with the first 1/150th of the problem.*

The NUREG–2214 should be enhanced by avoiding the view that we are only interested in the 20- to 60-year time frame. At present, if an aging mechanism is not expected to be significant within that period of interest, the current text just says it is "not credible." We would prefer that the full life of the subject material be provided, and if it is unknown, then that can be stated.

This would make the document useful for planning for the longer time scales we assert are necessary for a prudent spent fuel storage plan to be developed.

This document is based on an invalid assumption. It is not credible that spent fuel storage systems can exist for only 20 to 60 years. To make such an assumption is patently imprudent.

NRC Response: This comment is the subject of a petition for rulemaking currently under review by the NRC (Lutz, 2018), and no changes were made in response to the comment.

Should the regulations be changed to require that applicants demonstrate storage system performance beyond the requested license term, the staff would evaluate whether revisions to aging management guidance would be necessary.

Reference

Petition for Rulemaking, "Requirements for the Indefinite Storage of Spent Nuclear Fuel," Raymond Lutz and Citizens Oversight, Inc. PRM-72-8. Regulations.Gov Docket ID: NRC-2018-0017, January 2, 2018.

2.4 Comment from Marvin Lewis

Comment 2.4.1

Comment: *This is a review which attempts to look at all combinations and environments which might cause severe degradation. Although this approach is very time and expertise intensive, this approach has a history of failure to expose the obvious at Fukushima, Arkansas One, GINNA ad nauseam, ad infinitum (argument to nausea and from repetition). There is no reason to believe that this approach will be more predictive for MAPS.*

At Fukushima this approach failed to foresee a 9+ earthquake and a tsunami greater than 35 ft which allowed several reactors to fail catastrophically. Arkansas One experienced a crane drop wherein a crane dropped a half million pound stator through a floor and the ceiling of a switch gear room. The switch gear room flooded and remained flooded for 11 hours. Courageous work by electrician restarted the cooling fluid flow before the liquid level in the spent fuel pool dropped to the point of allowing a fire to start therein.

Conversely the approach of looking at history to learn lessons is abandoned in this draft, a serious deficiency. History is a great teacher. If we fail to learn from history, surely we will repeat the mistakes of the past.

At Fukushima management refused to see the possibility of a 9+ earthquake. It happened. At Arkansas One crane maintenance was delayed for economic considerations. A half million pound stator dropped. I hope that all reactors have separated electrical runs properly.

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As noted in NUREG–2214 Chapter 3, potential aging mechanisms were identified from reviews of gap assessments of DSSs, relevant technical literature, standard guides and reports, and operating experience from nuclear and non-nuclear applications. Each of the example AMPs in Chapter 6 includes a review of operational experience that supports the determination that the AMP is capable of maintaining SSC functions in the period of extended operation. NUREG–2214 also recommends that licensees adopt a learning aging management approach, as described in Section 3.6.1.10 of NUREG–1927, Revision 1. The staff's reviews of AMPs ensure that the application includes provisions to conduct periodic future reviews of operating experience to confirm the effectiveness of the AMPs or identify a need to enhance or modify an AMP.

CHAPTER 3: RESPONSES TO PUBLIC COMMENTS ON INTRODUCTION

The comments on Introduction relate to Chapter 1 of NUREG–2214.

3.1 Comment from NAC International

<u>Comment 3.1.1</u>

Comment: *P1-3, L3 - Add parenthetical "(CNWRA)" after "Analyses." Acronym is used in third sentence but hasn't been defined yet.*

NRC Response: The staff agrees with the comment and made the recommended change.

3.2 Comments from Nuclear Energy Institute

<u>Comment 3.2.1</u>

Comment: *P1-2, L21 - Suggest re-wording "contains one acceptable method" to "describe acceptable methods."*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 3.2.2</u>

Comment: *P1-2, L35 - Suggest adding NUHOMS HD (CoC 1030).*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As noted in NUREG–2214 Section 1.2, the guidance evaluates selected storage system designs to address near-term renewal applications. The NUHOMS HD design was not selected and evaluated, and, therefore, cannot be included in the scope of the guidance.

CHAPTER 4: RESPONSES TO PUBLIC COMMENTS ON DEFINITIONS

The comments on Definitions relate to Chapter 2 of NUREG–2214.

4.1 Comments from NAC International

Comment 4.1.1

Comment: *P2-3, Table 2-2 - Recommend expansion of Fully Encased (steel) (FE) to include neutron shielding and gamma shielding materials within sealed or welded steel enclosures such as the Transfer Cask Body and Shield Doors, and Vertical Concrete Cask Shield Plug and Lid. Currently some of these components are identified as embedded-NS or embedded-lead, which is not as good a description.*

NRC Response: The staff agrees with the comment and revised the guidance.

The staff added a sentence to the fully encased or lined (FE) term to read, “Neutron shielding and gamma shielding materials are often encased within a metal liner.”

Comment 4.1.2

Comment: *P2-8, Table 2-3 - Wet corrosion and blistering would only occur during repeated loading and drying, not single load storage operation. Given the relatively short duration of loading and drying, this mechanism is probably not substantial and would not occur over the period of extended operation (PEO).*

NRC Response: The staff agrees with the comment and revised the guidance.

The staff recognizes that blistering of aluminum-boron carbide laminate composites (e.g., Boral®) could potentially occur only during repeated wetting and drying, not after a single drying operation. The staff also recognizes that this conclusion is solely based on experimental testing, and that no operating experience has been documented in the United States on blistering of aluminum-boron carbide laminate composites used in loaded DSSs. Although indications of wet corrosion have been observed during DSS loadings (via indirect measurements of generated hydrogen), there is no evidence that this mechanism has led to blistering that could compromise the intended function of the aluminum-boron carbide laminate composite. The staff revised Table 2-3 and Section 3.4.2.3 accordingly.

4.2 Comments from Nuclear Energy Institute

Comment 4.2.1

Comment: *P2-2, Table 2-1, Term "Zirconium-based alloys" - The definition of high burnup fuel embedded in the definition of Zirconium-based alloys seems odd. The burnup of the fuel is not a material. Suggest deleting and defining separately in the text.*

NRC Response: The staff agrees with the comment and has removed the definition of high burnup fuel from the term.

The definition of high burnup fuel is provided on P3-86, L1-3 of the draft NUREG.

Comment 4.2.2

Comment: *P2-1, Table 2-1, Term "Boralyn, Metamic" - Suggest including Metamic-HT as it is a different material than the Metamic mentioned in the table with a different purpose, usage, and composition.*

NRC Response: The staff agrees with the comment and revised the guidance.

The staff added notes to the term to read, "Metamic-HT™ is a successor to the Metamic™ composite material. It possesses the necessary mechanical properties for fuel basket applications by strengthening its aluminum matrix with nano-particles of aluminum oxide."

Comment 4.2.3

Comment: *P2-2, Table 2-1 - Add definition of Metal Matrix Composite (MMC) to the list.*

NRC Response: The staff agrees, in part, with the comment and revised NUREG-2214 to clarify.

Table 2-1 is intended to define the specific materials as they appear in the Chapter 4 aging management tables (e.g., Metamic™). Rather than include a definition of a broad material class in Table 2-1, the staff expanded the existing description of metal matrix composites in Section 3.4.2.

<u>Comment 4.2.4</u>

Comment: *P2-3, Table 2-2, Term "Fully encased or lined (FE)" - Recommend expansion of Fully Encased (steel) (FE) to include neutron shielding and gamma shielding materials within sealed or welded steel enclosures such as the Transfer Cask Body and Shield Doors, and Vertical Concrete Cask Shield Plug and Lid. Currently some of these components are identified as embedded-NS or embedded-lead, which is not as good a description.*

NRC Response: The staff agrees with the comment and revised the guidance.

See the response to Comment 4.1.1.

<u>Comment 4.2.5</u>

Comment: *P2-3, Table 2-2, Term "Air–outdoor (OD)" - For "OD," it is overly conservative to assume outdoor air for transfer casks (TCs) subject to indoor air because the TCs are generally not exposed to precipitation, wind and salt-laden air for extended periods of time (if at all for many vertical systems). It would be more appropriate for a TC to be assigned the "SH" environment.*

NRC Response: The staff disagrees with the comment; however, a minor change was made in response to the comment.

As noted in the aging management review (AMR) tables in Chapter 4 for various transfer cask designs, the TC external components are exposed to both indoor and outdoor air environments. Also, licenses and CoCs typically do not include requirements for where the TC is to be stored between uses. Therefore, it is appropriate to conservatively assume an outdoor air environment. The staff changed the description of the TC environment in Table 2-2 to "indoor/outdoor air."

<u>Comment 4.2.6</u>

Comment: *P2-7, Table 2-3, Term "Shrinkage" - Shrinkage cracking also occurs from sharp corner geometries in the design, such as on outlet vents. In these instances, once the crack forms, the stress is relieved and the crack does not grow.*

NRC Response: The staff agrees, in part, with the comment; however, no changes were made in response to the comment.

The staff agrees with the comment that sharp corner geometries in the design are prone to shrinkage cracking. However, the referenced definition of shrinkage in Table 2-3 is intended to describe potential aging-related changes to the material that cause concrete to be susceptible to

such cracking. Also, as summarized in Table 3-5, the staff concluded that cracking due to shrinkage was not a credible aging effect in the 20- to 60-year period of extended operation.

<u>**Comment 4.2.7**</u>

Comment: *P2-8, Table 2-3, Term "Wet corrosion and blistering" - It seems inappropriate to include wet corrosion and blistering as an aging mechanism for dry storage systems. The described phenomenon occurs during loading and drying, not storage operation, so it's not a mechanism that's possible in a dry storage cask over the PEO. Blistering, if it occurs, does not affect functionality of the material.*

NRC Response: The staff agrees with the comment and revised the guidance.

See the response to Comment 4.1.2.

<u>**Comment 4.2.8**</u>

Comment: *P2-9, Table 2-4, Term "Loss of criticality control" – "Loss of criticality control" is poor wording because it connotes a <u>complete</u> loss of criticality control or even a critical condition. Suggest using "Reduction in neutron attenuation" instead.*

NRC Response: The staff agrees with the comment and revised the guidance.

The staff notes that NUREG–1801, "Generic Aging Lessons Learned (GALL) Report," Revision 2, Section IX.E uses the term "Reduction of neutron-absorbing capacity" as an aging effect resulting from Boraflex degradation. Boraflex is a neutron-absorbing material used in spent fuel storage racks. Therefore, the staff changed the term "Loss of criticality control" to "Reduction of neutron-absorbing capacity" throughout the guidance.

<u>**Comment 4.2.9**</u>

Comment: *P2-9, Table 2-4, Terms "Loss of bond" and "Loss of material" – "Loss of bond" and "Loss of material" should include reference to these effects on coatings.*

NRC Response: The staff disagrees with the comment; however, a minor change was made in response to the comment.

Although coating degradation and loss of material of the base metal are related, the staff intends the definition of loss of material to describe only the specific mechanisms by which materials degrade.

In its review of this comment, the staff noted that “loss of bond” is used only in the Reinforced Concrete Structures AMP as a parameter monitored. Loss of bond is related to the loss of cohesion between the steel reinforcement bar and surrounding concrete. However, since loss of bond is not defined in the acceptance criteria of ACI 349.3R-02, the staff has opted to remove the term from NUREG-2214.

<u>**Comment 4.2.10**</u>

Comment: *P2-9, Table 2-4, Term “None” - Is it necessary to include “None”? Isn’t it obvious that that’s a choice?*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

This term provides clarity of the term “none identified” used in the AMR tables.

CHAPTER 5: RESPONSES TO PUBLIC COMMENTS ON EVALUATION OF AGING MECHANISMS

The comments on Evaluation of Aging Mechanisms relate to Chapter 3 of NUREG–2214.

5.1 Comments from Donna Gilmore/SanOnofreSafety.org

Comment 5.1.1

Comment: *I reviewed NUREG/CR–7198 (and Rev 1 which replaces the original) referenced in NUREG–2214. It does not provide sufficient evidence that it can be used as confirmatory research to justify approval of the storage (or transport) of high burnup fuel or the safety of cladding. Neither NUREG mentions it is applicable to seismic loading, nor that it is applicable to all fuels and claddings. Where is the other evidence to support high burnup cladding will not fail?*

NRC Response: The staff reviewed the comment and revised the guidance to clarify the basis for the high burnup fuel recommendations.

NUREG/CR–7198, Revision 1, had not been published at the time that NUREG–2214 was issued for public comment. Therefore, NUREG–2214 does not provide an assessment of the test results in NUREG/CR–7198, Revision 1. The staff has issued a draft report for public comment, NUREG–2224, "Dry Storage and Transportation of High Burnup Fuel," (ADAMS Accession No. ML18214A132) which provides the technical bases supporting the conclusion that hydride reorientation is inconsequential to the expected loads during design-basis drop accidents in storage and during seismic loading conditions. The applicability of this conclusion to different fuel types and cladding alloys is addressed in NUREG–2224. The staff has revised Section 3.6.1.1 to provide references to NUREG–2224 and NUREG/CR–7198, Revision 1. The staff has also ensured that the conclusions in both NUREG–2214 and NUREG–2224 are consistent.

Comment 5.1.2

Comment: *NUREG/CR–7198 was done assuming undamaged fuel pellets. As you likely know, the uranium pellets can change their grain structure from high burnup fuel and other causes. Is there another study that addresses damaged fuel pellets?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

NUREG/CR–7198, Revision 1, did not assess the condition of the fuel pellet prior to testing. Therefore, the statement in the comment that testing was based on undamaged fuel pellets is unsubstantiated. The staff's analysis of the NUREG/CR–7198 test results is documented in NUREG–2224, "Dry Storage and Transportation of High Burnup Fuel, Draft Report for Comment" (ADAMS Accession No. ML18214A132). The conclusions in NUREG–2214 were

developed in consideration of the microstructure of high burnup fuel pellets, including the effects of the potential for fuel pellet fragmentation during reactor irradiation. The staff also notes that the safety review of the structural performance of high burnup fuel assemblies loaded, to date, conservatively assumes that the fuel pellets do not lend structural support to the cladding. Therefore, the mechanical properties of the fuel pellet are not a consideration during the safety review of dry storage systems.

In addition, the staff is not aware of the nomenclature used by the commenter ("damaged" or "undamaged" fuel pellets) and thus cannot comment on the implications of these pellet conditions.

Comment 5.1.3

Comment: *NUREG/CR–7198 mentions the cladding failures were at the pellet to pellet boundary, which is an indicator it relied on undamaged and intact fuel pellets for structural support. Therefore, if the cladding is dependent on the fuel pellets, that is an unresolved technology gap, so is insufficient to use as confirmatory information without the data on other than intact fuel pellets.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

NUREG/CR–7198, Revision 1, does not conclude that failures occurred at the pellet-pellet interface due to the presence of "undamaged" pellets (see the response to Comment 5.1.2). The staff notes that tensile stresses are expected to be highest at the pellet-pellet interface, which may have contributed to the resultant failure location.

Comment 5.1.4

Comment: *Uranium structure can change with higher burnup fuels. NUREG/CR–7198 use of HB Robinson for irradiated fuel is not typical high burnup fuel, likely due to the relatively low linear power compared to others. Therefore, it is not a conservative evaluation of high burnup fuel or fuel cladding.*

Characterization of High-Burnup PWR and BWR Rods, Hanchung Tsai (htsai@anl.gov), Mike Billone (Billone@anl.gov), Argonne National Laboratory, Nuclear Safety Research Conference 2002, October 28-30, 2002 https://www.nrc.gov/docs/ML0230/ML023050234.pdf

Slide 4

Limerick and H. B. Robinson Rod Characterization
Fission Gas Release

- *H.B. Robinson rods: 1.4-2.5%*
  - *Probably due to the relatively low linear power: ~8 kW/ft BOL decreasing to 3-4 kW/ft EOL*

- *Limerick rods: 5-17%*
  - *Relatively high release attributable possibly to fuel microstructure*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff recognizes that the pellet microstructure of high burnup fuel is different than for low burnup fuel. However, the staff disagrees with the assessment that the fission gas considered in the NUREG–2224 technical bases is inadequate. The conclusions in NUREG–2224 are not based on the actual end-of-life pressures for the H.B. Robinson fuel tested under the research program discussed in NUREG/CR–7198, Revision 1. The end-of-life pressures considered for all stress-driven aging mechanisms is conservative, as it considers bounding fission and decay gas releases from fuel rods with zirconium-diboride burnable absorbers. These pressures would be the highest in the fleet of fuel that can be loaded for dry storage.

Comment 5.1.5

Comment: *The below Oak Ridge slides show some other variables. For example, in the heated samples (heated from the outside in) a bonding of the pellets to the cladding occurred. See other items in red. More unknowns rather than sufficient data.*

CIRFT Testing of High Burnup Used Nuclear Fuel from PWR and BWRs (slides), J.-A. Wang, H. Wang, H. Jiang, Y. Yan, B. Bevard, Oak Ridge National Laboratory, 2016 Nuclear Waste Technical Review Board (NWTRB), February 17, 2016 http://www.nwtrb.gov/docs/default-source/meetings/2016/february/bevard.pdf?sfvrsn=8

Slide 11

HBR Spent Nuclear Fuel (SNF) S-N data indicates a hydrogen content dependency

Hydrogen was estimated from oxide thickness; detailed hydrogen measurements are needed to further quantify any hydrogen-dependent failure mechanism

Slide 17

Annealed SNF CIRFT testing is needed to allow an accurate comparison between HBU CIRFT data and HR CIRFT data

The hydride reorientation (HR) sample preparation has the potential to introduce a material bias into test results due to the followings:

- *Induces a thermal annealing effect in the clad tubing structure;*

- *Heat source of HR samples is initiated from the clad outer surface (~400 °C), which may reduce clad compressive radial stress and generate a thermal gradient from clad to fuel*

- *In-situ pressurization of 3,500 psi pressure may reduce the radiation induced clad crimping effect (of 2,450 psi coolant pressure) and counter/decrease the radial compressive residual stress that occurs during clad irradiation;*

- *Combined effects of thermal annealing & clad pressurization could permanently change clad geometry (i.e. enlarge the clad inner wall radius) and reduce the pellet support to the clad.*

Slide 18

Initial observations from CIRFT testing include: [why only talking about HBR (HB Robinson fuel)]?

- *The fuel provides strength (flexural rigidity) to the fuel/clad system*

- *When the clad is fatigued to failure, failure occurs primarily at the pellet-pellet interface*

- *The fuel pellets retain their shape (dishing and chamfering is evident) and do not become fragmented - very little residue is released from rods that are broken into two pieces*

- *Considering the complexity and nonuniformity of the HBU fuel cladding system, it was significant to find that the strain to failure data for the SNF was characterized by a curve expected of standard uniform materials*

- *It was significant to find that the HBU HBR exhibited an endurance limit, if an endurance limit is defined by survival of >10^7 cycles*

- *At low loads, the PWR HBR fuel did not fail after >10M cycles*

Slide 19

Other observations from the CIRFT testing:

- *Pellet-clad-interaction includes P-C bonding efficiency*

- *Hydrogen concentration does affect SNF system strength*

- *The SNF system has significant stress concentrations and residual stress distributions*

- *It appears that transient shock accumulated damage may reduce the SNF fatigue lifetime*

- *In addition to the fatigue strength data, fracture toughness data of the SNF system is also essential to assist in the SNF vibration reliability study, especially in a high-rate loading arena*

NRC Response: The staff reviewed the comment and revised the guidance to clarify the basis for the high burnup fuel recommendations.

See the response to Comment 5.1.1.

Comment 5.1.6

Comment: *Here are the two reports I reference regarding damaged fuel issues. Were these considered in NUREG–2214?*

Impact of High Burnup Uranium Oxide and Mixed Uranium-Plutonium Oxide Water Reactor Fuel on Spent Fuel Management, IAEA Nuclear Energy Series, No. NF-T-3.8, VIENNA, 2011. http://www-pub.iaea.org/MTCD/Publications/PDF/Pub1490_web.pdf

Page 36:

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).[9] The rim portion of high burnup fuel will have much higher burnups than the pellet average and forms restructured fine subgrains at pellet average burnups > 40 GWd/tU. The subgrain sizes are generally between 0.1 μm to 0.3 μm [39, 49-51]. As the burnup of the rim increases the original as-fabricated grain boundaries begins to disappear as the subgrain structure becomes dominant. This restructured rim is not present in the older fuel where rod or bundle burnups did not exceed 33 GWd/tU.

