FINAL SAFETY EVALUATION REPORT

FOR THE RANCHO SECO

INDEPENDENT SPENT FUEL STORAGE INSTALLATION

LICENSE RENEWAL

DOCKET NO. 72-11

LICENSE NO. SNM-2510

Office of Nuclear Material Safety and Safeguards
United States Nuclear Regulatory Commission

March 2020

CONTENTS

TABLES

INTRODUCTION

Under Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," the U.S. Nuclear Regulatory Commission (NRC) issued a specific license for the Rancho Seco independent spent fuel storage installation (ISFSI), Special Nuclear Material (SNM) License No. SNM-2510, for 20 years, with an expiration date of June 30, 2020. SNM-2510 authorizes Sacramento Municipal Utility District (SMUD) to receive, possess, transfer, and store spent fuel from the Rancho Seco Nuclear Generating Station in the Rancho Seco ISFSI. The Rancho Seco ISFSI is located within the owner-controlled area of the Rancho Seco site, which is owned and operated by SMUD. The Rancho Seco site comprises approximately 2,480 acres in Sacramento County, CA. The Rancho Seco ISFSI pad is approximately 225 feet by 170 feet in size and is contained within a licensed area of approximately 14 acres.

By letter dated March 19, 2018, SMUD submitted an application for renewal of SNM-2510 for the Rancho Seco ISFSI for an additional 40 years beyond the initial license term, as supplemented June 25, 2018; August 6, 2018; September 26, 2018; April 22, 2019; June 26, 2019; July 12, 2019; and January 23, 2020 (SMUD, 2018b, 2018c, 2018d, 2018e, 2018f, 2019a, 2019b, 2019c, 2020). The staff generally refers to this application, as supplemented, as the "license renewal application" (LRA) in this safety evaluation report (SER). Any specific references to sections of the LRA are to Revision 3, which was included in SMUD's submittal of July 12, 2019 (SMUD, 2019c). The license renewal, if approved, would authorize the applicant to continue storing spent fuel in the Rancho Seco ISFSI until June 30, 2060. The applicant submitted the LRA in accordance with the regulatory requirements of 10 CFR 72.42, "Duration of license; renewal." Because SMUD submitted the LRA more than 2 years before the license expiration date, this application constitutes a timely renewal under 10 CFR 72.42(c).

The Rancho Seco specific license, SNM-2510, provides for storage of all spent fuel assemblies and control components from the Rancho Seco Nuclear Generating Station in the Rancho Seco ISFSI. The storage technology used at the Rancho Seco ISFSI is a NUHOMS® canister-based system consisting of a dry shielded canister (DSC) and a reinforced concrete horizontal storage module (HSM). The three types of DSCs used for storage of spent fuel are (1) fuel only (FO), (2) fuel with control components (FC), and (3) failed fuel (FF). The Rancho Seco DSC designs are based on the Standardized NUHOMS®-24P DSC design, except that the FO and FC DSCs include fixed neutron absorbers. In addition, modifications have been made to the DSC cavity basket and spacer disc design. All three types of DSCs containing spent fuel assemblies are stainless steel alloy welded pressure vessels that provide confinement of the radioactive material; encapsulate the fuel in an inert atmosphere; and provide axial biological shielding during DSC closure, transfer operations, and storage. The Rancho Seco ISFSI configuration consists of 18 FC DSCs, 2 FO DSCs, and 1 FF DSC. The Rancho Seco ISFSI also provides for storage of 100 percent of Rancho Seco's greater-than-class-C waste in a greater-than-class-C DSC.

The DSCs are stored in HSMs similar to the Standardized NUHOMS® HSM design. The HSM is a low-profile, modular, reinforced concrete structure whose primary functions are passively removing spent fuel decay heat, giving structural support and environmental protection to the loaded DSC, and providing radiation shielding protection.

Additional SSCs include a transfer cask (TC) and other canister transfer and auxiliary equipment used to support DSC loading and transfer operations. The TC, designated as the

NUHOMS® MP187 Transfer Cask, facilitated fuel loading and unloading as well as onsite transfer of a loaded DSC. In addition to these primary components, the ISFSI made use of auxiliary equipment consisting of a vacuum drying system, TC lifting yoke, DSC automatic welding system, hydraulic ram system, and a transfer trailer equipped with a TC skid, during loading operations. No more DSCs are to be loaded at the Rancho Seco ISFSI.

In the LRA, the applicant documented the technical bases for renewal of the license and proposed actions for managing potential aging effects on the structures, systems, and components (SSCs) of the ISFSI that are important to safety to ensure that these SSCs will maintain their intended functions during the period of extended operation. The applicant presented general information about the ISFSI design and a scoping evaluation to determine the SSCs within the scope of license renewal (the “in-scope” SSCs) and subject to an aging management review. The applicant further screened the in-scope SSCs to identify and describe the subcomponents that support their intended functions. For each in-scope SSC with an identified aging effect, the applicant proposed an aging management program or provided a time-limited aging analysis to ensure that the SSC will maintain its intended function(s) during the period of extended operation.