Damaged Spent Nuclear Fuel at U.S. DOE Facilities, Experience and Lessons Learned, by INL, Nov 2005 INL/EXT-05-00760, https://inldigitallibrary.inl.gov/sites/sti/sti/3396549.pdf

Page 4&5:

The uranium metal SNF [Spend Nuclear Fuel] within the DOE inventory contains many elements whose cladding was breached during reactor discharge, subsequent handling, or storage. Initial cladding failures varied from minor cracks to severed fuel elements. The reaction of exposed uranium metal with water produces uranium dioxide and hydrogen. This reaction is not a result of chemical impurity of the basin water. It is a chemical reaction of the water with the uranium metal. Uranium hydride forms from the available hydrogen, particularly where there is a limited amount of oxygen (see Reference 3). The lower densities of the uranium oxide and uranium hydride products relative to the uranium metal cause swelling of the material within the cladding and subsequent additional cladding damage. Additional water reaction then occurs with the newly exposed uranium metal. Each cycle of fuel-water reaction results in fission product releases and contamination of water in the canister or the storage pool. Examples of uranium metal SNF element damage after extended water storage are shown in Figure 3. In extreme cases, the uranium metal has also been known to completely oxidize and form a mud-like mixture with the water.

The generation of high surface area uranium metal SNF fragments and uranium hydride necessitates additional measures during SNF drying, dry storage, and transportation because of

the pyrophoric nature of these materials when exposed to air. As a result, degraded uranium metal fuels are stored and transported in inerted canisters after removal from the basin and drying. Radiolysis of water within the SNF-water corrosion products must also be addressed for long-term storage because of the ability of the resultant gases to overpressurize containers, embrittle welds on containers, and reach flammable concentrations.

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

In response to the cited statement on page 36 of IAEA Nuclear Energy Series, Report No. NF-T-3.8, the staff has determined that changes to the guidance are not necessary. The conclusions in NUREG–2214 were developed in consideration of the microstructure of high burnup fuel pellets. The staff also notes that the safety review of the structural performance of high burnup fuel assemblies loaded, to date, conservatively assumes that the fuel pellets do not lend structural support to the cladding. Therefore, the mechanical properties of the fuel pellet are not a consideration during the safety review of dry storage systems.

In response to the cited statement on pages 4 and 5 of Report INL/EXT-05-00760, the staff disagrees with the comment. NUREG–2214 only considers uranium oxide fuel. The cited reference pertains to uranium metal SNF stored by the U.S. Department of Energy, which is not representative of fuel used for commercial power generation.

__Comment 5.1.7__

Comment: *Would you please provide the technical references used to substantiate the below statement in NUREG–2214. If I'm understanding this correctly and after reviewing other information, it's claiming cladding embrittlement is only a problem in something as severe as a transport accident, such as with a 30 foot drop, or when retrieving fuel assemblies. Is there a seismic analysis that was done that addresses cladding embrittlement in dry storage in a high risk earthquake zone such as San Onofre?*

Page 3.6:

**Although hydride reorientation and hydride-induced embrittlement of high-burnup cladding is credible, these mechanisms are only expected to potentially compromise intended functions under pinch-type loads. Such loads are not expected to be present during storage.*

NRC Response: The staff reviewed the comment and revised the guidance to clarify the basis for the high burnup fuel recommendations.

The staff has issued a draft report for public comment, NUREG–2224, "Dry Storage and Transportation of High Burnup Fuel," (ADAMS Accession No. ML18214A132) which provides this assessment and technical bases supporting the conclusion that hydride reorientation is inconsequential to the expected loads during design-basis drop accidents in storage and during

seismic loading conditions. The staff has revised Section 3.6.1.1 to provide a reference to NUREG–2224 and ensured that the conclusions in both documents are consistent.

5.2 Comments from NAC International

Comment 5.2.1

Comment: *P3-2, Table 3-2 - It seems that all nine environments in Table 3-1 should be accounted for between the Credible and Noncredible Environment columns. The assumption is that the ones not listed are also noncredible.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As noted in Section 3.1, not all combinations of materials, environments, and aging mechanisms are evaluated in each major component area. This occurs because some material-environment combinations do not exist in every major component area or, in some instances, aging mechanisms were not considered to be reasonably plausible, and thus an evaluation was not performed.

Comment 5.2.2

Comment: *P3-6, Table 3-6 - Should include Stainless Cladding fuel cladding materials*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As noted in Section 3.6, the cladding materials evaluated in the guidance are zirconium-based alloys. The staff considers the aging management review of stainless steel cladding alloys to be best addressed on a case-by-case basis.

Comment 5.2.3

Comment: *P3-7, L6 - Indoor air is not a listed environment, sheltered is probably the environment being referred to.*

NRC Response: The staff agrees, in part, with the comment; however, no changes were made in response to the comment.

As noted in the definition of “Air-Outdoor” in Table 2.2, the combination of indoor and outdoor air exposure for transfer cask components is conservatively evaluated as outdoor air. However, the staff considered it appropriate to note the indoor air environment in the subject text.

Comment 5.2.4

Comment: *P3-12, L11 - Although microbiologically influenced corrosion (MIC) is considered credible, it probably isn't plausible on embedded steel with an engineered soil that is well drained and clean under the pad.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As discussed in the concrete aging mechanism reviews in Section 3.5, the staff finds that there is sufficient operating experience of concrete degradation over extended service lives to warrant the consideration of groundwater interactions with embedded steel. However, a renewal applicant may proposed an alternative approach to this generic evaluation that is supported by site-specific considerations and operating experience.

Comment 5.2.5

Comment: *P3-22, L28 – ‘eubcomponents’ should be subcomponents*

NRC Response: The staff agrees with the comment and has corrected the error.

Comment 5.2.6

Comment: *P3-34, L28 - Editorial - Return error on last sentence*

NRC Response: The staff agrees with the comment and has corrected the error.

Comment 5.2.7

Comment: *P3-34, L27&28 – Formatting*

NRC Response: The staff agrees with the comment and has corrected the error.

Comment 5.2.8

Comment: *P3-66, L6 - Groundwater monitoring would be conducted when a pad is in scope and there is reason to believe that the groundwater is conducive to concrete degradation. If a site has historical groundwater monitoring demonstrating low risk for concrete degradation, then this should be allowed to be used to show no AMP is necessary, or at least a reduced monitoring frequency, since groundwater chemistry has been shown to change very little over many years of trending when there are no new contributors to a site.*

NRC Response: The staff agrees with the comment; however, it concluded that changes to the guidance are not necessary.

The staff considers that Element 4 of the Reinforced Concrete Structures AMP allows ample flexibility for proposing a groundwater chemistry monitoring frequency that is supported by site-specific considerations and operating experience. Also, as noted in Section 1.1, this guidance provides a generic evaluation of aging mechanisms and example aging management programs. A renewal applicant may proposed an alternative approach.

Comment 5.2.9

Comment: *P3-36, L6 - Editorial - Return error on sentence*

NRC Response: The staff agrees with the comment and has corrected the error.

Comment 5.2.10

Comment: *P3-36, L6 – Formatting*

NRC Response: The staff agrees with the comment and has corrected the error.

Comment 5.2.11

Comment: *P3-69, L35-37 - Minor and moderate calcium leaching is a common occurrence identified during the annual vertical concrete cask (VCC) inspections. NAC believes a threshold is needed for an acceptance criterion, so sites are not evaluating numerous areas that are not compromising function. A reasonable threshold would be one that is beyond the minor and moderate leaching seen and offer that "Excessive leaching of calcium hydroxide" can be used as a criterion.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As noted in Table 6-3, Element 6, the acceptance criteria for visual inspections of concrete are commensurate with the 3-tier quantitative criteria in ACI 349.3R-02. The staff considers this consensus industry standard to be appropriate for generic guidance.

Comment 5.2.12

Comment: *P3-72, L14 - Although MIC is considered credible, it probably isn't plausible on concrete with an engineered soil that is well drained and clean under the pad, especially with no operating experience to show any history.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment

See the response to Comment 5.2.4.

Comment 5.2.13

Comment: *P3-73, L18 – "C-S-H" term has not been defined.*

NRC Response: The staff agrees with the comment and has defined the term calcium-silicate-hydrate.

Comment 5.2.14

Comment: *P3-75, L24 – Formatting*

NRC Response: The staff agrees with the comment and has corrected the error.

Comment 5.2.15

Comment: *P3-85, Sec3.6 - Does not include Stainless Clad Fuel Assemblies. LACBWR fuel is 348H SS.*

NRC Response: The staff agrees with the comment; however, no changes were made in response to the comment.

See the response to Comment 5.2.2.

Comment 5.2.16

Comment: *P3-87, L31 – Formatting*

NRC Response: The staff agrees with the comment and has corrected the error.

Comment 5.2.17

Comment: *P3-101, Sec3.6.2 - LACBWR assemblies have Inconel 600 and 304 SS components.*

NRC Response: The staff agrees with the comment and revised the guidance.

As noted in the AMR tables in Chapter 4, only a general description is given for the materials of construction. The staff changed the third sentence of Section 3.6.2 to read, "The other components are fabricated using various nickel alloys and stainless steel."

5.3 Comments from Nuclear Energy Institute

Comment 5.3.1

Comment: *P3-5, Table 3-5 - Differential settlement is not an aging mechanism associated with Air-Outdoor environment or Sheltered environment.*

NRC Response: The staff agrees with the comment and has revised the guidance to associate differential settlement only with concrete structures in contact with "groundwater/soil" environments.

The staff notes that, although portions of concrete structures in contact with air can exhibit the effects of settlement, the direct interaction with the underlying soil is the cause of the aging mechanism.

<u>Comment 5.3.2</u>

Comment: *P3-5, Table 3-5 - It is hard to imagine what kind of environment would qualify as an aggressive chemical attack environment that would be associated with Outdoor Air for concrete. This table should delete this environment.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As noted in Section 3.5.1.5, the source of aggressive ions or acids can be present in groundwater, seawater, and rainwater. Therefore, aggressive chemical attack of concretes exposed to sheltered, outdoor, and groundwater or soil (below-grade) environments is considered credible, and aging management is required.

<u>Comment 5.3.3</u>

Comment: *P3-5, Table 3-5 - Microbiological degradation of concrete is not a common mode of degradation and generally occurs in environments that are not typical of ISFSIs, such as seawater service and sewage treatment. Since OE indicates that it is not occurring in concrete in nuclear plant environments, there is insufficient justification for including it in MAPS. Even if it were to occur, it would produce similar results to aggressive chemical attack (it is indeed a form of that phenomenon) which is included. It should be classified as not credible.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The staff recognizes that no cases of microbiological degradation of concrete have been reported in nuclear applications. However, as noted in Section 3.5.1.12, non-nuclear related operating experience supports a conclusion that the degradation mode is credible. The staff agrees that several concrete aging mechanisms can lead to similar effects (e.g., loss of material); however, the staff did not use that as a criterion when considering whether a particular aging mechanism was credible. No changes were made to the guidance in response to the comment.

<u>Comment 5.3.4</u>

Comment: *P3-5, Table 3-5 - Salt scaling cannot happen below grade and hence the GW environment should be excluded. The “freeze line” NRC must be referring to is what is most commonly known as the “frost line” which is by definition, the depth below which soil does not freeze in the winter. Salt scaling, if it occurs, would only be above ground. Salts and thawing agents are generally prohibited from ISFSIs in plants subject to these conditions.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As noted in Section 3.5.1.1, for below-grade concrete structures above the freeze line, water that resides in soil can also be subject to freezing conditions, potentially promoting freeze and thaw damage. This below-grade environment is also applicable to salt scaling because it is closely related to freeze and thaw damage. Therefore, salt scaling of concretes exposed to groundwater/soil (below-grade) environment is considered credible, and aging management is required. The staff added a sentence to the second paragraph of Section 3.5.1.14 to read, "For below-grade concrete structures above the freeze line, saline water that can reside in soil in contact with concrete can be subject to freezing conditions, potentially promoting salt scaling."

The staff recognizes that freeze line and frost line are two interchangeable terms, and therefore, it is not necessary to change the term "freeze line" used in the guidance.

Comment 5.3.5

Comment: *P3-8, L18 - Suggest adding a parenthetical definition of "deliquescence."*

NRC Response: The staff agrees with the comment and added a footnote to define the term to read, "Deliquescence refers to a process wherein the species, such as a hygroscopic salt, absorbs water from the air in conditions of high relative humidity."

Comment 5.3.6

Comment: *P3-8, L35 - Are there temperature-humidity combinations where general corrosion becomes life-limiting? Change wording to ".......to outdoor and sheltered environments are potentially present, but general corrosion, although plausible, will not propagate at rates sufficient to affect component intended function. Therefore aging management of general corrosion is not required during the 60 year time frame."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

General corrosion of steel exposed to outdoor air is a well-known phenomenon, as evidenced by common maintenance practices at nuclear power facilities to periodically inspect outdoor steel components for evidence of corrosion. In the development of its generic conclusion on the credibly of this aging mechanism, the staff did not perform component-specific analyses of part dimensions, applied stresses, and corrosion rates that would be necessary to determine if particular component functions could be challenged over a 60-year service life. However, a renewal applicant may propose an alternative approach to the staff's generic evaluation that is supported by component-specific and site-specific considerations.

Comment 5.3.7

Comment: *P3-9, L23-25 - Steel exposed to an embedded concrete environment should be protected and general corrosion is not expected. Only when the concrete becomes faulted and allows the rebar to be exposed to other environments does the possibility of corrosion occur. This document should not assume that the concrete will become compromised and the rebar exposed. Aging management of concrete should prevent this type of corrosion from occurring.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As discussed in the concrete aging mechanism reviews in Section 3.5, the staff finds that there is sufficient operating experience of concrete degradation over extended service lives to warrant the consideration of environmental interactions with embedded steel. The evaluation of steel exposed to an embedded concrete environment is similar to that of reinforcing bar corrosion. As noted in Section 3.5.1.6, corrosion of the reinforcing steel in concrete can potentially initiate and propagate within the 60-year timeframe for concretes of moderate to low quality. As concrete degrades with time, steel can be exposed to aqueous electrolytes conducive to corrosion mechanisms (i.e., general corrosion, pitting and crevice corrosion, and MIC). Thus, these corrosion mechanisms are considered to be credible for steels exposed to an embedded (concrete) environment, and therefore, aging management is required.

Comment 5.3.8

Comment: *P3-10, L40 - Define the term "electroless."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment

Electroless nickel coatings are commonly used on carbon steel components by the nuclear industry for corrosion protection. The first appearance of the term "electroless nickel" is in the definition for "steel" on P2-2 of the guidance, where it is used in the context of nickel plating. The staff believes that it is not necessary to define this term in greater detail.

Comment 5.3.9

Comment: *P3-10, L15-16&21-23 - Steel exposed to an embedded concrete environment should be protected and pitting and crevice corrosion is not expected. Only when the concrete becomes faulted and allows the rebar to be exposed to other environments does the possibility of this corrosion occur. This document should not assume that the concrete will become compromised and the rebar exposed. Aging management of concrete should prevent this type of corrosion from occurring.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.7.

Comment 5.3.10

Comment: *P3-11&12, Sec3.2.1.4 - This section discusses MIC, but does not justify the inclusion of steel embedded in concrete environments as a credible environment. It assumes that the concrete exposed to soil is in a sufficiently degraded condition to allow rebar to be exposed to microbes (if they are present and are of an aggressive type). We have not seen this kind of damage in nuclear plants which indicates that this scenario is highly unlikely and does not warrant inclusion. This document should not assume that the concrete will become compromised and the rebar exposed. There are more likely credible environments that will result in pad inspections without confusing the issue with improbable scenarios.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the responses to Comments 5.2.4 and 5.3.7.

Comment 5.3.11

Comment: *P3-14, L10 - What is "significant" creep? Undefined term casts doubt. Either define or delete. Delete "significant."*

NRC Response: The staff agrees with the comment and deleted the term "significant."

The staff concluded that the term "significant" was vague and was not necessary in the discussion of creep. As a general rule-of-thumb, temperatures greater than about 40 percent of a metal's melting point (in Kelvin) are required for creep deformation to credibly alter a component's configuration and affect its intended functions.

Comment 5.3.12

Comment: *P3-19, L4 - The initiation of SCC requires the presence of stress, environment and susceptible material. Absent anyone of these, there is no SCC. What stresses are considered? Not stated here. Should be included with a comment on the magnitude relative to SCC susceptibility. Page 3-19 line 1 change to: ".....are known to be precursors to SCC in the presence of weld residual stresses of sufficient magnitude to initiate SCC."*

NRC Response: The staff agrees with the comment. As noted in Section 3.2.2.5, SCC requires the presence of a tensile stress, which commonly exists at welds originating from fabrication processes, contacts between components, and bolted structures. The staff revised the sentence to read, "However, both pitting and crevice corrosion are known to be precursors to SCC if sufficient stress exists."

Comment 5.3.13

Comment: *P3-22, L1 - What is meant by "indoor/outdoor environments"? Doesn't the outdoor part of this statement contradict the previous section? Shouldn't "Indoor Environment" be listed in Table 3-1?*

NRC Response: The staff reviewed the comment and revised the guidance to clarify the use of "indoor/outdoor."

As stated in the responses to comments 4.2.5 and 5.2.3, transfer casks are exposed to both indoor and outdoor environments, but they are generally evaluated as being exposed to outdoor air to be conservative. However, SCC of stainless steel transfer cask components is evaluated as a special case because periodic rinsing of the cask removes halides that could cause SCC. As a result, a separate "outdoor air" conclusion is necessary for stainless steel transfer cask components. Section 3.2.2.5 was revised to clarify.

Comment 5.3.14

Comment: *P3-22, L28 - Typo - Should be "subcomponents"*

NRC Response: The staff agrees with the comment and has corrected the error.

Comment 5.3.15

Comment: *P3-25, L1-27 - In this section, thermal aging of 17-4 PH in a helium environment is deemed a credible aging mechanism based on experiences in the reactor coolant system which is a significantly different environment. The ASME temperature limits for this alloy seems to be a reasonable approach to take, so the recommended action from the users should entail demonstrating that it stays below this temperature limit or that the material meets the necessary material requirements if modelling shows it does not meet these temperature limits.*

NRC Response: The staff agrees with the comment; however, the staff concluded that no changes to the guidance are necessary.

As noted in Section 3.2.2.8, the degree of embrittlement of a specific SSC will depend on the service temperature and time duration, as well as the initial heat treatment condition of the SSC. As such, a review of the thermal aging effects should be performed on a case-by-case basis for all SSCs constructed from Type 17-4 PH stainless steel. A renewal applicant is expected to provide a bounding analysis to show that a reduction in mechanical properties due to thermal aging is not expected to compromise the SSC's intended function.

Comment 5.3.16

Comment: *P3-27, L35 - Wording question? Maybe better to state "could occur in spite of the passive oxide layer on the surface of aluminum materials."*

NRC Response: The staff agrees with the comment and revised the sentence to read, "However, localized corrosion in the form of pitting or crevice corrosion could occur on aluminum subcomponents, especially in the presence of halides."

Comment 5.3.17

Comment: *P3-36, L28 - The corrosion damage calculated (0.6 mm over 60 years), which is certainly conservative, is hardly concerning from a general corrosion standpoint and shows why copper is used all over the world in exposed application. General corrosion should not be credible. It should be the same as pitting and crevice corrosion.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The staff recognizes that copper alloys have been commonly used for various applications exposed to outdoor air. However, in the development of its generic conclusion on the credibly of this aging mechanism, the staff did not perform component-specific analyses of part dimensions, applied stresses, and corrosion rates that would be necessary to determine if particular component functions could be challenged over a 60-year service life. In the absence of aging-related calculations or analyses of specific SSCs fabricated from copper alloys, material thinning from general corrosion up to 0.6 mm [23.6 mils] may have the potential to challenge an SSC's intended functions. As such, general corrosion of copper alloys exposed to an outdoor air environment is considered to be credible, and therefore, aging management is required. This response is similar to that for Comment 5.3.6.

Comment 5.3.18

Comment: *P3-49, L3 - Sentence should read as follows: "...materials. Hydrogen in the water reduces the energy..."*

NRC Response: The staff agrees with the comment and revised the guidance.

Neutron shielding materials are known to achieve effective shielding by combining hydrogen and oxygen as moderators with boron as an absorber. The staff revised the sentence to read, “Hydrogen and oxygen in shielding materials reduce the energy of the neutrons such that the neutrons are more effectively absorbed by the boron.”

Comment 5.3.19

Comment: *P3-49, L4 - Please clarify what is meant by “possible relocation of shielding materials,” Is this meant to be dislocation or cracking?*

NRC Response: The staff reviewed the comment and revised the guidance to clarify.

The wording in question is meant to indicate possible changes in physical reconfiguration of shielding materials. The staff revised the sentence to read, “The degradation and possible changes in physical reconfiguration of shielding materials may be mitigated by encasing or reinforcing materials.”

Comment 5.3.20

Comment: *P3-51, L15&17 - Lines should begin with “EPRI.”*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The staff follows the formatting convention described in the NRC Style Guide, page 60, (NRC, 2009) for multiple references from the same author.

Reference

NRC, NUREG-1379, Revision 2, “NRC Editorial Style Guide,” ADAMS Accession No. ML093280744, Washington, DC: U.S. Nuclear Regulatory Commission. May 2009.

Comment 5.3.21

Comment: *P3-51, L32&35 - Lines should begin with “NRC.”*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.20.

Comment 5.3.22

Comment: *P3-52, L1&3 - Lines should begin with "NRC."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.20.

Comment 5.3.23

Comment: *P3-54, L44 - Typo - The word "verticle" should be "vertical."*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 5.3.24

Comment: *P3-54, L23-35 - On page 3-54, lines 23 and 24, it states that for borated stainless steel, "boron depletion is not considered to be credible, and therefore, aging management is not required during the 60-year timeframe." The next paragraph (i.e., Line 25) changes the language to "boron depletion in borated stainless steel is not generally considered to be a credible aging mechanism."*

The addition of "generally" is incorrect. It appears that the third paragraph in Section 3.4.1.1 is simply trying to say that any past boron depletion materials in the original application basis be treated in the renewal application (vs. ignoring it). The term "generally" may have been included based on past calculations by some vendors to determine the impact of potential boron depletion. The term should be deleted as these calculations were, and are still not necessary for this impact.

NRC Response: The staff agrees with the comment and revised the sentence to remove "generally."

Comment 5.3.25

Comment: *P3-55, Sec3.4.2 - This section does not appear to address Metamic-HT which is a different MMC than the Boral or Metamic described.*

NRC Response: The staff agrees with the comment and added a sentence to Section 3.4.2 to read, "The Metamic™ composite material is used in the Standardized NUHOMS, HI-STORM 100, and HI-STAR 100 systems for neutron poisons, while Metamic-HT™ is used to construct one of the Holtec's multipurpose canister fuel basket designs to fulfill both neutron absorbing and structural functions."

Comment 5.3.26

Comment: *P3-56, L43 - The following statement begins, "It is important to note that, because only a trace amount of water will be left in a dry storage cask after dehydration and helium backfill, the occurrence of wet corrosion and blistering will be minimal in a dry cask environment during the period of extended operation." This continues on page 3-57, line 3 which states: "corrosion and blistering will be minimal." This should also state that since DCS system temperature decays over time and there is a very small, finite source of water in a dried and sealed cask, the progression of any corrosion and blistering will be minimal during the period of extended operation.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff notes that Section 3.4.2.3 provides a sufficient technical basis for determining the credibility of wet corrosion and blistering for aluminum boron carbide laminate composites in DSSs.

Comment 5.3.27

Comment: *P3-57, L22-26 - Boron depletion, states that the generic evaluation does not identify Boron depletion as a significant aging mechanism, but depletion analyses (thus aging management work) is still needed? This seems inconsistent.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to clarify when a time-limited aging analysis is required.

As noted in Section 3.4.2.4 and the first footnote in Table 3-4, when a boron depletion analysis is included in the original design bases, applicants must provide a time-limited aging analysis

(TLAA) to demonstrate that depletion will not challenge subcriticality in the period of extended operation. This is necessary to meet the requirements of 10 CFR 72.42(a)(1) and 72.240(c)(2).

Comment 5.3.28

Comment: *P3-59, L11 - Reference is not used in the text.*

NRC Response: The staff agrees with the comment and deleted the reference.

Comment 5.3.29

Comment: *P3-59, L26&29 - Lines should begin with "EPRI."*

NRC Response: See the response to Comment 5.3.20.

Comment 5.3.30

Comment: *P3-59, L34 - Reference is not used in the text.*

NRC Response: The staff agrees with the comment. The reference in question "Gibeling, 2000" was deleted.

Comment 5.3.31

Comment: *P3-60, L1 - Reference is not used in the text.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment. The reference in question "Hanson et al., 2012" was cited in Section 3.4.

Comment 5.3.32

Comment: *P3-60, L15 - Line should begin with "NRC."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.20.

Comment 5.3.33

Comment: *P3-62, L27&35 - There does not appear to be a "NRC 2012" reference.*

NRC Response: The staff agrees with the comment and added the reference (NRC, 2012) to Section 3.5.3.

Reference

NRC. "Three Mile Island, Unit 2, ISFSI–NRC Inspection of the Independent Spent Fuel Storage Installation, Inspection Report 07200020/2012-001." ADAMS Accession No. ML12228A457. Washington, DC: U.S. Nuclear Regulatory Commission. 2012.

Comment 5.3.34

Comment: *P3-62, L36-38 - One of the main ways to protect against freeze-thaw damage is to specify air-entrained concrete (your reference ACI 2008c). Utilities should be allowed to take credit for properly spec'd air entrained concrete if they are in a freeze-thaw zone as an alternative to conducting aging management for this degradation mode.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As noted in the gap assessments for DSSs, concrete parameters, such as the degree of air entrainment, which must be controlled to mitigate freeze-thaw damage, are well known. However, there is no protection against freeze and thaw damage. Therefore, aging management of concrete for freeze and thaw degradation in outdoor and groundwater or soil (below-grade) environments above the freeze line is required.

Comment 5.3.35

Comment: *P3-69, L35-37 - Your reference Sindelar 2011 concludes "The requisite conditions of refreshed water and the overall kinetics of the coupled processes involved in calcium leaching renders it an insignificant degradation mechanism for the concrete in the pad or cask of the DCSS for EST." This is consistent with industry observations. We believe that while it is observed, it does not require aging management. Also, this reference does not appear to be a*

“Draft Report” as stated in the reference section on page 3-83. The IAEA reference also supports this position.

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

The staff recognizes that Sindelar et al. (2011) indicate that calcium leaching from concrete is expected to be insignificant. However, as noted in Section 3.5.1.8, operating experience indicates that leaching of calcium hydroxide is a mechanism that can be exacerbated by other degradation mechanisms or designs that do not adequately prevent ingress of precipitation into the sheltered structure. As such, leaching of calcium hydroxide in concrete exposed to outdoor, sheltered, and groundwater or soil (below grade) environments is considered to be credible, and therefore, aging management is required. No changes were made to the guidance for this portion of the comment.

The reference was revised to remove the wording in question “Draft Report.” Also, the staff changed the reference “NRC, 2011a” cited on P3-69 to “Sindelar et al., 2011.”

<u>Comment 5.3.36</u>

Comment: *P3-72, L41-43 - It is for precisely this reason that this can be excluded from further consideration. ISFSI pads are constructed on pad bedding using high quality fill materials and not on polluted soils. Furthermore, there are embedded concretes elsewhere in the nuclear plant that are subjected to much more severe conditions (i.e. circulating water systems). The absence of OE for MIC on concrete in these areas further illustrates that this mechanism is not expected in this kind of service.*

NRC Response: The staff disagrees with this comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.3.

<u>Comment 5.3.37</u>

Comment: *Sec3.6.1 - Of the 11 fuel cladding degradation mechanisms, 9 were determined not to be credible during a 60 year storage timeframe. It is not clear if the studies referenced considered effects of pinhole or hairline cracks in the cladding.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance were are necessary.

As noted in Section 3.6, the spent fuel assembly components evaluated in this section include the zirconium-based cladding and fuel assembly hardware that provide structural support to

ensure that the spent fuel is maintained in a known geometric configuration. Therefore, the conditions of the assembly and cladding are reviewed for aging mechanisms and effects that may lead to a change in the analyzed fuel configuration. Pinhole and hairline cracks in the cladding, per existing review guidance, are not included in the evaluation as they are not considered to affect cladding's intended functions.

Comment 5.3.38

Comment: *P3-86, Sec3.6.1.1 - For hydride reorientation, dissolved hydrogen is associated with the high temperatures experienced during vacuum drying. Does this apply to systems that dry HBU fuel with forced helium dehydration?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance were are necessary.

The staff recognizes that the amount of dissolved hydrogen depends on the peak cladding temperature during the vacuum drying operations, which is not to exceed 400°C [752 °F] per current staff guidance. The peak cladding temperature during forced helium dehydration is expected to be lower than this limit. As such, the amount of dissolved hydrogen assumed for vacuum drying is considered bounding to forced helium dehydration.

Comment 5.3.39

Comment: *P3-86. L44 - Maximum dissolved hydrogen at 400°C is 200 ppm, but isn't this material dependent? Suggest qualifying the 200 ppm.*

NRC Response: The staff reviewed the comment and revised to guidance to clarify.

The 200 wppm hydrogen at 400°C [752°F] is based on research on Zircaloy-2 and Zircaloy-4. The staff revised the sentence in question to read, "For example, the maximum dissolved hydrogen at 400 degrees C [752 degrees F] is approximately 200 wppm for Zircaloy-2 and Zircaloy-4 based on representative solubility correlations (Kammenzind et al., 1996; Kearns et al., 1967)." The dissolved hydrogen in these alloys is considered bounding to other advanced cladding alloys (ZIRLO™, M5®) with lesser hydrogen pickup during reactor operation. The staff has also issued a draft report for public comment, NUREG–2224, "Dry Storage and Transportation of High Burnup Fuel," (ADAMS Accession No. ML18214A132) which includes additional discussion on the material dependency of hydrogen solubility.

Comment 5.3.40

Comment: *P3-87, L8 - The statement is made that cladding with hydrides "...has been shown to have reduced ductility under pinch-load stresses...." Is this still true for cladding with fuel in it? Did the references test with fuel in the cladding?*

NRC Response: The staff reviewed the comment and revised to guidance to clarify.

The staff recognizes that the referenced studies provide results for cladding from defueled rods (i.e., not accounting for the presence of the pellet). The staff has issued a draft report for public comment, NUREG–2224, "Dry Storage and Transportation of High Burnup Fuel," (ADAMS Accession No. ML18214A132) which provides this assessment and technical bases supporting the conclusion that hydride reorientation is inconsequential to the expected loads during design-basis drop accidents in storage and during seismic loading conditions. The staff has revised Section 3.6.1.1 to provide a reference to NUREG–2224 and ensured that the conclusions in both documents are consistent.

Comment 5.3.41

Comment: *P3-87, Sec3.6.1.1 - Section discusses potential cladding failures when fuel is subject to pinch-load stresses. It's not apparent when a pinch-load can occur during normal storage, or how fuel pellet-cladding bounding is considered.*

NRC Response: The staff reviewed the comment and revised to guidance to clarify.

The staff has provided additional clarification in Section 3.6.1.1 on the potential occurrence of pinch loads during dry storage operations. Pinch loads may be experienced during design-bases drop accidents (i.e., postulated drops during the removal or transfer of a canister or cask retrieval at the end of storage operations, as described in the approved design bases). Pinch loads could occur due to rod-to-grid spacer contact, rod-to-rod contact, or rod-to-basket contact during the drop accident. See the response to Comment 5.3.40 on pellet mechanical support.

Comment 5.3.42

Comment: *P3-87, L19 - Sentence beginning on line 19 - Says HRO is driven by hoop stress which is determined by peak cladding temperature. This is overly pessimistic. The hydrides precipitate and potentially reorient as the cladding cools over time. The stress at the time the hydrides precipitate will be less than the peak stress at the time of peak cladding temperature.*

NRC Response: The staff agrees with the comment and revised the guidance.

The staff revised the sentence to read, "The primary driving force for radial hydride reorientation is the cladding hoop stresses. The minimum level or threshold hoop stress for hydride reorientation depends on temperature, alloy composition, and fabrication process. A review indicates that there is no consensus in the literature on threshold hoop stresses needed to reorient hydrides for a given cladding alloy and temperature, as discussed in the following references."

Comment 5.3.43

Comment: *P3-88, L20 - Mentions the negative characteristic of RXA cladding in that it is more susceptible to HRO due to larger fraction of grain boundaries in the radial direction. The paragraph should include a counter to that in that RXA has lower hydride concentrations, so there are far fewer hydrides available to reorient.*

NRC Response: The staff agrees with the comment and revised to guidance to clarify.

The staff agrees that RXA claddings have lesser hydrogen pickup during reactor irradiation and, therefore, a lower overall concentration of hydrides. Section 3.6.1.1 has been revised, accordingly.

Comment 5.3.44

Comment: *P3-88, L24 - Discusses the impact of cooling rate on HRO and concludes the slow cooling rate under actual dry storage conditions will not inhibit HRO. However, a key phenomenon that is ignored is the effect of annealing. Due to the slow cooling rate, the cladding remains at somewhat higher temperature for long periods. This time at temperature provides some annealing and repairing of irradiation defects. This needs to be included in the cooling rate discussion in this paragraph.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The referenced studies show that the cooling rate affects the hydride precipitate size, orientation, and distribution. The synergistic effect of annealing of irradiation hardening and hydride reorientation are not well understood. The staff is unaware of a technical basis supporting the conclusion that slow cooling rates experienced post-drying and during dry storage would inhibit the precipitation of radial hydrides.

Comment 5.3.45

Comment: *P3-91, L31 - Editorial - add "to" after "expected"*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 5.3.46

Comment: *P3-98, L17 - Editorial - add a space between "350" and "degrees"*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 5.3.47

Comment: *P3-99, Sec3.6.1.9 - With regards to the statement that SCC of the cladding is not credible and aging management is not required during the 60 year storage timeframe, is this applicable to all cladding types? The evidence provided seems to focus on Zr-2 and Zr-4 versus the advanced alloys.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff recognizes that advanced zirconium alloys are expected to have a higher SCC stress threshold. Therefore, SCC of the cladding is not considered credible, and aging management is not required. In addition to the discussion on Zircaloy-2 and Zircaloy-4, some discussion on other zirconium alloys such as ZIRLO™ was included in Section 3.6.1.9.

Comment 5.3.48

Comment: *P3-99, Sec3.6.1.10 - With regards to radiation embrittlement, the conclusion that embrittlement is not considered credible is based on a cumulative fluence (10^{22} n/cm^2) not expected during storage. However, in the second paragraph states that embrittlement of cladding is observed in reactor due to cumulative fast neutron exposure on the order of 10^{22} n/cm^2. Thus, as worded it appears that the cladding has already reached the necessary fluence for embrittlement before getting into dry storage.*

NRC Response: The staff reviewed the comment and revised the guidance to clarify the basis of the conclusion.

While reactor-related fluence is known to affect cladding properties, the staff evaluates that effect in its review of initial licenses to store spent fuel. To date, the staff has concluded that the change in properties does not prevent the cladding from fulfilling its intended functions during the initial dry storage term.

NUREG–2214 evaluates whether storage beyond 20 years introduces any additional aging effects, beyond those already considered in the initial license, that must be managed to ensure that the cladding can continue to perform its intended functions. As noted in the third paragraph of Section 3.6.1.10, the cumulative neutron fluence in dry storage is expected be five orders of magnitude less than in reactor service. As such, the extended 20- to 60-year storage term does not introduce any radiation embrittlement effects that were not already considered in the initial licensing of the storage systems, and radiation embrittlement is not considered a credible mechanism that could prevent the cladding from performing its functions. Therefore, aging management is not required.

5.4 Comment from Marvin Lewis

<u>Comment 5.4.1</u>

Comment: *P3-63, Section 3.5.1.3 - Since much of this storage is in concrete overpacks, the chances of alkali-silica reactions are abundant. My concern is that alkali-silica reactions can often be mistaken for other processes and vice versa. These confounding processes include, but are not limited to*

- *Poor quality causing degradation and failures. High quality concrete is in short supply; a condition money is often used to cure. When money is short, low quality concrete flows.*

- *Degradation due to acid exposure resembles alkali-silica reaction. The cure for alkali-concrete reaction and acid exposure will degrade concrete further and faster.*

- *Poor quality handling or production can produce concrete that appears as alkali-silica problem. Cures are ineffective over a length of time.*

- *These possibilities seem to be outside this NUREG, but are not outside the scope of reality.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff recognizes that concrete durability is directly related to the quality of the concrete. Section 3.5.1.3 provides an evaluation of alkali-silica reaction (ASR)-induced degradation of concrete. As noted in Section 3.5.1.3, due to the uncertainties in screening tests that can effectively be used to eliminate the potential for ASR and previous ASR operating experience at a nuclear facility, the aging mechanism is considered credible in concrete exposed to any environment with available moisture, and therefore, aging management is required.

The example of an AMP for concrete structures in Table 6-3 relies on the licensee's corrective action program to ensure that conditions that may lead to a loss of intended function will be reviewed and dispositioned by trained personnel. If a particular aging effect is detected, part of the licensee's corrective action may include a root-cause evaluation to determine the cause of

the aging effect. As such, the aging effects caused by ASR and the confounding processes of concerns will be effectively managed during the period of extended operation.

CHAPTER 6: RESPONSES TO PUBLIC COMMENTS ON ANALYSIS OF DRY STORAGE SYSTEMS AND SPENT FUEL ASSEMBLIES

The comments on Analysis of Dry Storage Systems and Spent Fuel Assemblies relate to Chapter 4 of NUREG–2214.

6.1 Comments from NAC International

Comment 6.1.1

Comment: *P4-159, SSC "Top neutron shield bolt, vent & drain port cover bolts" - Stress Relaxation was said to be not credible per 3.2.1.10 for CS and again in 3.2.2.10 for SS. Aging Management should therefore be "No." Currently refers to bolted seal AMP.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment

As noted in Section 3.2.1.10, stress relaxation of steel bolting exposed to sheltered environments is considered to be credible, and therefore, aging management is required.

Comment 6.1.2

Comment: *P4-169, L23 - The next to last sentence should read: "The square fuel tubes in the BWR basket, including four oversized cross section fuel tubes, may include stainless-steel encased Boral® sheets on up to two sides for criticality control."*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.1.3

Comment: *P4-170, L35 - The fifth sentence, should read as follows: "The Yankee Rowe and Connecticut Yankee PWR fuel tubes are covered with stainless- steel encased Boral® sheets on all four sides for criticality control."*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.1.4</u>

Comment: *P4-170, L38 - Add new last sentence: "The LaCrosse BWR-MPC fuel tubes have Boral® sheets on up to four sides encased in stainless-steel sheets."*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.1.5</u>

Comment: *P4-176, SSC "Structural lid" - Incorrectly identifies Structural Lid as exposed to a helium environment. The Structural Lid is installed above and encases the Shield Lid. Environment between the two lids would be indoor air.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment of the lid.

The staff reviewed the NAC-UMS Safety Analysis Report (SAR) and notes that the structural lid is installed on the top of the shielding lid and then welded to the canister shell. The interior structural lid surface is in contact with the top shielding lid surface. Therefore, the interior structural lid surface is exposed to an embedded (stainless steel) environment. The staff changed the interior structural lid surface environment to embedded (stainless steel).

<u>Comment 6.1.6</u>

Comment: *P4-176, SSC "Structural lid" - The Structural Lid interior surface environment is Fully Encased, not Sheltered. The exterior Structural Lid surfaces are in a sheltered environment.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment of the lid.

See the response to Comment 6.1.5.

<u>Comment 6.1.7</u>

Comment: *P4-177, SSC "Spacer ring" - The Spacer Ring is in the airspace between the structural lid and the shield lid which are both welded in place. Therefore, the environment is not Sheltered but rather a dead airspace that is not conducive to CISCC and this component should not have an AMP assigned. Fully Encased would be a more appropriate environment.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment of the spacer ring.

The staff reviewed the NAC-UMS SAR and notes that the spacer ring is installed in place on the structural lid within the airspace between the structural lid and the shielding lid. Two sides of the spacer ring are in contact with the structural lid. As such, the spacer ring is exposed to an embedded (stainless steel) environment. The staff changed the environment to which the spacer ring is exposed to embedded (stainless steel). As noted in Sections 3.2.2.2 and 3.2.2.5, SCC and pitting and crevice corrosion of stainless steel exposed to embedded environments are not considered to be credible, and therefore, aging management is not required.

Comment 6.1.8

Comment: *P4-177, SSC "Shield lid, support ring" - The Shield Lid should have its own row with both Helium and Fully Encased environments.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the environment.

The staff reviewed the NAC-UMS SAR and notes that the shield lid is installed on the top of the canister and then welded to the canister shell. The interior shield lid surface is exposed to the internal helium environment. The top shield lid surface is in contact with the interior structural lid surface. Therefore, the top shield lid surface is exposed to an embedded (stainless steel) environment. The staff added a separate set of AMR items for the shield lid exposed to embedded (stainless steel) and helium environments.

Comment 6.1.9

Comment: *P4-177, SSC "Port cover" - The Port Cover environment is Fully Encased, not Sheltered.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment of the port covers.

The staff reviewed the NAC-UMS SAR and notes that the port covers are installed in the shield lid above the quick disconnect valve couplings and then welded to the shield lid. As such, the underside of the port covers is exposed to the internal helium environment. The top port cover surfaces are in contact with the interior structural lid surface and therefore exposed to an embedded (stainless steel) environment. The staff changed the AMR items for the port covers exposed to embedded (stainless steel) and helium environments.

Comment 6.1.10

Comment: *P4-177, SSC "Shield lid, support ring" - Leaves out Shield Lid, though shield lid support ring is identified. The Shield Lid (top) would be encased by the stainless steel Structural Lid and the interior surface exposed to the internal helium environment. Add immediately following Spacer Ring.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the environment.

See the response to Comment 6.1.8.

Comment 6.1.11

Comment: *P4-177&178, SSC "Port cover" - Port Covers are installed in the shield lid above the quick disconnect valve couplings and only underside of port covers would be potentially exposed to a helium environment. The top welded side of the Port Covers would be exposed to the encased indoor air environment.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment.

See the response to Comment 6.1.9.

Comment 6.1.12

Comment: *P4-180, SSC "Fuel basket support disk" - MAPS cite 17-4 PH as needing aging management per Section 3.2.2.8, however did not transfer it to Table 4-12, page 4-180 as on MPC Table 4-15, page 4-206.*

NRC Response: The staff agrees with the comment and has added an AMR item on thermal aging.

Comment 6.1.13

Comment: *P4-180, SSC "Fuel basket support disk" - MAPS Report proposes aging management of unwelded 17-4 PH fuel basket support disks per Section 3.2.2.8 for thermal aging in a helium environment. However, the identified threshold temperature is at the maximum calculated disk temperature for a UMS system loaded to maximum decay heat capacity of 23 kWs, and is significantly below ASME Code material specification limits.*

Additionally, the MPC 17-4 PH fuel basket support disks are significantly below the threshold temperature identified in the report. Therefore, this aging mechanism should be deleted from Table 4-15 and not included in Table 4-12.

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As noted in Section 1.1, this guidance provides a generic evaluation of the aging mechanisms that could affect the ability of SSCs to fulfill their important-to-safety functions. A renewal applicant is expected to provide a sufficient technical basis for determining the credibility of the thermal aging mechanism for its specific storage design and facility. See the response to Comment 5.3.15.

<u>Comment 6.1.14</u>

Comment: *P4-181-183, Maine Yankee related SSCs - Please use the term "Damaged Fuel Can" after Maine Yankee for those components on pages 4-181 mid page to 4-183.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.1.15</u>

Comment: *P4-181-183, Maine Yankee related SSCs - Revise nomenclature for "Maine Yankee Fuel Can" to "Maine Yankee Damaged Fuel Can (DFC)"*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.1.16</u>

Comment: *P4-185, SSC "Concrete shell" - Minor and moderate calcium leaching is a common occurrence identified during the annual VCC inspections. NAC believes a threshold is needed for an acceptance criterion, so sites are not evaluating numerous areas that are not compromising function. A reasonable threshold would be one that is beyond the minor and moderate leaching seen and offer that "Excessive leaching of calcium hydroxide" can be used as a criterion.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As noted in Table 6-3, Element 6, the acceptance criteria for visual inspections of concrete are commensurate with the 3-tier quantitative criteria in ACI 349.3R-02. This comment is similar to Comment 5.2.11.

<u>Comment 6.1.17</u>

Comment: *P4-185, SSC "Concrete shell" - Reinforcing Steel environment is air-outdoor. These components are not exposed to groundwater, so this environment should be deleted.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.1.18</u>

Comment: *P4-186, SSC "Inner shell" - Inner Shell main safety functions are as a gamma shielding component and a heat transfer component, not structural, so intended safety function should be identified as "SH, TH", or "SH, TH, SR" not "SH, SR."*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

The staff changed the intended safety function to add heat transfer as a function. However, the staff also maintained the structural function, as Section 3 of the NAC-UMS SAR states that the steel liner is a principal structural member of the vertical concrete cask.

<u>Comment 6.1.19</u>

Comment: *P4-186&187 - General comment on recommending use of the vendor terminology for components. Example: NAC believes the term Base Plate Assembly refers to both the NAC Baffle Weldment & Base Weldment components combined, Pedestal Plate refers to the NAC Base Plate and the Pedestal Cover refers to the NAC cover. If there is confusion on what components require AMPs, the Licensee might not address it during inspection.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.1.20</u>

Comment: *P4-186&187 - The VCC Inner Shell, Pedestal Plate and Base Plate Assembly have an External Surfaces Monitoring of Metallic Components AMP assigned for corrosion; however*

these are sheltered areas and will only be seen during the canister inspections. These should probably be termed as an Internal VCC Metallic Components Monitoring AMP.

NRC Response: The staff disagrees with the comment; however, the guidance was revised to clarify the scope of the AMP.

The staff believes that the proposed example AMP provides clear guidance for the management of aging of both accessible and normally inaccessible surfaces. The recommendations for these two types of surfaces significantly differ only with respect to the inspection frequency and sample size; the recommendations for the remaining AMP attributes are similar. The staff notes that the decision to provide a unified example AMP does not preclude an applicant from proposing separate AMPs in its renewal application. No changes were made to the AMP as a result of this comment. However, to avoid potential confusion regarding what is meant by an "external" surface, the AMP title has been changed to "Monitoring of Metallic Surfaces."

Comment 6.1.21

Comment: *P4-188, SSC "Base plate nelson studs" - The Base Plate Nelson Studs identify either a TLAA or AMP to address general and pitting/crevice corrosion; however, NAC doesn't see where an AMP can address this because the studs are embedded in concrete and are not accessible for inspection.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff notes that a case-specific TLAA, AMP or a supporting analysis is recommended for managing two credible corrosion mechanisms (i.e., general corrosion and pitting and crevice corrosion). A renewal applicant is expected to propose an aging management approach that is adequate to address the aging mechanisms and effects of the base plate nelson studs.

Comment 6.1.22

Comment: *P4-188, SSC "Outlet vent assembly" - Is the galvanic corrosion mechanism on the Outlet Vent Assembly intended to be with the vent screens?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff reviewed the NAC-UMS SAR and notes that galvanic corrosion may develop between the steel outlet top and the stainless steel screen tab or dowel pin.

Comment 6.1.23

Comment: *P4-188, SSC "Base plate nelson studs" - Base Plate Nelson Studs are an embedded (concrete) components and should be addressed by a Reinforced Concrete Structure AMP like the embedded steel reinforcements.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.21.

Comment 6.1.24

Comment: *P4-188&189, SSC "Inlet and outlet vent hardware" - Outlet Vent hardware is defined as a structural component, but its main intended safety function is thermal (external protection of the outlet vent from entry and blockage of foreign materials), so intended safety function should be "TH, SH", not "SR."*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

The staff reviewed the NAC-UMS SAR and notes that the outlet vent hardware of the UMS vertical concrete cask (VCC) includes screen tab and dowel pin. NAC-UMS SAR Table 2.3-1 states that the screen tab has a structural safety function and that the dowel pin has an operational function. But the dowel pin of the CY-MPC VCC has a structural safety function, as listed in NAC-MPC SAR Table 2.3-2. As such, SR is assigned for the outlet vent hardware. The staff changed the intended safety function for the inlet and outlet vent hardware to "SH, SR, TH."

Comment 6.1.25

Comment: *P4-188&189, SSC "Inlet and outlet vent hardware" - The listing does not include the Inlet Vent Hardware, so Outlet Vent Hardware should be revised to be "Inlet and Outlet Vent Hardware."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The subject line items are titled "inlet and outlet vent hardware," which is considered to adequately describe the inlet vent hardware.

Comment 6.1.26

Comment: *P4-189, SSC "Lid" - Is the galvanic corrosion mechanism on the Lid intended to be with the bolts?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff reviewed the NAC-UMS SAR and notes that galvanic corrosion may develop between the steel lid and the stainless steel lid bolt or washer.

Comment 6.1.27

Comment: *P4-191, SSC "Neutron shield (Shield plug)" - MAPS state a TLAA is needed for NS-3 Boron Depletion. NS-3 used in UMS/MPC VCC shield plugs is a concrete composite without added boron and therefore, would not require a Boron Depletion TLAA.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As stated in Table 2.1, NS-3 may contain boron to enhance neutron attenuation. If a renewal applicant does not use a version of the NS-3 shielding material that is blended with boron, then that could be provided as a basis for excluding the boron depletion aging mechanism for its specific storage designs.

Comment 6.1.28

Comment: *P4-191, SSC "Neutron shield (Shield plug)" - Change the Neutron Shield environment from embedded in steel to fully encased in steel.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff reviewed the NAC-UMS SAR and notes that the NS-4-FR and NS-3 shield materials are enclosed in the shield plug and therefore exposed to a fully encased (steel) environment. The staff changed the AMR items for the neutron shield exposed to fully encased (steel) environment.

Comment 6.1.29

Comment: *P4-191, SSC "Neutron shield (Shield plug)" - The neutron shield materials contained in the Shield Plug should be defined as "Fully Encased (FE) (Steel)" rather than "Embedded (steel)."*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 6.1.28.

Comment 6.1.30

Comment: *P4-195, SSC "Neutron shield (Cask body)" - Why is Neutron Shield (Cask Body) identified as requiring a TLAA/AMP for Thermal Aging, whereas it is not identified as required for the Steel MAGNASTOR Transfer Cask, but is for the Stainless Steel MAGNASTOR Transfer Cask? Please clarify or correct.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff notes that a case-specific TLAA, AMP or a supporting analysis is consistently recommended for managing thermal aging of polymer-based neutron shielding materials for the subject transfer casks.

Comment 6.1.31

Comment: *p4-194&195 - NAC defines the Transfer Cask body and shield door neutron shielding and gamma shielding as Fully Encased (FE) (Steel) as these components are fully encased in steel components.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff reviewed the NAC-UMS SAR and notes that both the NS-4-FR and the lead shield in the TC are enclosed within a welded steel shell and therefore exposed to a fully encased (steel) environment. The staff changed the AMR items for the neutron shield and gamma shield exposed to fully encased (steel) environment.

Comment 6.1.32

Comment: *P4-197, SSC "Shield door rails" - Shield Door Rails, which are coated steel, are identified as requiring aging management by the Transfer Cask AMP for Galvanic Corrosion for loss of material when the doors rails are not connected to other non-carbon steel materials. Please clarify or delete.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff reviewed the NAC-UMS SAR and notes that galvanic corrosion may develop between the steel shield door rail and the stainless steel shield door lock bolt. Also coatings are not necessarily a reliable method of mitigating galvanic corrosion, as the conductivity of the coating and the presence of coating holidays (which can cause local accelerated corrosive attack) must be considered.

Comment 6.1.33

Comment: *P4-200, SSC "Shell" - Section 3.2.2.9 states that radiation embrittlement is not credible, so the TLAA/AMP for the shell helium environment should be removed.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.1.34

Comment: *P4-202, SSC "Closure ring; spacer ring" - The Spacer Ring is in the airspace between the structural lid and the shield lid which are both welded in place. Therefore, the environment is not sheltered but rather a dead airspace that is not conducive to CISCC and this component should not have an AMP assigned. Fully Encased would be a more appropriate environment.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment.

The staff reviewed the NAC-MPC SAR and notes that the spacer rings for the Yankee Rowe and Connecticut Yankee configurations are installed in place on the structural lid within the airspace between the structural lid and the shielding lid. Two sides of the spacer ring are in contact with the structural lid. As such, the spacer ring is exposed to an embedded (stainless steel) environment. The staff added a separate set of AMR items for the spacer ring exposed to embedded (stainless steel). As noted in Sections 3.2.2.2 and 3.2.2.5, SCC and pitting and

crevice corrosion of stainless steel exposed to embedded environments are not considered to be credible, and therefore, aging management is not required.

Comment 6.1.35

Comment: *P4-203, SSC "Port cover" - The Port Covers are in the airspace between the structural lid and the shield lid which are both welded in place. Therefore, the environment is not sheltered but rather a dead airspace that is not conducive to CISCC and this component should not have an AMP assigned. Fully Encased would be a more appropriate environment.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment.

The staff reviewed the NAC-MPC SAR and notes that the port covers for the Yankee Rowe and Connecticut Yankee configurations are installed in the shield lid above the quick disconnect valve couplings and then welded to the shield lid. As such, the underside of the port covers is exposed to the internal helium environment. The top port cover surfaces are in contact with the interior structural lid surface and therefore exposed to an embedded (stainless steel) environment. The staff added a separate set of AMR items for the port covers exposed to embedded (stainless steel) and helium environments.

Comment 6.1.36

Comment: *P4-203, SSC "Shield lid, support ring" - Leaves out Shield Lid (for Yankee MPC and CY-MPC only), though shield lid support ring is identified. The Shield Lid (top) would be encased by the stainless steel Structural Lid and exposed to the internal helium environment. Add immediately following Closure Lid assembly spacer. The MPC-LACBWR has a closure lid/closure ring in place of the standard MPC shield lid and structural lid.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the environment.

The staff reviewed the NAC-MPC SAR and notes that the shield lid for the Yankee Rowe and Connecticut Yankee configurations is installed on the top of the canister and then welded to the canister shell. The interior shield lid surface is exposed to the internal helium environment. The top shield lid surface is in contact with the interior structural lid surface. Therefore, the top shield lid surface is exposed to an embedded (stainless steel) environment. The staff added a separate set of AMR items for the shield lid exposed to embedded (stainless steel) and helium environments.

Comment 6.1.37

Comment: *P4-202&203, SSC "Closure lid assembly spacer" - The Closure lid Spacer of the MPC-LACBWR design, which is aluminum, is identified as affected by Thermal Aging and Creep and requiring aging management, although the lid spacer's principle function is to act as a spacer to limit movement of the fuel assemblies located in the central section of the fuel basket, and has no defined structural loading conditions. However, the aluminum heat transfer disks in the MPC-LACBWR and in both the NAC-UMS, and Yankee-MPC and CY-MPC TSC designs, which are exposed to higher temperatures in the central section of the fuel basket, do not require aging management for these aging mechanisms. Please clarify and/or correct.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As noted in Sections 3.2.3.5 and 3.2.3.7, creep and thermal aging are considered to be credible for aluminum components that have a structural function. Therefore, a case-specific TLAA, AMP or a supporting analysis is recommended for managing creep and thermal aging of the aluminum closure lid assembly spacer but not for the aluminum heat transfer disks. A renewal applicant is expected to provide a sufficient technical basis to justify the credibility of the creep and thermal aging mechanisms for its specific storage designs.

Comment 6.1.38

Comment: *P4-203, SSC "Port cover" - The MPC-LACBWR Closure Lid outer redundant port cover plate is exposed to a Sheltered environment, like the Closure Ring, and the inner surface is exposed to trapped air between the two redundant port covers. The inner port cover plate is encased in steel on the outer surface and potentially helium or indoor air on the inner surface.*

NRC Response: The staff agrees with the comment and revised Table 4-15 to include AMR items for the port covers exposed to sheltered, embedded (stainless steel) and helium environments.

Comment 6.1.39

Comment: *P4-206, SSC "Fuel basket support disk" - Fuel Basket Support Disk for all MPC TSCs is 17-4 stainless steel, identical to NAC-UMS PWR support disks. However, MPC TSC table requires aging management for thermal aging for the stainless-steel support disks although it is not required for the UMS stainless-steel support disks, only the steel support disks of the BWR fuel basket assembly. Please correct to delete the aging management requirement or clarify discrepancy. (See earlier comment on Page 4-180 and Table 4-12).*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff has revised Table 4-12 to include an AMR item on thermal aging of the 17-4 stainless steel support disk that requires aging management. See also the response to Comment 6.1.13.

Comment 6.1.40

Comment: *P4-215, SSC "Inner shell" - Inner Shell main safety functions are as a gamma shielding component and a heat transfer component, not structural, so intended safety function should be identified as "SH, TH", or "SH, TH, SR", not "SR."*

NRC Response: The staff agrees with the comment and revised the intended safety function for the inner shell to "SH, SR, TH."

Comment 6.1.41

Comment: *P4-215-217 - The VCC Inner Shell, Pedestal Plate and Base Plate Assembly have an External Surfaces Monitoring of Metallic Components AMP assigned for corrosion; however, these are sheltered areas and not external, so they will be seen only during the canister inspections. Also, only the accessible surfaces of the Pedestal Plate can be viewed by remote visual versus the entire plate. These should probably be termed as an Internal VCC Metallic Components Monitoring AMP.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to clarify the scope of the AMP.

See the response to Comment 6.1.20.

Comment 6.1.42

Comment: *P4-217, SSC "Base plate nelson studs" - The Base Plate Nelson Studs have both TLAA and AMP to address general and pitting/crevice corrosion; however, NAC doesn't see where an AMP can address this because the studs are embedded in concrete and are not accessible for inspection.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.21.

Comment 6.1.43

Comment: *P4-217, SSC "Base plate nelson studs" - Base Plate Nelson Studs are embedded (concrete) components, and should be addressed by a Reinforced Concrete Structure AMP like the embedded steel reinforcements.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.21.

Comment 6.1.44

Comment: *P4-218, SSC "Inlet and outlet vent hardware" - Outlet Vent hardware is defined as a SH, SR, TH component, but its main intended safety function is thermal (external protection of the outlet vent from entry and blockage of foreign materials) and shielding of access points, so intended safety function should be "TH, SH."*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

See the response to Comment 6.1.24.

Comment 6.1.45

Comment: *P4-218, SSC "Inlet and outlet vent hardware" - Inlet Vent Hardware should have the same defined functions as the outlet vent hardware of "SH, TH", not SR.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

See the response to Comment 6.1.24.

Comment 6.1.46

Comment: *P4-219, SSC "Inlet vent supplemental shielding assembly" - Note that only the MPC-LACBWR VCC has fixed inlet vent supplemental shielding assemblies, and YR-MPC has removable inlet vent supplemental shielding assemblies. CY-MPC is not certified with inlet vent shielding assemblies.*

NRC Response: The staff agrees with the comment; however, the staff concluded that changes to the guidance are not necessary.

As stated in Section 1.1, although NUREG–2214 generically addresses SSCs for several storage designs, scoping for renewal is design and license specific. NUREG–2214 should not be used to determine whether a particular SCC is within or outside the scope of renewal (that is addressed in NUREG–1927, Revision 1).

Comment 6.1.47

Comment: *P4-219&220, SSC "Lid assembly" - The Lid Assembly (for LACBWR-MPC design) incorporates a concrete neutron shield fully encased (FE) (steel). Revise to show it as FE, not E-S. Also note that it is not feasible for Radiation Damage or Reaction with Aggregates aging effects to be managed by an External Surfaces Monitoring of Metallic Component AMP. If aging management is required, a TLAA or Shield Effectiveness management plan would be recommended. Please correct or clarify.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff reviewed the NAC-MPC SAR and notes that the concrete neutron shield of the lid assembly for LACBWR-MPC is enclosed by steel liners and therefore exposed to a fully encased (steel) environment. The staff changed the AMR items for the concrete neutron shield exposed to fully encased (steel) environment. As noted in Section 3.5.1.9, radiation damage of concrete exposed to fully encased environments are not considered to be credible, and therefore, aging management is not required. Also, as noted in Section 3.5.1.3, reaction with aggregates is considered to be credible in concrete exposed to any environment with available moisture. Because concrete fully encased in steel has limited exposure to water, this aging mechanism is not considered to be credible, and therefore, aging management is not required. The staff revised the AMR items for radiation damage and reaction with aggregates to state that no aging management is required.

Comment 6.1.48

Comment: *P4-219&220, SSC "Lid assembly" - The Lid Assembly for the MPC-LACBWR, which incorporates the function of the Shield Plug in the Yankee-MPC and CY-MPC designs, has internal Lid Nelson Studs identified. These components are embedded in concrete and fully encased in steel (FE), and potential aging effects from General Corrosion and Pitting and Crevice Corrosion are not expected in the fully encased environment. Please correct or clarify.*

NRC Response: The staff agrees with the comment and revised the guidance.

The staff reviewed the NAC-MPC SAR and notes that the lid nelson studs for LACBWR-MPC are embedded in concrete that is enclosed by steel liners and therefore exposed to an

embedded (concrete fully encased in steel) environment. The staff changed the AMR items for the nelson studs exposed to embedded (concrete fully encased in steel) environment. As noted in Sections 3.2.1.1 and 3.2.1.2, general corrosion and pitting and crevice corrosion exposed to fully encased environments are not considered to be credible, and therefore, aging management is not required. The staff revised the AMR items for general corrosion and pitting and crevice corrosion to state that no aging management is required.

Comment 6.1.49

Comment: *P4-219&220, SSC "Lid assembly" - The Lid Assembly for the MPC-LACBWR has Fully Encased Concrete like Hi-Storm 100. The MPC-LACBWR Lid Assembly is identified as requiring aging management for radiation damage. On the Hi-Storm 100, no aging management is identified for the encased concrete. Please clarify or correct.*

NRC Response: The staff reviewed the comment and revised the guidance to resolve the discrepancy.

The staff revised the AMR item for radiation damage to state that no aging management is required. See the response to Comment 6.1.47.

Comment 6.1.50

Comment: *P4-220, SSC "Lid center support, nelson studs" - The "Lid Center Support, Nelson Studs" is specific to Dairyland and not applicable to Connecticut Yankee or Yankee Rowe.*

NRC Response: The staff agrees with the comment; however, the staff concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.46.

Comment 6.1.51

Comment: *P4-221&222, SSC "Neutron shield (Shield plug)" - The Shield Plug for Yankee-MPC and CY-MPC has neutron shielding that is fully encased (FE) in steel, not embedded. Please correct.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff reviewed the NAC-MPC SAR and notes that the NS-4-FR and NS-3 shield materials are enclosed in the shield plug and therefore exposed to a fully encased (steel) environment.

The staff changed the AMR items for the neutron shield exposed to fully encased (steel) environment.

Comment 6.1.52

Comment: *P4-224, SSC "Neutron shield (Cask body)" - Why is Neutron Shield (Cask Body) identified as requiring a TLAA/AMP for Thermal Aging, whereas it is not identified as required for the Steel MAGNASTOR Transfer Cask, but is for the Stainless Steel MAGNASTOR Transfer Cask? Please clarify or correct.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.30.

Comment 6.1.53

Comment: *P4-223&224, SSC "Gamma shield (Cask body)" & "Neutron shield (Cask body)" - NAC defines the Transfer Cask body neutron shielding and lead gamma shielding as Fully Encased (FE) (Steel) as these components are fully encased in steel components. Correct "Gamma Shielding" to "Neutron Shielding" for NS-4-FR.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment.

The staff reviewed the NAC-MPC SAR and notes that both the NS-4-FR and the lead shield in the TC are enclosed within a welded steel shell and therefore exposed to a fully encased (steel) environment. The staff changed the AMR items for the neutron shield and gamma shield exposed to a fully encased (steel) environment. The staff notes that a correction of the NS-4-FR component description was not necessary, as it is defined as a neutron shield.

Comment 6.1.54

Comment: *P4-226, SSC "Shield door rails" - Shield Door Rails, which are coated steel, are identified as requiring aging management by the Transfer Cask AMP for Galvanic Corrosion for loss of material when the doors rails are not connected to other non-carbon steel materials. Please clarify or delete.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff reviewed the NAC-MPC SAR and notes that galvanic corrosion may develop between the steel shield door rail and the stainless steel shield door lock bolt.

<u>**Comment 6.1.55**</u>

Comment: *P4-229, SSC “Closure lid” - Intended Safety Function should also list Confinement (CO) in addition to SR.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>**Comment 6.1.56**</u>

Comment: *P4-230, SSC “Closure ring” - Intended Safety Function should also list Confinement (CO) in addition to SR.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>**Comment 6.1.57**</u>

Comment: *P4-239, SSC “Damaged fuel can screens” - Under Damaged Fuel Can Screen also add Wiper to the list as the wiper extends out further than the screens and acts as the CO boundary between the DFC lid and DFC collar.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>**Comment 6.1.58**</u>

Comment: *P4-246, SSC “Lid assembly, Concrete” - Radiation Damage (both cracking and loss of strength) refers to TLAA or AMP but 3.5.1.9 says radiation damage to concrete is not credible. Aging management should therefore be “No.”*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff reviewed the NAC-MAGNASTOR SAR and notes that the concrete of the lid assembly is enclosed by steel liners and therefore exposed to a fully encased (steel) environment. The staff revised the AMR items for the concrete cask lid exposed to fully encased (steel)

environment. As noted in Section 3.5.1.9, radiation damage of concrete exposed to fully encased environments are not considered to be credible, and therefore, aging management is not required. Also, as noted in Section 3.5.1.3, reaction with aggregates is considered to be credible in concrete exposed to any environment with available moisture. Because concrete fully encased in steel has limited exposure to water, this aging mechanism is not considered to be credible, and therefore, aging management is not required. This comment is similar to Comment 6.1.47.

Comment 6.1.59

Comment: *P4-250&251, SSC "Inner shell" - Inner Shell main safety functions are as a gamma shielding component and a heat transfer component, not structural, so intended safety function should be identified as "SH, TH", not "SR, SH, TH."*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

The staff reviewed the NAC-MAGNASTOR SAR and notes that the structural design description of the transfer cask includes the inner shell as one of the principal structural components, so this remained as an intended safety function in the AMR table. As recommended by the comment, the staff also added the shielding and heat transfer functions, as the SAR thermal and shielding analyses include the contributions of the inner shell.

Comment 6.1.60

Comment: *P2-245-247, SSC "Lid assembly" - The Lid Assembly incorporates a concrete neutron shield embedded in concrete. It is not feasible for Reaction with Aggregates aging effects to be managed by an External Monitoring of Metallic Component AMP. If aging management is required, a TLAA would be required. Please correct or clarify. Also, the environment should be Fully Encased (FE) (steel).*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff reviewed the NAC-MAGNASTOR SAR and notes that the concrete neutron shield of the lid assembly is enclosed by steel liners and therefore exposed to a fully encased (steel) environment. The staff revised the AMR items for the concrete neutron shield exposed to fully encased (steel) environment. As noted in Section 3.5.1.9, radiation damage of concrete exposed to fully encased environments are not considered to be credible, and therefore, aging management is not required. Also, as noted in Section 3.5.1.3, reaction with aggregates is considered to be credible in concrete exposed to any environment with available moisture. Because concrete fully encased in steel has limited exposure to water, this aging mechanism is not considered to be credible, and therefore, aging management is not required. This comment is similar to Comment 6.1.47.

Comment 6.1.61

Comment: *P4-247-248, SSC "Lid anchor (standard and alternate configurations)" - The Lid Anchor (standard and alternate configuration) steel components that are embedded in concrete cannot be monitored for aging management for General Corrosion or Pitting and Crevice Corrosion by an External Surfaces Monitoring of Metallic Component AMP. If aging management is required, a TLAA or the Reinforced Concrete Structures AMP would be appropriate. Please correct or clarify.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff notes that a case-specific TLAA, AMP or a supporting analysis is recommended for managing the two credible corrosion mechanisms (i.e., general corrosion and pitting and crevice corrosion) in an embedded concrete environment. A renewal applicant is expected to propose an aging management approach that is adequate to address the aging mechanisms and effects of the lid anchor.

Comment 6.1.62

Comment: *P4-293, SSC "Reinforced concrete: ISFSI pad" - The proposed AMP to look for cracking and loss of strength by Aggressive Chemical Attack under the Groundwater/Soil environment implies that the site would be looking at the pad concrete below ground level. The site will be taking this to mean that this will be an opportunistic inspection in the case of other work that exposes the pad below ground level, especially considering the site will be taking groundwater samples to monitor for an aggressive environment.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As stated in the Reinforced Concrete Structures AMP in Table 6-3, visual inspections of below-grade (underground) areas are recommended to be opportunistic; inspections are performed when excavations occur for any reason.

6.2 Comments from Nuclear Energy Institute

Comment 6.2.1

Comment: *P4-1, Table 4-1 - Add NUHOMS HD System to the evaluated Storage system design; NRC Docket Number 72-1030.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 3.2.2.

Comment 6.2.2

Comment: *P4-6, Sec4.2.3 - HSM described, but heat shield not mentioned. It is part of the HSM, and is shown on page 4-41.*

NRC Response: The staff agrees with the comment and revised the guidance to better describe the system.

The staff added a sentence to the second paragraph of Section 4.2.3 to read, "Heat shields are placed above and to either side of the DSC to protect the concrete surfaces of the storage module from thermal radiation effects."

Comment 6.2.3

Comment: *P4-10, Table 4-2 - Listed in this Table are the AMR results of the subcomponents of all the NUHOMS DSCs certified in CoC 1004. However, the Report does not list the DSC Type associated with a specific subcomponent. For example, the first and second row lists the AMR results for Guide Sleeves and Over sleeves. It would be useful to add in the first Column "Spacer Disc Type DSC Basket" instead of "DSC Basket." The Standardized NUHOMS System provides for 2 alternate basket designs "spacer disc" and "Tube" type basket design. Further, the next line item lists the AMR results for Aluminum Plate. This subcomponent is only present in high heat load DSCs and not present in the earlier DSC designs. Hence, here also, it would add to the clarity of this Table if it is annotated that this subcomponent is only present in high heat load DSC designs. This is a generic change suggested for the entire Table 4-2.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As noted in Section 1.1, this guidance is not intended to be a scoping document for each storage system, but rather to provide a generic evaluation of the aging mechanisms relevant to the material-environment combinations present in dry storage.

Comment 6.2.4

Comment: *P4-33, Table 4-4 - Listed in this Table are the AMR results of the subcomponents of all the NUHOMS HSMs certified in CoC 1004. However, the report does not list the HSM Type associated with a specific subcomponent. For example, the DSC Support Structure for HSM*

Model 80 is quite different from that provided for HSM-H or HSM- HS. Hence, it would be useful to annotate each row of Table 4-4 by adding the HSM Type in Column 1.

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.2.3.

Comment 6.2.5

Comment: *P4-69, Table 4-6 - Listed in this Table are the AMR results of the subcomponents of all the NUHOMS Transfer Casks. The Standardized NUHOMS System provides for 5 different TC Types: Standardized TC, OS197, OS197H, OS200, and OS197L. Each of these TCs has a unique design with materials of construction which differ from one TC Type to another. To avoid any confusion, it would be useful to annotate each row of Table 4-6 by adding the TC Type in Column 1.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.2.3.

Comment 6.2.6

Comment: *Sec4.3 - This section is limited to the HI-STORM 100 and its associated variants. Other Holtec dry cask storage systems, that have separate Certificates (HI-STORM FW and HI-STORM UMAX) have been placed into service and utilize larger capacity canisters.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As noted in Section 1.2, the guidance evaluates selected storage system designs to address near-term renewal applications, which do not include the HI-STORM FW and HI-STORM UMAX systems. No changes were made to the guidance in response to the comment.

Comment 6.2.7

Comment: *P4-91, L43 - Fix typo (“HI-TORM 100”)*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.8

Comment: *P4-94, L9 - Recommend changing, "Boral and METAMIC" to "Either Boral or METAMIC" to make it clear that a given basket is comprised of only one type of neutron absorber.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.9

Comment: *P4-94, L11 - Recommend noting that the pocket enclosure for neutron absorbers does not apply to METAMIC-HT (MPC-68M) configuration.*

NRC Response: The staff agrees with the comment and revised the sentence in question to clarify that the pocket enclosures do not pertain to the Metamic-HT basket.

Comment 6.2.10

Comment: *P4-94, L23-24 - The bolted on shield lid atop the MPC lid is not a standard configuration. In fact it was used one time at one site. MAPS should state this is not present on all MPCs. There is no further consideration for this shield lid in Table 4-7, or how if present can affect aging management of the MPC lid.*

NRC Response: The staff agrees with the comment and deleted the discussion of the subject configuration.

The staff reviewed the HI-STORM 100 SAR Table 2.2.6 and notes that the shield lid is not included in the table and therefore is not within the scope of renewal. The sentence in question "A shield lid is bolted to the top of the MPC lid and provides radiation shielding." was deleted.

Comment 6.2.11

Comment: *P4-99, Sec4.3.5 – It's not apparent why the Transfer Cask is being included for a document that is focused on extended storage.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The certificates of compliance for the storage systems described in the MAPS Report include transfer casks as an SSC important to safety. 10 CFR 72.240 requires TLAAs and AMPs to address aging issues associated with SSCs important to safety. No changes were made to the guidance in response to the comment.

Comment 6.2.12

Comment: *P4-104, SSCs "BWR fuel basket" and "Neutron absorber" - Boron depletion (under subsections for Metamic-HT, Boral and Metamic) is not an apparent aging mechanism for dry storage as there is practically no thermal neutron flux under the conditions for normal storage. Stating that a TLAA may be required should not be necessary.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

As noted in Section 5.1, per 10 CFR 72.42(a)(1) and 72.240(c)(2), renewal applicants are required to reevaluate all aging-related calculations or analyses involving time limited assumptions that were contained in the original design bases. Therefore, when a boron depletion analysis is included in the design bases, applicants must provide a TLAA to demonstrate that depletion will not challenge subcriticality in the period of extended operation. See also the response to Comment 5.3.27.

Comment 6.2.13

Comment: *P4-110, SSC "Concrete shield: radial shield, shield block, pedestal shield, lid shield" - With regards to the items combined under "concrete shield", for those items that are fully encased in steel, including the reaction with aggregates as an aging effect should be a 'no', since these do not have the susceptibility mechanisms (for example large aggregate surface area for reaction or exposure to moisture) described in Section 3.5.1.3. Likewise these components do not relay on the concrete for structural strength.*

NRC Response: The staff agrees with the comment and made the recommended change.

As noted in Section 3.5.1.3, reaction with aggregates is considered to be credible in concrete exposed to any environment with available moisture. Because concrete fully encased in steel has limited exposure to water, this aging mechanism is not considered to be credible, and therefore, aging management is not required. The AMR items for the concrete shield were revised to state that no aging management is required. This comment is similar to Comment 6.1.47.

Comment 6.2.14

Comment: *P4-111, SSC "Gamma shield cross plates" - Should aging management of external vent screens be included with gamma shields, or can the condition of the vent screens be used as an indication of gamma shield aging?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff reviewed the HI-STORM 100 SAR Table 2.2.6 and notes that the external vent screens have an operational function and their safety classification is not important to safety (NITS). The staff considers the use of vent screen inspections as an indication of gamma shielding condition to be most appropriately evaluated on a case-by-case basis (rather than in the generic guidance of NUREG-2214).

Comment 6.2.15

Comment: *P4-132, SSC "Neutron shield" - In the HI-STAR, the safety function of the neutron shield is only shielding. Thus, aging effects should not include structural related items (fracture toughness and loss of ductility). These aging effects refer to MAPS Section 3.3.1.2 which does not describe either of these. Additionally, these effects are covered under thermal aging, which describe the breakdown of polymers under elevated temperatures. It would seem that over an extended storage period, elevated temperatures should not be a concern.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to clarify the relevant aging effects.

As noted in Section 2.4, loss of fracture toughness and loss of ductility may result from thermal aging embrittlement. As also noted in Section 3.3.1.2, polymer-based neutron-shielding materials, such as Holtite-A, can undergo molecular scission and cross linking when exposed to elevated temperatures, causing embrittlement, shrinkage, decomposition, and changes in physical configuration. The staff revised the fourth sentence of Section 3.3.1.2 to read, "These reactions may cause embrittlement (e.g., loss of fracture toughness or ductility), shrinkage, decomposition, and changes in physical configuration (e.g., loss of hydrogen or water) (EPRI, 2002; McManus and Chamis, 1996)."

Comment 6.2.16

Comment: *P4-140, SSC "Radial lead shield" - If aging management of the transfer cask is deemed necessary, then should lead shield material gapping or slumping be included as an aging mechanism?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

Lead slumping is evaluated in the staff's review of initial storage licenses and certificates of compliance, and this phenomenon is associated with the potential for plastic deformation and redistribution of lead shielding during an impact. The staff does not consider lead slumping to be a potential aging effect during the 20- to 60-year period of extended operation.

Comment 6.2.17

Comment: *P4-142, Table 4-10 - Should pool lid seals be included with the other related HI-TRAC components considered for aging effects?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff reviewed the HI-STORM 100 SAR Table 2.2.6 and notes that the pool lid gasket has an operational function and its safety classification is NITS.

Comment 6.2.18

Comment: *Section 4.3 - A note should be added to the section stating that not all HI-STORM systems include all the SSCs provided in Tables 4-7 through 4-10.*

NRC Response: The staff agrees with the comment and revised the Introduction to Chapter 4 to reinforce that, as noted in Section 1.1, the MAPS Report does not address the scoping of SSCs for specific-license or CoC renewal, and applicants should perform the scoping review of their systems in accordance with Chapter 2 of NUREG–1927, Revision 1.

Comment 6.2.19

Comment: *P4-176, SSC "Structural lid" - Incorrectly identifies Structural Lid as exposed to a helium environment. The Structural Lid is installed above and encases the Shield Lid. Environment between the two lids would be indoor air.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment of the lid.

See the response to Comment 6.1.5.

<u>Comment 6.2.20</u>

Comment: *P4-177, SSC "Shield lid, support ring" - Leaves out Shield Lid, though shield lid support ring is identified. The Shield Lid (top) would be encased by the stainless steel Structural Lid and the interior surface exposed to the internal helium environment. Add immediately following Spacer Ring.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the environment.

See the responses to Comments 6.1.8 and 6.1.10.

<u>Comment 6.2.21</u>

Comment: *P4-177&178, SSC "Port cover" - Port Covers are installed in the shield lid above the quick disconnect valve couplings and only underside of port covers would be potentially exposed to a helium environment. The top welded side of the Port Covers would be exposed to the encased indoor air environment.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment of the port covers.

See the responses to Comments 6.1.9 and 6.1.11.

<u>Comment 6.2.22</u>

Comment: *P4-185, SSC "Concrete shell" - Reinforcing Steel environment is air-outdoor. These components are not exposed to groundwater, so this environment should be deleted.*

NRC Response: The staff agrees with the comment and made the recommended change.

See the responses to Comment 6.1.17.

<u>Comment 6.2.23</u>

Comment: *P4-186, SSC "Inner shell" - Inner Shell main safety functions are as a gamma shielding component and a heat transfer component, not structural, so intended safety function should be identified as "SH, TH", or "SH, TH, SR" not "SH, SR."*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

See the response to Comment 6.1.18.

Comment 6.2.24

Comment: *P4-188&189, SSC "Inlet and outlet vent hardware" - Outlet Vent hardware is defined as a structural component, but its main intended safety function is thermal (external protection of the outlet vent from entry and blockage of foreign materials)*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

See the response to Comment 6.1.24.

Comment 6.2.25

Comment: *P4-188&189, SSC "Inlet and outlet vent hardware" - The listing does not include the Inlet Vent Hardware, so Outlet Vent Hardware should be revised to be "Inlet and Outlet Vent Hardware."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 6.1.25.

Comment 6.2.26

Comment: *P4-188&189, SSC "Inlet and outlet vent hardware" - Outlet Vent hardware is defined as a structural component, but its main intended safety function is thermal (external protection of the outlet vent from entry and blockage of foreign materials), so intended safety function should be "TH, SH", not "SR."*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

See the response to Comment 6.1.24.

<u>Comment 6.2.27</u>

Comment: *P4-191, SSC "Neutron shield (Shield plug)" - The neutron shield materials contained in the Shield Plug should be defined as "Fully Encased (FE) (Steel)" rather than "Embedded (steel)."*

NRC Response: The staff agrees with the comment and made the recommended change.

See the responses to Comments 6.1.28 and 6.1.29.

<u>Comment 6.2.28</u>

Comment: *P4-195, SSC "Neutron shield (Cask body)" - Why is Neutron Shield (Cask Body) identified as requiring a TLAA/AMP for Thermal Aging, whereas it is not identified as required for the Steel MAGNASTOR Transfer Cask, but is for the Stainless Steel MAGNASTOR Transfer Cask? Please clarify or correct.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.30.

<u>Comment 6.2.29</u>

Comment: *P4-194&195 - NAC defines the Transfer Cask body and shield door neutron shielding and gamma shielding as Fully Encased (FE) (Steel) as these components are fully encased in steel components.*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 6.1.31.

<u>Comment 6.2.30</u>

Comment: *P4-197, SSC "Shield door rails" - Shield Door Rails, which are coated steel, are identified as requiring aging management by the Transfer Cask AMP for Galvanic Corrosion for loss of material when the doors rails are not connected to other non-carbon steel materials. Please clarify or delete.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.32.

Comment 6.2.31

Comment: *P4-206, SSC "Fuel basket support disk" - Fuel Basket Support Disk for all MPC TSCs is 17-4 stainless steel, identical to NAC-UMS PWR support disks. However, MPC TSC table requires aging management for thermal aging for the stainless-steel support disks although it is not required for the UMS stainless-steel support disks, only the steel support disks of the BWR fuel basket assembly. Please correct to delete the aging management requirement or clarify discrepancy.*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 6.1.39.

Comment 6.2.32

Comment: *P4-215, SSC "Inner shell" - Inner Shell main safety functions are as a gamma shielding component and a heat transfer component, not structural, so intended safety function should be identified as "SH, TH", or "SH, TH, SR", not "SR."*

NRC Response: The staff agrees with the comment and revised the intended safety function for the inner shell to "SH, SR, TH."

See the response to Comment 6.1.40.

Comment 6.2.33

Comment: *P4-220, SSC "Lid assembly, Concrete" - Radiation Damage (both cracking and loss of strength) refers to TLAA or AMP but 3.5.1.9 says radiation damage to concrete is not credible. Aging management should therefore be "No."*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 6.1.58.

Comment 6.2.34

Comment: *P4-224, SSC "Neutron shield (Cask body)" - Why is Neutron Shield (Cask Body) identified as requiring a TLAA/AMP for Thermal Aging, whereas it is not identified as required for the Steel MAGNASTOR Transfer Cask, but is for the Stainless Steel MAGNASTOR Transfer Cask? Please clarify or correct.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the responses to Comments 6.1.30 and 6.1.52.

Comment 6.2.35

Comment: *P4-223-224, SSC "Gamma shield (Cask body)" & "Neutron shield (Cask body)" - NAC defines the Transfer Cask body neutron shielding and lead gamma shielding as Fully Encased (FE) (Steel) as these components are fully encased in steel components. Correct "Gamma Shielding" to "Neutron Shielding" for NS-4-FR materials.*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the environment.

See the response to Comment 6.1.53.

Comment 6.2.36

Comment: *P4-226, SSC "Shield door rails" - Shield Door Rails, which are coated steel, are identified as requiring aging management by the Transfer Cask AMP for Galvanic Corrosion for loss of material when the doors rails are not connected to other non-carbon steel materials. Please clarify or delete.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.54.

Comment 6.2.37

Comment: *P4-239, SSC "Damaged fuel can screens" - Under Damaged Fuel Can Screen also add Wiper to the list as the wiper extends out further than the screens and acts as the CO boundary between the DFC lid and DFC collar.*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 6.1.57.

Comment 6.2.38

Comment: *P4-246, SSC "Lid assembly, Concrete" - Radiation Damage (both cracking and loss of strength) refers to TLAA or AMP but 3.5.1.9 says radiation damage to concrete is not credible. Aging management should therefore be "No."*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 6.1.58.

Comment 6.2.39

Comment: *P2-245-247, SSC "Lid assembly" - The Lid Assembly incorporates a concrete neutron shield embedded in concrete. It is not feasible for Reaction with Aggregates aging effects to be managed by an External Monitoring of Metallic Component AMP. If aging management is required, a TLAA would be required. Please correct or clarify. Also, the environment should be Fully Encased (FE) (steel).*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 6.1.60.

Comment 6.2.40

Comment: *P4-247&248, SSC "Lid anchor (standard and alternate configurations)" - The Lid Anchor (standard and alternate configuration) steel components that are embedded in concrete cannot be monitored for aging management for General Corrosion or Pitting and Crevice Corrosion by an External Surfaces Monitoring of Metallic Component AMP. If aging management is required, a TLAA or the Reinforced Concrete Structures AMP would be appropriate. Please correct or clarify.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.1.61.

Comment 6.2.41

Comment: *P4-250&251, SSC "Inner shell" - Inner Shell main safety functions are as a gamma shielding component and a heat transfer component, not structural, so intended safety function should be identified as "SH,TH", not "SR,SH,TH."*

NRC Response: The staff agrees, in part, with the comment and revised the guidance to clarify the intended safety functions.

See the response to Comment 6.1.59.

Comment 6.2.42

Comment: *P4-259, L41 - Replace "of the" with "adjacent."*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.43

Comment: *P4-292, SSC "Reinforced concrete: ISFSI pad" - Differential settlement is not an aging mechanism associated with Air-Outdoor environment.*

NRC Response: The staff agrees with the comment and has revised the guidance to associate differential settlement only with concrete structures in contact with "groundwater/soil" environments.

See the response to Comment 5.3.1.

Comment 6.2.44

Comment: *P4-297, Sec4.8 - From a fundamental standpoint, should there even be an AMP for fuel? It is the fuel we are trying to protect in the dry storage system. Rather than focusing on degradation of the fuel, instead, we should ensure that any impacts from accidents and*

aging of the dry storage system do not create an impact on the fuel that would create unacceptable consequences.

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

As noted in Section 3.6, spent fuel assembly components (zirconium-based cladding and fuel assembly hardware) provide structural support to ensure that the spent fuel is maintained in a known geometric configuration. Although the spent fuel assembly is not an SSC of the ISFSI or DSS, depending on the particular design bases, the spent fuel must remain in its analyzed configuration during the period of extended operation for continuation of the approved design bases. Therefore, for these ISFSIs and DSSs, the conditions of the spent nuclear fuel (SNF) assembly and cladding are within the scope of renewal and are reviewed for aging mechanisms and effects that may lead to a change in the analyzed fuel configuration.

Comment 6.2.45

Comment: *P4-297, Sec4.8 - How does this integrate with the revised definition of RETRIEVABILITY which allows for canister-based retrievability to meet this requirement?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The specific storage systems and amendments evaluated in NUREG–2214 (see Table 4-1) do not define retrievability on a canister basis. Regardless of how retrievability is defined, a renewal applicant must propose a scoping and aging management approach that is applicable to its design bases. Section 2.4.2.1 of NUREG–1927, Revision 1, provides guidance that an applicant may consider when defining the functions of its fuel assemblies. The staff notes that many of the fuel assembly AMR items in Table 4-25 are not associated with retrievability, but rather with other roles the fuel assembly subcomponents have in supporting the safety analyses of the DSS or ISFSI (e.g., maintain fuel configuration for subcriticality). NUREG–1927 also clarifies that, if a license or CoC holder wishes to revise the safety analyses for the approved design-bases fuel configuration, then it should pursue such a change through an amendment or revision request, and not as part of a renewal application.

Comment 6.2.46

Comment: *P4-297, L18 - Line reads "...neutron absorber rods and burnable poison rods..." These are the same thing. Maybe meant to say "control rods and burnable poison rods."*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the component description.

The staff revised the sentence in question to read, “They also provide channels for insertion of a rod cluster control assembly, a neutron source assembly, a burnable poison assembly, or a thimble plug assembly, depending on the position of the particular fuel assembly in the core.”

Comment 6.2.47

Comment: *P4-297, L29 - Line should read as follows: “...rods. The fuel rods are hollow cladding tubes fabricated from Zircaloy-2 filled with uranium dioxide pellets.”*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.48

Comment: *P4-297, L34 - Line should read as follows: “...channels). The channels...” Deleted text is repeated in previous paragraph.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.49

Comment: *P4-297, L36 - Line should read as follows: “...levels. Both the upper and lower tie plates…”*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.50

Comment: *P4-300, SSC “Fuel rod cladding” - For Fuel rod cladding, Hydride-induced embrittlement Aging Mechanism, page 3-89, line 35 requires an AMP.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the need for an AMP.

As stated in Section 3.6.1.1, to address hydride reorientation, the guidance recommends the evaluation of field-obtained evidence via the example AMP or the performance of a defense-in-depth analysis that assumes fuel reconfiguration.

Although hydride reorientation and hydride-induced embrittlement are two interchangeable terms, the staff changed “hydride-induced embrittlement” in Table 4-25 to “hydride reorientation” for consistency.

<u>**Comment 6.2.51**</u>

Comment: *P4-300, SSC “Fuel rod cladding” - Hydride-induced embrittlement should be “Hydride reorientation” per the title of Section 3.6.1.1. Also, Table 4-25 says no aging management, but Section 3.6.1.1 provides 2 approaches for aging management to address hydride reorientation: A defense in depth with consequence analysis or using results from a demo.*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 6.2.50.

<u>**Comment 6.2.52**</u>

Comment: *P4-300, SSC “Guide tubes (PWR) or water channels (BWR)” - For Guide tubes (PWR) or water channels (BWR), the reference for Radiation embrittlement should be 3.6.2.5 not 3.6.1.10.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>**Comment 6.2.53**</u>

Comment: *P4-300, SSC “Guide tubes (PWR) or water channels (BWR)” - For Guide tubes (PWR) or water channels (BWR), the reference for Fatigue should be 3.6.2.6 not 3.6.1.11.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.54

Comment: *P4-301, SSC "Spacer grids, Zirconium-based alloy" - For Spacer grids, Zirconium-based alloy, the reference for Radiation embrittlement should be 3.6.2.5 not 3.6.1.10.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.55

Comment: *P4-301, SSC "Spacer grids, Zirconium-based alloy" - For Spacer grids, Zirconium-based alloy, the reference for Fatigue should be 3.6.2.6 not 3.6.1.11.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.56

Comment: *P4-301, SSC "Spacer grids, Inconel" - For Spacer grids, Inconel, the reference for Radiation embrittlement should be 3.6.2.5 not 3.6.1.10.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.57

Comment: *P4-301, SSC "Spacer grids, Inconel" - For Spacer grids, Inconel, the reference for Fatigue should be 3.6.2.6 not 3.6.1.11.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.2.58

Comment: *P4-301, SSC "Lower and upper end fittings, Stainless steel" - For Lower and upper end fittings, Stainless steel, the reference for Radiation embrittlement should be 3.6.2.5 not 3.6.1.10.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.2.59</u>

Comment: *P4-301, SSC "Lower and upper end fittings, Inconel" - For Lower and upper end fittings, Inconel, the reference for Fatigue should be 3.6.2.6 not 3.6.1.11.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.2.60</u>

Comment: *P4-302, SSC "Fuel channel (BWR)" - For Fuel channel (BWR), the reference for Radiation embrittlement should be 3.6.2.5 not 3.6.1.10.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.2.61</u>

Comment: *P4-302, SSC "Fuel channel (BWR)" - For Fuel channel (BWR), the reference for Fatigue should be 3.6.2.6 not 3.6.1.11.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.2.62</u>

Comment: *P4-302, SSC "Poison rod assemblies (PWR)" - For Poison rod assemblies (PWR), the reference for Radiation embrittlement should be 3.6.2.5 not 3.6.1.10.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.2.63</u>

Comment: *P4-302, SSC "Poison rod assemblies (PWR)" - For Poison rod assemblies (PWR), the reference for Fatigue should be 3.6.2.6 not 3.6.1.11.*

NRC Response: The staff agrees with the comment and made the recommended change.

<u>Comment 6.2.64</u>

Comment: *P4-303, L6&8 - Lines should begin with "FuelSolutions."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.20.

<u>Comment 6.2.65</u>

Comment: *P4-303, L16 - Line should begin with "Holtec International."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.20.

<u>Comment 6.2.66</u>

Comment: *P4-303, L21,22&24 - Lines should begin with "NAC International."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.20.

<u>Comment 6.2.67</u>

Comment: *P4-303, L30&32 - Lines should begin with "NRC."*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.20.

6.3 Comments from Holtec International

<u>Comment 6.3.1</u>

C*omment:* *P4-94, L28&29 - Suggest also noting that the enclosed helium also provides an inert environment that makes corrosion issues non-credible.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff's evaluations of corrosion aging mechanisms to determine whether they are credible or noncredible are provided in Chapter 3 of the guidance.

<u>Comment 6.3.2</u>

Comment: *P4-99, L20 - The statement says that the pool lid incorporates two gasket seals, however, the remainder of the sentence correctly notes that those seals are actually one between the pool lid and bottom flange and a second between the MPC and transfer cask, near the top. Suggest removing the "pool lid incorporates two gasket seals," statement.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff reviewed the HI-STORM 100 SAR and notes that only the seal between the pool lid and the bottom flange is designed to hold clean demineralized water in the HI-TRAC inner cavity. The staff revised the third and fourth sentences of this paragraph to read, "In addition to providing shielding in the axial direction, the pool lid incorporates a seal between the pool lid and the bottom flange that is designed to hold clean demineralized water in the HI-TRAC inner cavity. The seal provides a barrier from contamination of the exterior of the MPC by the spent fuel pool water."

<u>Comment 6.3.3</u>

Comment: *P4-106, SSC "Basket supports" - The table does not include the "Basket Shim" component (used only in the MPC-68M), which have a thermal function. However, they are stored in the same inert environment, so no aging management is expected to be needed.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff reviewed the HI-STORM 100 SAR Table 2.2.6 and notes that the safety classification of basket shims is NITS.

Comment 6.3.4

Comment: *P4-108, SSC "Vent and drain tubes" - The table identifies these as having a structural function, but these components are not credited with any structural function. These components are only used during initial loading and drying, and are not necessary for long term storage.*

NRC Response: The staff agrees with the comment and removed the "Vent and drain tubes" AMR items from Table 4-7.

The staff reviewed the HI-STORM 100 SAR Table 2.2.6 and notes that these components have an operational function.

Comment 6.3.5

Comment: *P4-109, SSC "Damaged fuel container" - The table identifies that the Damaged Fuel Container has a confinement "CO" function. However, the DFCs have screens at the top and bottom and are not credited as a confinement boundary. It may be more appropriate to add a criticality function to the DFCs, as they are credited in that analysis.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.6

Comment: *P4-109, SSC "Threaded disc, plug adjustment" - These components are not credited as part of the confinement boundary for the MPC.*

NRC Response: The staff agrees with the comment and removed the "Threaded disc, plug adjustment" AMR items from Table 4-7.

The staff reviewed the HI-STORM 100 SAR Table 2.2.6 and notes that these components have an operational function.

Comment 6.3.7

Comment: *P4-111, SSC “Lid inner ring” - The lid inner ring is also credited for shielding.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.8

Comment: *P4-111, SSC “Lid outer ring” - The lid outer ring also has a structural function.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.9

Comment: *P4-112, SSC “Outer shell” - The outer shell is also credited for shielding.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.10

Comment: *P4-113, SSC “Inner shell, lid bottom plate, and lid shell” - These components are also credited for shielding.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.11

Comment: *P4-115, SSC “Lid stud” - The material for the lid stud appears to be incomplete, see the HI-STORM 100 FSAR Table 2.2.6.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

Table 4-8 includes the lid stud component made of either steel or stainless steel, which is consistent with the materials listed in the HI-STORM 100 SAR Table 2.2.6 for this component.

Comment 6.3.12

Comment: *P4-119, SSC "Closure lid steel (HI-STORM 100U)" - This component is also credited in the structural analysis for missile protection.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.13

Comment: *P4-129, SSC "Inner shell" - The inner shell also provides a structural function, and a thermal function due to heat conduction.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.14

Comment: *P4-132, SSC "Intermediate Shells" - The intermediate shells also provide a thermal function.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.15

Comment: *P4-132, SSC "Removable shear ring" - The removable shield ring does not have a shielding function; its primary safety function is structural.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.16

Comment: *P4-133, SSC "Pocket trunnion" - The main intended safety function of the pocket trunnions is structural.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.17

Comment: *P4-134, SSC "Lifting trunnion" - The lifting trunnions are also credited in shielding.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.18

Comment: *P4-135, SSC "Enclosure shell panels and enclosure shell return" - These components also provide a thermal and shielding purpose.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.19

Comment: *P4-144, SSC "Top lid shielding" - This component does not have a thermal function.*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 6.3.20

Comment: *P4-144, SSC "Fill port plugs" - This component does not have a shielding function.*

NRC Response: The staff agrees with the comment and made the recommended change.

CHAPTER 7: RESPONSES TO PUBLIC COMMENTS ON TIME-LIMITED AGING ANALYSES

The comment on Time-Limited Aging Analyses relates to Chapter 5 to NUREG–2214.

7.1 Comment from Nuclear Energy Institute

Comment 7.1.1

Comment: *P5-1, L4 - After "design basis," insert "described in, or incorporated by reference in the ISFSI or cask FSAR."*

NRC Response: The staff disagrees with the comment and did not make any changes in response to the comment.

The staff notes that the text in the draft document reflects the definition of TLAAs in 10 CFR 72.3, and the proposed change may be inappropriately restrictive. The staff recognizes that the design bases could include analyses not mentioned in a final safety analysis report (FSAR), such as those that may be included in a license condition, exemption, or technical specification.

CHAPTER 8: RESPONSES TO PUBLIC COMMENTS ON EXAMPLE AGING MANAGEMENT PROGRAMS

The comments on Example Aging Management Programs relate to Chapter 6 of NUREG–2214.

8.1 Comments from NAC International

Comment 8.1.1

Comment: *P6-1, Table 6-1 - Suggest breaking Section 6.7 into two separate AMPS. One for readily accessible metallic external surfaces exposed to outdoor atmospheres and one for sheltered, internal metallic surfaces available for inspection during remote camera inspections of the canister. There will be different inspection programs repairs inspection frequencies, etc. for these two cases.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to clarify the scope of the subject AMP.

See the response to Comment 6.1.20.

Comment 8.1.2

Comment: *P6-6, Table 6-2, Element 1, 3rd bullet - The scope of this bullet should be revised to "Known areas of the canister to which temporary supports or attachments...." This would be based on document reviews of locations where temporary attachments were used.*

NRC Response: The staff agrees with the comment and revised the text to reflect the recommended change.

See also the responses to Comments 8.1.3, 8.1.5, and 8.2.2.

Comment 8.1.3

Comment: *P6-6, Table 6-2, Element 1, 3rd bullet - Add ", if known" to end of sentence.*

NRC Response: The staff agrees with the comment and revised the text to reflect the recommended change.

See also the responses to Comments 8.1.2, 8.1.5, and 8.2.2.

Comment 8.1.4

Comment: *P6-6-16, Table 6-2 - Current draft AMP specifies the use of volumetric examination methods for determining size and depth of pits or cracks indication potential for SCC. However, such inspection techniques are not currently commercially available with a proven performance record, and therefore, would be difficult to implement at the current time. The AMP should be modified to specify that volumetric examination methods may be implemented when proven effective for the inspection conditions and have proven results.*

NRC Response: The staff agrees with the comment and clarified the guidance accordingly.

Additional guidance has been added to Section 6.5 and AMP element 4 to address the possible situation where visual examination has identified a potential area of aging but the available methods are not suitable to conduct surface and/or volumetric examination of the identified area. In these cases, the licensee should propose alternative evaluation methods, analyses and/or mitigation strategies as necessary to ensure that the important to safety functions of the system are maintained throughout the period of extended operation.

Comment 8.1.5

Comment: *P6-6-16, Table 6-2 - It is noted that not all areas of temporary attachments were mapped relative to the longitudinal weld of the canister and therefore it may be impossible to identify all such areas on canisters fabricated in the early 2000s. However, it is noted that all NAC fabricated canisters had temporary attachments removed correctly, surfaces prepped, and dye penetrant examined in accordance with the requirements of the NAC Fabrication Specification. The AMP should be revised to state that areas of removed temporary attachments will be identified to the extent possible based on available fabrication records and canister surface accessibility for inspection.*

NRC Response: The staff agrees with the comment and revised the text to reflect the recommended change.

See also the responses to Comments 8.1.2, 8.1.3, and 8.2.2.

Comment 8.1.6

Comment: *P6-6-16, Table 6-2 - What level of examination of the canister surface is expected, e.g., 75%, 50%, or best effort?*

NRC Response: The staff reviewed the comment and revised the guidance to clarify the expected inspection coverage.

The staff recommends that the extent of examination should focus on accessible areas of the canister and the focus be on the surfaces of the canisters where welds are located. The staff revised the scope of program to state "Examinations should be focused on accessible canister welds, weld heat affected zone (HAZ) areas and known areas of the canister to which temporary supports or attachments were attached by welding and subsequently removed (based on available fabrication records)." The AMP provides a list of significant attributes of canister areas for the focused inspection.

Comment 8.1.7

Comment: *P6-7, Table 6-2, Element 4, 3rd sentence under the Volumetric Examination section - Recommend not implying that cleaning is to be performed simply because it can be done. A cleaning assessment may be more appropriate prior to cleaning than the assumption to clean simply because it's possible. This was evident on the vertical surfaces of the GTCC canister at Maine Yankee which were absent of surface accumulation and would not need a cleaning. Also, there have been discussions at the ASME code case meetings on whether the industry wants to disturb the as-found condition on the canisters as with a cleaning.*

NRC Response: The staff agrees with the comment and has removed the statement that may imply that cleaning is to be performed.

The staff notes that ASME Code Section XI, Article IWA-2210, "Visual Examination," references Section V, Article 9, which states that visual examinations shall be performed in accordance with a written procedure. The licensees written procedures should address cleaning methods, if required. The sentence in question was edited as follows: "For accessible areas, ~~where adequate cleaning can be performed,~~ remote visual examination meeting the requirements for VT-1 Examination (ASME Code Section XI, IWA-2211) may be used to determine the type of degradation present (e.g., pitting corrosion or SCC) and the location of degradation." In addition, the first three sentences of this paragraph have been moved to a more appropriate location under the "Visual Examination" heading.

Comment 8.1.8

Comment: *P6-7, Table 6-2, Element 4, Visual Examination section - A performance demonstration is overly conservative for this type of inspection and not warranted as corrosion areas will be readily apparent to prompt further investigation. The visual inspection requirements in IWA 2200 that MAPS cites further down the paragraph are adequate.*

NRC Response: The staff agrees with the comment and has revised the guidance to state: "Visual examination procedures should follow ASME B&PV Section V, Article 9 and ASME Code Section XI, Article IWA 2200 for VT-1 and VT-3 examinations (ASME, 2007) and BWRVIP 03 (Selby, 2005) for EVT-1 examinations."

Comment 8.1.9

Comment: *P6-9, Table 6-2, Element 5, 1st bullet - Likely, the CoC renewal application will not be approved before or at the beginning of the period of extended operation, therefore NAC is questioning how a baseline would be performed if the AMP is not approved yet. The site would be taking a risk that the AMP is changed during the application review process and therefore the baseline would not really be a representative baseline if the AMP is changed.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the recommendation for AMP implementation schedules.

The commenter's concern is in regards to sites that have systems that have been in operation for more than 20 years and are operating under a license or CoC in timely renewal. The staff revised the example AMPs to note that an applicant can propose an inspection implementation schedule that accommodates such scenarios. Section 3.6.3 of NUREG-1927, Revision 1, provides recommendations for AMP implementation schedules that consider timely renewal.

Comment 8.1.10

Comment: *P6-10, Table 6-2, Element 6, 2nd and 3rd paragraphs - ASME Section XI Class 1 acceptance criteria is not designed for this application, whereas the EPRI guidance in document 3002008193 has been generated for canister inspections and acceptance criteria. Recommend using the EPRI acceptance criteria.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the recommended options for establishing acceptance criteria.

The staff revised the guidance to recommend that an applicant "may" (rather than "must") use the ASME Code acceptance criteria since this is an example AMP and alternative acceptance criteria may be used. A specific reference to EPRI Report 3002008193, "Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless Steel Canisters," has been added as an example of such an alternative. This comment is similar to Comments 8.2.7 and 8.2.11.

Comment 8.1.11

Comment: *P6-10, Table 6-2, Element 6, 4th paragraph - This removal of iron deposits and rust stains should be reserved for welds and their associated heat affected zones. While this section implies this, it is not clearly stated, and the section should be revised accordingly.*

NRC Response: The staff agrees with the comment and has revised the guidance to state that the removal of iron deposits and rust stains is specific to accessible welds and weld heat affected zones.

Comment 8.1.12

Comment: *P6-10, Table 6-2, Element 6, 2nd bullet - Establishing an acceptance criteria of a single 1 mm diameter red/orange colored is overly conservative and may extend a site's inspection well beyond what it was intended on being due to having to investigate numerous non-relevant indications. The corrosion specimens presented at various conferences/committees reflect a cluster of circular corrosion indications versus just a single spot, therefore NAC recommends basing this criteria on the density of corrosion indications and/or a larger diameter threshold with referenced basis.*

NRC Response: The staff agrees with the comment and has revised the acceptance criteria to state that an "accumulation" of corrosion products are subject to additional examination and disposition.

Comment 8.1.13

Comment: *P6-12, Table 6-2, Element 7, last sentence - At ISFSI-only sites the canister cannot be taken out of service like an active component, and therefore, NAC recommends the course of action would be for the site to enter the condition into their corrective action program and allow the site to evaluate the situation to determine the proper action.*

NRC Response: The staff disagrees with the implication of the comment (removal of a degraded canister is the only appropriate corrective action); however, the guidance was revised to clarify.

Although the staff considers the original text in the draft guidance to be correct, the staff revised the guidance to ensure that it does not unintentionally imply that the only option for addressing a canister that does not meet the evaluation criteria is to remove the canister from service. However, taking a canister out of service remains an option, absent other means of ensuring adequate protection of public health and safety and protection of the environment.

As stated in the response to Comment 2.1.9, the staff recommends that corrective actions be determined on a case-specific basis, given the wide range of potential inspection findings, storage system designs, and available repair and recovery options. The NRC will evaluate whether the licensees' corrective actions are effective and adequate to maintain the intended functions of the canisters, and whether the licensee remains compliant with the requirements in 10 CFR Part 72.

<u>Comment 8.1.14</u>

Comment: *P6-18, Table 6-3, Element 1, Item 3 - Radiation surveys are initially obtained per Technical Specification requirements and are also taken routinely at the site, therefore additional surveys would be unnecessary.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to describe alternatives to performing additional radiation surveys in an aging management program.

The staff agrees that radiation surveys are initially obtained per Technical Specification requirements and radiation monitoring is also conducted at the sites. However, after storage systems are loaded and placed on the ISFSI support pad, ongoing surveys are generally not conducted in a manner that allows for monitoring for concrete degradation of the individual storage systems. Nevertheless, an applicant may demonstrate that the radiation surveys and monitoring conducted at its site are capable of ensuring that the shielding performance of the concrete overpacks is maintained per the approved design basis. Such a proposal should include a description of the surveys and monitoring, including when and how they are performed, and justification of how they are sufficient to identify degradation of the shielding performance due to aging effects for individual storage systems.

Alternatively, the staff has determined that, if supported by a technical justification, visual inspections of the concrete per ACI 349.3R-02 can be an acceptable approach to managing loss of shielding due to concrete degradation. The use of this approach should be supported by a shielding evaluation that demonstrates that the ACI 349.3R-02 acceptance criteria (developed to assess structural performance) are sufficiently conservative to provide for timely identification of concrete degradation and corrective actions before a loss of shielding performance.

The staff has revised the introductory section of the Reinforced Concrete Structures AMP to describe an acceptable approach for the use of ACI 349.3R-02 visual inspections to manage loss of shielding. In its technical justification for the use of this approach, an applicant may reference shielding evaluations performed by the NRC on selected storage system designs that identify instances where the use of visual inspections in lieu of radiation surveys may be justified (NRC, 2019). An applicant may reference the NRC evaluations, provided that (1) the applicant can justify that the NRC evaluations apply to, or are bounding for, the applicant's design, including consideration of the assumptions and system parameters (both design and contents) used in the NRC evaluations and (2) the NRC evaluations indicate that the use of visual inspections for that design would be sufficiently conservative for ensuring against a loss of shielding performance.

<u>References</u>

NRC. "Study of ACI 349.3R Concrete Evaluation Criteria Impacts on Dose Rates for Several Spent Fuel Dry Storage System Designs." Washington, DC. ADAMS Accession No. ML19072A031. 2019.

Comment 8.1.15

Comment: *P6-18, Table 6-3, Element 1, Item 2 - Need additional clarification for groundwater chemistry monitoring - is this deep wells or surface conditions in direct contact with the concrete being monitored? This should be ground in immediate vicinity/depth of concrete being monitored vs drilling deep wells.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The guidance on sampling locations for groundwater chemistry is defined in Table 6-3, Element 4, "Detection of Aging Effects", "Sample Size". This guidance is consistent with ACI 349.3R-18, "Report on Evaluation and Repair of Existing Nuclear Safety-Related Concrete Structures," (2018) which states that sampling locations should be representative of the environment to which the structure is subjected.

Reference

ACI 349.3R-18, "Report on Evaluation and Repair of Existing Nuclear Safety-Related Concrete Structures." American Concrete Institute. 2018.

Comment 8.1.16

Comment: *P6-18, Table 6-3, Element 1, Item 3 - Item 3 is not related to aging management. Sites are already required to perform periodic monitoring of boundary doses which are cited as sufficient to detect failed systems and abnormal conditions. Sufficient radiation monitoring will already be performed during execution of other aging management activities above and beyond surveys required by DSS and ISFSI TS requirements. A radiation monitoring requirement embedded in an AMP is not only not substantiated but it will cause undue work and exposure to station employees for no gained benefit. All other references to radiation surveys should be removed from Table 6-3.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to describe alternatives to performing additional radiation surveys in an aging management program.

See the response to Comment 8.1.14.

Comment 8.1.17

Comment: *P6-19, Table 6-3, Element 2 - The first paragraph should be deleted. This paragraph is not substantiated by the MAPS document and is not related to or mitigate any aging management issues. TS monitoring requirements (via temperature monitoring or*

inlet/outlet inspections) are not changed during license renewal and will continue to be required on a more frequent basis than prescribed here. Any abnormal conditions identified by this already required monitoring will be corrected by licensee's corrective action program.

NRC Response: The staff disagrees with the comment; however, the guidance was revised to clarify the role of the vent monitoring.

Inlet/outlet vent monitoring (via direct temperature monitoring or general area walkdowns) per Technical Specification requirements are credited as a preventive action to ensure that thermal dehydration of the concrete remains non-credible during the period of extended operation. The staff revised Section 3.5.1.11 to tie these activities with the conclusions on the technical basis related to thermal dehydration of concrete.

Comment 8.1.18

Comment: *P6-21, Table 6-3, Element 4, Frequency of Inspection section - 2nd paragraph lists 5-year interval which seems to contradict first paragraph of inspections per ACI 349.3R-02.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The inspection frequencies for above-grade (both readily accessible and normally inaccessible) areas are consistent with Table 6.1 of ACI 349.3R-02 and Table 6 of the revised ACI 349.3R, "Report on Evaluation and Repair of Existing Nuclear Safety-Related Concrete Structures."

Comment 8.1.19

Comment: *P6-21, Table 6-3, Element 4, Sample Size section - 1st paragraph - 100% surface area concrete structure inspection should be limited to sampling size of 1 or 2, similar to canister inspection. Detailed inspection of a few systems will be sufficient to identify initiation of issues that would warrant increased inspections per licensee's CAP program (extent of condition evaluation). Performing 100% surface exam of all systems will result in undue work and exposure with no measurable benefit.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff considers that the evaluation of a subset of structures is an adequate approach for minimizing undue work and radiation exposure while ensuring that adverse conditions are identified on a timely manner. Such approach would be similar to that of the AMP for stainless steel canisters. The staff recognizes that, per the licensee's approved Quality Assurance Program, any condition adverse to quality, such as failure, malfunction, deficiencies, deviation,

defective material and equipment, and nonconformance, will be promptly identified and corrected. In the case of a significant condition identified as adverse to quality, the measures must ensure that the cause of the condition is determined and corrective action is taken to preclude repetition. Further, the licensee will be expected to evaluate the need for an extent of condition per their Corrective Action Program.

The staff, therefore, has revised the discussion in Table 6-3 to allow for the evaluation of readily accessible areas from a subset of the reinforced concrete structures within the scope of renewal. The staff has proposed that a minimum of two of the same structures (i.e., same design bases) be evaluated every 5 years, consistent with the minimum inspection frequency and acceptance criteria in Table 6-3. An applicant is expected to justify the selection of the two representative structures per their length of service, environment (heat and radiation exposure, location), accessibility, prior repairs and/or observed degradation. For example, an applicant may provide justification for the selection of the earliest-loaded overpack and the overpack loaded with the highest heat-load canister. If the specific structures to be evaluated are not predefined, then the applicant would be expected to define their criteria for selection.

The aforementioned approach should be complemented with periodic walkdowns of all reinforced concrete structures by personnel meeting the qualifications of either the responsible engineer or inspection personnel, per Chapter 7 of ACI 349.3R-02. The walkdowns should allow for a gross non-quantitative evaluation of all other structures to ensure that the limited sample size for evaluation per ACI 349.3R acceptance criteria remains adequate.

Further, the staff has revised the introduction in Section 6.6 to clarify that an applicant may credit maintenance activities conducted per the FSAR for the general area walkdowns described in this AMP, if these activities meet all 10 elements of this AMP.

Comment 8.1.20

Comment: *P6-18-25, Table 6-3 - Reinforced Concrete Structures AMP - Currently the draft AMP specifies doing 100% inspection of external concrete structures for 100% of the concrete casks deployed at a given site. Would it be considered appropriate to perform the inspections of large deployments of casks (e.g., 60 or 100+ casks per site) to do the inspections in tranches of 20% of deployed casks based on age instead of the frequency and timing specified. Also, once a concrete cask enters the PEO, or GL determines it appropriate, would it be acceptable to defer or eliminate the normal VCC maintenance inspections performed annually or biannually as specified in the FSAR as they would be effectively redundant with the AMP?*

NRC Response: The staff reviewed the comment and revised the guidance to recommend inspecting a sample of concrete structures, supplemented by period walkdowns of all structures. See the response to Comment 8.1.19.

Regarding the crediting of AMP activities for normal maintenance, the staff is unable to generically recommend that normal maintenance can be eliminated once the cask enters the period of extended operation. That determination must be evaluated on a case-by-case basis, considering whether the characteristics of the AMP activities are able to fully meet the maintenance requirements.

Comment 8.1.21

Comment: *P6-18-25, Table 6-3 - Under Scope of Program, Item 3, Radiation Survey requirements does not appear to be necessary or beneficial to determining the aging effects on a VCC. At the current time, all Licensees are required to verify compliance to 10 CFR 72.104 for off-site dose. Also prior to and during Reinforced Concrete Structure inspections, radiation protection personnel (RP) will survey the entire area of the planned work to establish dose rates for personnel access. Requiring quarterly additional individual dose assessments for each cask would require significant resources including man-lift(s) and several RP personnel to access all areas of the cask originally surveyed following loading. This would also result in significant additional personnel dose with no discernable benefit and not compliant with ALARA principles. Due to the decay of the stored fuel assemblies, all doses are expected to be lower than originally measured and the cask structure would have needed to withstand significant degradation to loss the effectiveness of the concrete and steel shields. Such degradation would be observable during the concrete structures or external/internal steel structures inspections.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to describe alternatives to performing additional radiation surveys in an aging management program.

See the response to Comment 8.1.14.

Comment 8.1.22

Comment: *P6-30, Table 6-4, Element 4 - The parameters monitored do not need a VT-3 level of inspection to identify loss of coating, general corrosion, missing hardware...etc. This level of inspection has been and can easily be accomplished by the resources at the sites. Recommend leaving the training and qualification to the sites for these inspections.*

NRC Response: The staff agrees with the comment and revised the visual inspection criteria.

The staff notes that, for metallic surfaces not associated with the confinement boundary, inspections by systems engineers or maintenance personnel provide reasonable assurance that aging will be identified prior to a loss of function. As a result, the staff has removed the recommendation for inspections to be performed to ASME VT-3 by certified inspectors. However, if a specific inspection standard is not cited, the staff considers it necessary for inspection procedures to be demonstrated to be capable of identifying degradation, and additional guidance to address this has been added to the AMP.

Comment 8.1.23

Comment: *P6-31&32, Table 6-4, Elements 4&5 - Timing/Baseline Inspection: The initial inspections should not start until the AMP is approved in the CoC renewal application.*

Section 3.6.3 under NUREG–1927, Revision 1 allows one year to implement the AMPs after the CoC is approved.

NRC Response: The staff agrees with the comment and revised the guidance to clarify the recommendation for AMP implementation schedules.

See the response to Comment 8.1.9.

<u>Comment 8.1.24</u>

Comment: *P6-32, Table 6-4, Element 6 - Having acceptance criteria of no coating defects and no corrosion products in the base metal is too conservative as this does not affect the function of the VCC structure. The first bullet is more appropriate to ensure the base metal is sound. Coating damage on outside surfaces has been both monitored and repaired at sites, but having an acceptance criterion of no coating damage is unrealistic. Coating damage found on sheltered/inaccessible areas shows superficial corrosion on exposed areas, which can be monitored and/or addressed with a TLAA.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff believes that the existing first bullet provides sufficient criteria to ensure that the degradation of metallic components is identified prior to a loss of intended function. The AMP has been revised to remove those acceptance criteria recommendations. However, to provide reasonable assurance that corrosion under coatings is identified and evaluated, the staff added additional guidance to the Detection of Aging Effects program element. Also, the staff notes that the coatings referenced in this program refer to those that are used for corrosion mitigation, but do not perform an important-to-safety function. For an important-to-safety coating, an applicant should propose aging management activities appropriate to maintain the coating's function.

<u>Comment 8.1.25</u>

Comment: *P6-33, Table 6-4, Element 10 - The operating experience cited for this AMP is about stainless steel CISCC which is outside the scope of this AMP. There is Op-Ex in the AMID database that applies to this AMP and describes inspection findings on a GTCC VCC cask. These examples include coating failures yet superficial corrosion on the carbon steel in the sheltered regions of the cask.*

NRC Response: The staff agrees, in part, with the comment and revised the description of operating experience that is relevant to the subject AMP.

The staff notes that the AMP manages cracking of stainless steel components, and thus the existing reference to the Information Notice on SCC is appropriate. However, because the AMP largely focuses on carbon and alloy steel components, The “Operating Experience” AMP element was revised to include observations of steel surfaces within the horizontal storage modules at Calvert Cliffs and the Three Mile Island Unit 2 storage facility.

Comment 8.1.26

Comment: *P6-35, Section 6.8 - This verification is done daily at the sites (per procedure OP-1 at the Yankee sites) to satisfy Tech Spec requirements, therefore this AMP would be redundant and probably lead to confusion at the sites. Recommend deleting this AMP.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff recognizes that sites currently have requirements for ensuring the operability of ventilation systems. The original impetus for the AMP was to recognize the role of ventilation monitoring in preventing concrete degradation due to elevated temperature exposure (as described in the Preventive Actions program element in the Reinforced Concrete Structures AMP). The staff considers that the reference to sites’ current operating practices in the Reinforced Concrete Structures AMP to be sufficient to capture the role of this activity. The Ventilation Systems AMP has been removed.

Comment 8.1.27

Comment: *P6-35, Section 6.8 - This entire section should be deleted. Validation of ventilation acceptability is already required by DSS TS and is not specifically related to or mitigate any aging management condition. DSS TS already have a monitoring requirement sufficient to detect abnormal conditions. Any abnormal condition will then be entered into the licensee’s CAP. This section will do nothing but provide additional administrative burden on the licensee.*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 8.1.26.

Comment 8.1.28

Comment: *P6-36-41, Table 6-5 - It does not appear that a separate AMP is required for Ventilation Systems, as temperature monitoring and/or visual inspection program requirements are currently specified in the CoC’s Technical Specifications. The inspection of the actual inlet and outlet vent assemblies can be appropriately incorporated into the internal or external*

surfaces monitoring of metallic components to determine if there are vent blockages that could negatively affect the thermal performance of the concrete cask structure.

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 8.1.26.

Comment 8.1.29

Comment: *P6-37-40, Table 6-5, Elements 4&9 - Temperature monitoring is not safety related or calibrated, at least at the NAC-UMS sites. It is not relied upon to satisfy Tech Specs as the loss of temperature monitoring does not impact the passive system and a visual surveillance will satisfy the Tech Spec. TEs and RTDs are inherently reliable and failures are self-evident by going high, therefore no calibration is warranted.*

NRC Response: The staff notes that the comment is no longer applicable because the subject AMP has been removed.

See the response to Comment 8.1.26.

Comment 8.1.30

Comment: *P6-60, Table 6-8, Element 1 - There should be a statement in the Scope that this AMP does not apply if the HBU fuel is canned or otherwise contained.*

NRC Response: The staff agrees with the comment and has clarified throughout Section 6.11 that the High-Burnup Fuel Monitoring and Assessment Program only applies to uncanned (undamaged or intact cladding) high-burnup fuel.

8.2 Comments from Nuclear Energy Institute

Comment 8.2.1

Comment: *P6-1, Table 6-1 - Suggest breaking Section 6.7 into two separate AMPS. One for readily accessible metallic external surfaces exposed to outdoor atmospheres and one for sheltered, internal metallic surfaces available for inspection during remote camera inspections of the canister. There will be different inspection programs repairs inspection frequencies, etc. for these two cases.*

NRC Response: The staff disagrees with this comment; however, the guidance was revised to clarify the scope of the AMP.

See the response to Comment 6.1.20.

Comment 8.2.2

Comment: *P6-6, Table 6-2, Element 1, 3rd bullet - The scope of this bullet should be revised to "Known areas of the canister to which temporary supports or attachments...." This would be based on document reviews of locations where temporary attachments were used.*

NRC Response: The staff agrees with the comment and revised the guidance to reflect the recommended change.

See the response to Comment 8.1.2.

Comment 8.2.3

Comment: *P6-6, Table 6-2, Element 1 - Regarding this item: Effort should be made to identify and prioritize examinations of areas on canisters that have two or more of the above attributes (e.g., canister surface that is cold relative to average surface temperature and also has a weld or weld heat affected zone). Why prioritize any area that is not near a weld? What does "prioritize" mean relative to coverage area? Is canister removal from overpack expected? The actions and scope should be commensurate with the safety significance and this needs to be conveyed in the document.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the recommended areas of focus.

The staff revised the "Scope of Program" AMP element to clarify that examinations should be focused on accessible canister welds and weld heat affected zones with specific attributes (e.g., relatively cold surfaces, surfaces with atmospheric deposits). Removal from the overpack to conduct initial visual examination of canisters is not expected to be necessary. See also the response to Comment 8.1.6.

Comment 8.2.4

Comment: *P6-7, Table 6-2, Element 4, Volumetric Inspection section - The phrase "adequate cleaning" needs to be qualified. There is evidence of some canisters requiring cleaning to remove heavy deposits of pollen, but there is evidence of other canisters having bare metal surfaces that require no cleaning.*

NRC Response: The staff notes that the comment is no longer applicable because the guidance regarding cleaning has been removed.

See the response to Comment 8.1.7.

Comment 8.2.5

Comment: *P6-7, Table 6-2, Element 4, Volumetric Inspection section - The phrase "For accessible areas where adequate cleaning should be performed...", "accessible" should be qualified as this is accessible by remote VT.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment

The staff notes that the phrase referenced in the comment was revised in response to Comment 8.1.7 to no longer address cleaning. The staff considers the revised text to be sufficiently clear with respect to the definition of an accessible area. Additional changes to address the comment are not considered necessary.

Comment 8.2.6

Comment: *P6-8, Table 6-2, Element 4, Sample Size section - Please change the first phrase to "For sites conducting a canister inspection,"*

NRC Response: The staff agrees with the comment and has revised the guidance to state: "For sites conducting a canister examination, there should be a minimum of one canister examined at each site."

Comment 8.2.7

Comment: *P6-10, Table 6-2, Element 6, 2nd and 3rd paragraphs - Use of ASME Section XI acceptance criteria for RCS piping is inappropriate for this application involving low pressure canisters of ductile material. AMP as it is not based on the canister design. There is a significant difference in response to defects between the high temperature and pressure of RCS piping and the low temperatures and pressures of dry storage. Acceptance criteria have been identified in EPRI-3002008193 that are appropriate for this service and they should be for use in this AMP.*

NRC Response: The staff agrees with the comment and revised to guidance to clarify the recommended options for establishing acceptance criteria.

See the response to Comment 8.1.10.

Comment 8.2.8

Comment: *P6-10, Table 6-2, Element 6, 4th paragraph - This removal of iron deposits and rust stains should be reserved for welds and their associated heat affected zones. While this section implies this, it is not clearly stated, and the section should be revised accordingly.*

NRC Response: The staff agrees with the comment and has revised the guidance to state that the removal of iron deposits and rust stains is specific to accessible welds and weld heat affected zones.

See the response to Comment 8.1.11.

Comment 8.2.9

Comment: *P6-10, Table 6-2, Element 6, 2nd bullet - What is the basis for the 1mm criteria? This appears to be the first use of this criteria and the basis for it has not been identified and justified for this application. In lieu of using this, acceptance criteria have been identified in EPRI-3002008193 that are appropriate for this service and they should be for use in this AMP.*

NRC Response: The staff reviewed the comment and has revised the acceptance criteria to state that an "accumulation" of corrosion products are subject to additional examination and disposition.

See the response to Comment 8.1.12.

Comment 8.2.10

Comment: *P6-10, Table 6-2, Element 6 - The guidance is not clear on how indications may be dispositioned. The language provided indicates that these are not actually criteria for acceptance, rather criteria for doing additional evaluations. The referenced standards do not specifically address the question of confinement integrity. What are examples of indications that can be accepted for continued service? What indications cannot be accepted?*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the acceptance criteria.

The staff has revised the acceptance criteria to clarify visual inspection findings that are subject to additional examination and the means by which an applicant can determine if the canister is suitable for continued service.

Comment 8.2.11

Comment: *P6-10, Table 6-2, Element 6 - This blanket statement: "No indications of localized corrosion pits, etching, crevice corrosion, SCC, red -orange-colored corrosion products emanating from crevice locations, or red-orange-colored corrosion products in the vicinity of canister fabrication welds, closure welds, and welds associated with temporary attachments during canister fabrication," may prevent any inspection from being acceptable without further evaluation. Consider using acceptance criteria provided in referenced document EPRI-3002008193.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the recommended options for establishing acceptance criteria.

See the responses to Comments 8.1.10 and 8.1.12.

Comment 8.2.12

Comment: *P6-11&12, Table 6-2, Element 7, last sentence - "Canisters with localized corrosion or SCC that do not meet the prescribed evaluation criteria are not permitted to remain in service without an engineering analysis or mitigation actions." - Is this referring to the acceptance criteria in element 6 or to some other prescribed evaluation criteria?*

NRC Response: The staff notes that the subject text was removed in response to Comment 8.1.13; therefore, no changes were made in response to the comment.

The staff notes that the "evaluation criteria" stated in the draft guidance was in reference to the criteria against which the canister is evaluated for continued service (i.e., the canister remains capable of fulfilling its intended functions and remains compliant with the requirements in 10 CFR Part 72). In response to Comment 8.1.3, the staff revised the guidance to clarify.

Comment 8.2.13

Comment: *P6-12, Table 6-2, Element 7, last sentence - The phrase "are not permitted to remain in service" does not properly recognize the nature of passive dry storage systems. Suggest rewording this last sentence as follows: "Canisters with localized corrosion or SCC that do not meet the prescribed acceptance criteria shall be entered into the licensee's corrective*

action program as a condition adverse to quality to allow for appropriate evaluations and follow-up."

NRC Response: The staff notes that the subject text was removed in response to Comment 8.1.13; therefore, no changes were made in response to the comment.

See the responses to Comments 8.1.13 and 8.2.12.

Comment 8.2.14

Comment: *P6-13, Table 6-2, Element 10, Operating Power Reactors section - This section, the one that follows, and the end of Table 6-4 are the only places in MAPS that identifies crack growth rates. With the exception of the Kosaki reference, these crack growth rates are "apparent" crack growth rates that are empirically derived from operating data. The nature of these numbers is well explained in EPRI 3002002528 which is cited in the next paragraph. This data is useful, but appropriate qualifications should be identified where they are cited to ensure that this fact is understood. A follow-up EPRI report, EPRI 3002002785 "EPRI Public Flaw Growth and Flaw Tolerance Assessment for Dry Cask Storage Canisters" provides a more in-depth treatment of this area and may be a useful reference.*

NRC Response: The staff agrees with the comment and has revised the Operating Experience section and added the recommended reference.

Comment 8.2.15

Comment: *P6-18, Table 6-3, Element 1, Item 2 and elsewhere in this AMP - Groundwater monitoring as part of concrete aging management - This would be conducted when a pad is in scope and there is reason to believe that the groundwater is conducive to applicable concrete degradation. If past characterization shows that the groundwater is below the cited limits, then new additional characterization, new groundwater monitoring wells, or routine monitoring requirements should **NOT** be required. Groundwater monitoring has shown little change over many years of trending when there are no new contributors to a site and this is not an insignificant expense.*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The staff considers that the specific scenario presented by the commenter can be evaluated on a case-by-case basis. The Reinforced Concrete Structures AMP is only an example approach. The staff will evaluate the technical basis (site-specific operating experience, engineering justification) provided by the applicant when determining the acceptability of proposed

deviations from the Reinforced Concrete Structures AMP. The staff considers that Element 4 of the Reinforced Concrete Structures AMP, "Detection of Aging Effects," allows ample flexibility for the proposed groundwater chemistry monitoring frequency pending an adequate technical basis.

Comment 8.2.16

Comment: *P6-18, Table 6-3, Element 1, Item 3 & Footnote 1 and elsewhere in this AMP - Radiation surveys as part of concrete aging management - Radiation surveys are initially conducted per Tech. Spec. requirements. Usually, this means the cask is surveyed and verified to meet requirements before it is allowed into storage. The need for follow-up surveys, as part of aging management, is unnecessary. Dry cask storage systems are routinely performed and evaluated IAW site approved procedures. This approach is more conservative and robust than an extra and unnecessary aging management driven survey process. This is not ALARA.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to describe alternatives to performing additional radiation surveys in an aging management program.

See the response to Comment 8.1.14.

Comment 8.2.17

Comment: *P6-18, Table 6-3, Element 1, Item 3 - Item 3 is not related to aging management. Sites are already required to perform periodic monitoring of boundary doses which are cited as sufficient to detect failed systems and abnormal conditions. Sufficient radiation monitoring will already be performed during execution of other aging management activities above and beyond surveys required by DSS and ISFSI TS requirements. A radiation monitoring requirement embedded in an AMP is not only not substantiated but it will cause undue work and exposure to station employees for no gained benefit (Not in keeping with ALARA principals). All other references to radiation surveys should be removed from Table 6-3.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to describe alternatives to performing additional radiation surveys in an aging management program.

See the response to Comment 8.1.14.

Comment 8.2.18

Comment: *P6-18&20, Table 6-3, Element 1, 4th bullet from the bottom and elsewhere in this AMP & Element 3 - Leaching of calcium silicate and efflorescence as part of concrete aging management. This statement should say "due to* ***excessive*** *leaching of calcium hydroxide." As discussed in NUREG/CR–7116, NUREG/CR–7153 and IAEA TECDOC–1025, calcium leaching is not considered a significant degradation mechanism and would not result in a significant impact to the safety functions of the concrete. There are many instances of leaching of calcium carbonate on dry storage canisters and most are self-limiting and benign. Only the more excessive ones with corresponding affected concrete needs to be tracked, evaluated and potentially remediated.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The staff has strived to ensure that all example AMPs in NUREG–2214 are consistent with NUREG–1927, Revision 1, Appendix A. Therefore, the staff does not agree with introducing ambiguous non-quantitative terminology into the Reinforced Concrete Structures AMP. The Reinforced Concrete Structures AMP implements quantitative acceptance criteria from the consensus standard guide, ACI 349.3R-02.

Comment 8.2.19

Comment: *P6-19, Table 6-3, Element 2 - The first paragraph should be deleted. This paragraph is not substantiated by the MAPs document and is not related to or required to mitigate any aging management issues. TS monitoring requirements (via temperature monitoring or inlet/outlet inspections) are not changed during license renewal and will continue to be required on a more frequent basis than prescribed here. Any abnormal conditions identified by this already required monitoring will be corrected by licensee's corrective action program.*

NRC Response: The staff disagrees with the comment; however, the guidance was revised to clarify the role of the vent monitoring.

See the response to Comment 8.1.17.

Comment 8.2.20

Comment: *P6-21, Table 6-3, Element 4, Sample Size section - Table 6-3 references 100% inspection of all concrete structures. This overly prescriptive and should be an inspection based on sampling a few systems at 100% and a gross inspection of all other structures. Detailed inspection of a few systems will be sufficient to identify initiation of issues that would warrant increased inspections per licensee's CAP program (extent of condition evaluation). Performing*

100% surface exam of all systems will result in undue work and exposure with no measureable benefit.

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 8.1.19.

Comment 8.2.21

Comment: *P6-22, Table 6-3, Element 4, Timing section - These types of inspections are conducted as required by the certificate or part 72 license for the period of initial operation (during the annual inspections). Therefore, the timing of the initial inspections required by the AMP should be after the period of initial operation, which is consistent with NRC language on prior renewals. NRC has allowed up to 300 days after the effective date of the renewal to implement the AMP. Initial inspection timing is driven by the AMP implementing documents. What is specified here is contrary to recent rulemaking at 82 Federal Register 57819 renewing certificate of compliance No. 1004 - TN Americas LLC, Standardized NUHOMS Horizontal Modular Storage System.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the recommendation for AMP implementation schedules

See the response to Comment 8.1.9.

Comment 8.2.22

Comment: *P6-30, Table 6-4, Element 4, Readily Accessible Surfaces section - Too Prescriptive - While it may be important for canister walls with controlled thickness on the walls, it is not necessary to perform visual inspections of coated carbon steel surfaces to VT-3 and this will indeed cause hardship on ISFSI Only sites who do not have qualified personnel for this. There are numerous examples of rust spots in exposed carbon steel and this is readily detected by average inspections. Most coatings are NQ and do not perform a safety function on metallic dry storage components and do not require this level of inspection.*

NRC Response: The staff agrees with the comment and revised the visual inspection criteria.

See the response to Comment 8.1.22.

Comment 8.2.23

Comment: *P6-31, Table 6-4, Element 4, Timing section - These types of inspections are conducted as required by the certificate or part 72 license for the period of initial operation (during the annual inspections). Therefore, the timing of the initial inspections required by the AMP should be after the period of initial operation, which is consistent with NRC language on prior renewals. NRC has allowed up to 300 days after the effective date of the renewal to implement the AMP. Initial inspection timing is driven by the AMP implementing documents. What is specified here is contrary to recent rulemaking at 82 Federal Register 57819 renewing certificate of compliance No. 1004 - TN Americas LLC, Standardized NUHOMS Horizontal Modular Storage System.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the recommendation for AMP implementation schedules.

See the response to Comment 8.1.9.

Comment 8.2.24

Comment: *P6-31, Table 6-4, Element 4, Normally Inaccessible Surfaces section - Table 6-2 does not include a statement similar to this one: "The extent of inspection coverage should be specified and demonstrated to sufficiently characterize the condition of the metallic components." It is unclear why this is applicable in Table 6-4 and not in Table 6-2.*

NRC Response: The staff agrees with the comment and revised the guidance to consistently recommend that AMPs include a defined extent of inspection.

The staff agrees that there is an inconsistency in defining the extent of inspection coverage. Because the area available to be inspected by remote techniques varies by the component and the storage system design, generic guidance for inspection coverage is not practical. As a result, the staff has revised the guidance in Tables 6-3 and 6-4 to recommend that applicants define a minimum inspection coverage on a case-by-case basis, considering accessibility and the capability to characterize the condition of normally inaccessible SSCs.

Comment 8.2.25

Comment: *P6-32, Table 6-4, Element 6 - Too prescriptive - These acceptance criteria are impossibly restrictive. For instance, minor coating failures for structures that have been in service for 10+ years are common place. As most coatings on external surfaces are NQ they have no impact on safety. Minor superficial rust in the areas of these coating failures is also evident. The first bullet here is fine as is the 4th bullet. The 2nd and 3rd bullets however, are unduly restrictive. This is important because of the corrective actions section. Failure to meet*

bullets 2 and 3 are not necessarily conditions adverse to quality and do not require apparent cause evaluations and root cause evaluations.

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 8.1.24.

Comment 8.2.26

Comment: *P6-33, Table 6-4, Element 7 - Too prescriptive - This section provides details that are in all licensee's corrective action programs and is unnecessary and may be counterproductive. For instance, based on the above comment, failure to meet bullets 2 and 3 are not necessarily conditions adverse to quality and do not require apparent cause evaluations and root cause evaluations. While such actions may be warranted on canister shells, they are not necessary on large, thick coated carbon steel structures, where unsatisfactory amounts of corrosion will be readily visible at inspection opportunities.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff notes that the specific acceptance criteria bullets discussed in the comment have been removed as a result of the resolution to Comment 8.1.24. With the removal of those criteria, the staff considers the existing Correction Actions recommendations to be appropriate.

Comment 8.2.27

Comment: *P6-33, Table 6-4, Element 10 - The operating experience cited for this AMP has nothing to do with the issues addressed by this AMP. There is Op-Ex in the AMID database that applies to this AMP.*

NRC Response: The staff agrees, in part, with the comment and revised the description of operating experience that is relevant to the subject AMP

See the response to Comment 8.1.25.

Comment 8.2.28

Comment: *P6-35, Sec6.8 - This AMP is not necessary. The passive ventilation systems of dry storage systems are subject to daily tech. spec. verification and the structures associated with forming the passive ventilation system are addressed in other AMPS. By creating this AMP, a*

source of confusion, conflicting requirements, and unnecessary administrative requirements are also being created. We recommend deleting this AMP.

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 8.1.26.

<u>Comment 8.2.29</u>

Comment: *P6-35, Sec6.8 - This entire section should be deleted. Validation of ventilation acceptability is already required by DSS TS and is not specifically related to or required to mitigate any aging management condition. DSS TS already have a monitoring requirement sufficient to detect abnormal conditions. Any abnormal condition will then be entered into the licensee's CAP. This section will do nothing but provide additional administrative burden on the licensee.*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 8.1.26.

<u>Comment 8.2.30</u>

Comment: *P6-37, Table 6-5, Element 4 - The statement "Temperature monitoring is performed with qualified and calibrated measurement devices or sensors that are maintained in accordance with the site QA program" is not generally true. As these sensors have no impact on safety and are generally very reliable and not subject to significant inaccuracy, they are not in scope and not subject to these requirement at most ISFSIs.*

NRC Response: The staff notes that the comment is no longer applicable because the subject AMP has been removed in the resolution to Comment 8.1.26.

<u>Comment 8.2.31</u>

Comment: *P6-40, Table 6-5, Element 9 - Same comment as above...this equipment is generally not calibrated. RTD's and T/C either operate or fail...failure is obvious based on the circuitry for these systems.*

NRC Response: The staff notes that the comment is no longer applicable because the subject AMP has been removed in the resolution to Comment 8.1.26.

Comment 8.2.32

Comment: *P6-43, Sec6.9 and Table 6-6 - Bolted cask seal leakage monitoring systems have a TS to ensure the pressure is not decreasing and to ensure the system is functioning properly. An AMP is not necessary and is not specifically related to or required to mitigate any aging management condition. DSS TS already have a monitoring requirement sufficient to detect abnormal conditions. Any abnormal condition will then be entered into the licensee's CAP. This section will do nothing but provide additional administrative burden on the licensee.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

The staff notes that the example AMP is provided as one acceptable approach to manage loss of material due to corrosion of the cask sealing components. 10 CFR 72.42 and 72.240 states that renewal applications must include a description of programs to manage aging, and the staff recognizes that existing site procedures can be a part of that demonstration.

Comment 8.2.33

Comment: *P6-46, Table 6-5, Element 4, Timing of Inspections section - These types of inspections are conducted as required by the certificate or part 72 license for the period of initial operation (during the annual inspections). Therefore, the timing of the initial inspections required by the AMP should be after the period of initial operation, which is consistent with NRC language on prior renewals. NRC has allowed up to 300 days after the effective date of the renewal to implement the AMP. Initial inspection timing is driven by the AMP implementing documents. What is specified here is contrary to recent rulemaking at 82 Federal Register 57819 renewing certificate of compliance No. 1004 - TN Americas LLC, Standardized NUHOMS Horizontal Modular Storage System.*

NRC Response: The staff agrees with the comment and revised the guidance to clarify the recommendation for AMP implementation schedules.

See the response to Comment 8.1.9.

Comment 8.2.34

Comment: *P6-53, Sec6.10 - Some discussion should be added here that TCs would generally be considered tools and scoped out for aging management if they were not classified as*

important to safety. They are not in continuous service and play no role in storage operations at the ISFSI.

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

See the response to Comment 6.2.11.

<u>Comment 8.2.35</u>

Comment: *P6-56, Table 6-7, Element 6 – "No coating defects" is an impossible acceptance criterion for transfer casks. Coating defects should be identified, but do not necessarily need to be repair unless this is required by the corrective action program.*

NRC Response: The staff agrees with the comment and made the recommended change.

See the response to Comment 8.1.24.

<u>Comment 8.2.36</u>

Comment: *P6-57, Table 6-7, References section - Appears to be something missing in the second reference.*

NRC Response: The staff disagrees with the comment; therefore, no changes were made in response to the comment.

See the response to Comment 5.3.20.

<u>Comment 8.2.37</u>

Comment: *P6-60, Table 6-8, Element 1 - There should be a statement in the Scope that this AMP does not apply if the HBU fuel is canned or otherwise contained.*

NRC Response: The staff agrees with the comment and has clarified throughout Section 6.11 that the High-Burnup Fuel Monitoring and Assessment Program only applies to uncanned (undamaged or intact cladding) high-burnup fuel.

See the response to Comment 8.1.30.

Comment 8.2.38

Comment: *P6-60, Table 6-8, Element 1 - Says the scope is to provide a description of the design bases characteristics of the HBU fuel. You do not "design" HBU fuel. You design the DSS. Suggest changing "design bases characteristics" to something like "characteristics and properties assumed in the DSS design." This comment applies throughout Table 6-8 HBU Fuel AMP.*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff revised the text with the proposed language and revised the introductory section in Section 6.10 to clarify the intent of the term "design-bases fuel." The staff notes that the terminology used is consistent with 10 CFR 72.3 and NUREG–1927, Revision 1. Therefore, the use of "design bases" always refers to the Independent Spent Fuel Storage Installation (for a specific license) or the dry storage system (for a CoC).

Comment 8.2.39

Comment: *P6-60, Table 6-8, Element 1 - Nominal burnups in the HDRP are 50-55 GWD/MTU assembly average (not 53-58).*

NRC Response: The staff agrees with the comment and made the recommended change.

The staff recognizes that the nominal burnups in the DOE-EPRI High Burnup Fuel Demonstration Program have been revised. Therefore, the cited nominal burnups for the DOE-EPRI High Burnup Fuel Demonstration Program have been revised to the correct loaded values in Table 6-7, Element 1.

Comment 8.2.40

Comment: *P6-60, Table 6-8, Element 1 - Change "is to be licensed" to "is licensed"*

NRC Response: The staff agrees with the comment and made the recommended change.

Comment 8.2.41

Comment: *P6-60, Table 6-8, Element 1 - Last paragraph in Item 1 says to justify the surrogate demonstration program is applicable by demonstrating it is bounding for the specific licensee. Suggest adding language to justify use of the demonstration program if it is not bounding, similar to language from ISG-24 - e.g., "...that the demonstration fuel is reasonably*

characteristic of the stored fuel and the added burn-up will not change the results determined by the demonstration."

NRC Response: The staff agrees with the comment and added the proposed clarification.

Comment 8.2.42

Comment: *P6-61, Table 6-8, Element 6 - Do these three bullets/acceptance criteria make sense? Do they have a safety basis?*

NRC Response: The staff reviewed the comment and concluded that changes to the guidance are not necessary.

The purpose of the High Burnup Fuel Monitoring and Assessment Program is for a licensee or CoC holder to monitor and assess data and other information regarding high burnup fuel performance to confirm that the design-bases high burnup fuel configuration is maintained during the period of extended operation. The acceptance criteria in Table 6-8, Element 6, are defined to ensure that the design-bases assumptions continue to be maintained during the period of extended operation. These assumptions include expected levels of hydrogen, moisture, and other oxidants that support the technical conclusions regarding various aging mechanisms and effects for high burnup fuel cladding and assembly components, as discussed in Section 3.6 of NUREG–2214. Further, the acceptance criteria is intended to confirm the conclusions regarding creep (Sections 3.6.1.3 and 3.6.1.4), hydride reorientation (Section 3.6.1.1), cladding oxidation (Section 3.6.1.6), cladding condition (as determined by the license and CoC technical specifications), and conclusions on the structural performance of the cladding (per the assumed mechanical properties in the approved design bases). Therefore, the acceptance criteria serves to confirm the conclusions of both the approved safety analyses and about the consequences of various aging mechanisms on the safe storage of high burnup fuel during the period of extended operation.