The NRC staff reviewed the applicant’s technical bases for safe operation of the ISFSI for an additional 40 years beyond the term of the current operating license. This SER summarizes the results of the staff’s review for compliance with 10 CFR 72.42. In its review of the LRA and development of the SER, the staff used the guidance in (1) NUREG-1927, Revision 1, “Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel,” issued June 2016 (NRC, 2016), (2) NUREG-2214, Revision 0, “Managing Aging Processes In Storage (MAPS) Report,” issued July 2019 (NRC, 2019), and (3) NUREG-1757, Volume 3, Revision 1, “Consolidated Decommissioning Guidance: Financial Assurance, Recordkeeping, and Timeliness,” issued February 2012 (NRC, 2012a), to ensure compliance with the NRC’s financial qualification requirements, including those associated with decommissioning the ISFSI, as appropriate. NUREG-2214 establishes a generic technical basis for the safety review of ISFSI LRAs, in terms of the evaluation of (1) aging mechanisms and effects that could affect the ability of ISFSI SSCs to fulfill their safety functions in the period of extended operation (i.e., credible aging mechanisms and effects) and (2) aging management approaches to address credible aging effects, including examples of aging management programs that are considered generically acceptable to address the credible aging effects to ensure that the design bases will be maintained during the period of extended operation. The staff evaluated the applicant’s technical basis for its aging management review and proposed aging management programs and compared it to the generic technical basis in NUREG-2214. For comparison to the generic technical basis in NUREG-2214, the staff ensured that the design features, environmental conditions, and operating experience for the Rancho Seco ISFSI are bounded by those evaluated in NUREG-2214.

This SER is organized into six sections. Section 1 provides the staff’s review of the general and financial information in the LRA. Section 2 presents the staff’s review of the scoping evaluation for determining which SSCs are within the scope of renewal. Section 3 includes the staff’s evaluation of the aging management review for the assessment of aging effects and aging management activities for SSCs within the scope of renewal. Section 4 documents the additions and changes to the license that resulted from the review of the LRA. Section 5 presents the staff’s conclusions from its review. Section 6 lists the references supporting the staff’s review and technical determinations.

1 GENERAL INFORMATION

1.1 Specific License Holder Information

The license renewal application (LRA) for the Rancho Seco independent spent fuel storage installation (ISFSI) includes general information on the specific license holder, Sacramento Municipal Utility District (SMUD, licensee, or applicant). The LRA includes the names and addresses of the applicants; a description of the business of the applicants and the State in which it is incorporated and does business; and the organization and management of the applicants, including the names, addresses, and citizenship of the directors and principal officers.

According to the applicant, SMUD was formed under the provisions of California's Municipal Utility District Act following a vote of the citizens of Sacramento in 1923. SMUD began operations in 1946 and is now the sixth largest community-owned electric utility in the Nation. SMUD is responsible for the acquisition, generation, transmission, and distribution of electric power to its service area with a population of approximately 1.4 million, including most of Sacramento County and small, adjoining portions of Placer and Yolo counties in California.

The U.S. Nuclear Regulatory Commission (NRC) staff finds that the applicant provided the information required in paragraphs (a)–(d) of Title 10 of the *Code of Federal Regulations* (10 CFR) 72.22, "Contents of Application: General and Financial Information."

1.2 Financial Qualifications

The LRA includes information on financial qualifications, in accordance with the requirements in 10 CFR 72.22(e), to show that the applicant will have the necessary funds available to cover estimated construction costs, estimated operating costs over the requested period of extended operation, and estimated decommissioning costs.

1.2.1 Independent Spent Fuel Storage Installation Construction Cost Estimate

The ISFSI pad is already constructed, and the licensee has indicated that there are no plans for its expansion. Therefore, no construction costs are associated with this license renewal.

1.2.2 Independent Spent Fuel Storage Installation Operating Cost Estimate

According to SMUD, the estimated operating costs for the Rancho Seco ISFSI are estimated to be $200 million (2017 dollars) for the duration of the 40-year license period of extended operation or approximately $5 million per year. The cost estimate for Rancho Seco considers factors such as ISFSI security, project management, cask maintenance, and equipment surveillance.

To evaluate the reasonableness of this estimate, the staff reviewed estimated ISFSI operations costs cited by various sources, including costs cited in a 2001 article, "Interim Storage of Spent Fuel in the United States," by Allison Macfarlane of the Massachusetts Institute of Technology, and two U.S. Government Accountability Office (GAO) reports that provide estimates of annual operations and maintenance costs for a centralized storage facility (2009), and safety and security system and annual operational costs for dry storage at a shutdown reactor site (2014).

The 2001 Macfarlane article estimates ISFSI operating costs at a shutdown reactor, with all spent fuel in dry storage, to be approximately $4 million per year. Accounting for inflation through 2017, the estimated cost would be approximately $5.7 million annually. The 2009 GAO report, "Nuclear Waste Management; Key Attributes, Challenges, and Costs for the Yucca Mountain Repository and Two Potential Alternatives," estimates the cost of annual operations and maintenance for a centralized storage facility (or centralized interim storage facility (CISF)). A CISF is larger than the Rancho Seco ISFSI. However, the staff cites the study in this safety evaluation report (SER) to further determine the reasonableness of operational cost data provided by SMUD for a spent fuel storage facility. The 2009 GAO report (page 54) estimates $8.8 million in operational costs for a CISF; accounting for inflation through 2017, the estimated cost would be approximately $10 million. The 2014 GAO report, "Spent Nuclear Fuel Management; Outreach Needed to Help Gain Public Acceptance for Federal Activities That Address Liability," estimates $2.5 to $6.5 million in annual operations and other related costs at a shutdown reactor site (page 52). Inflation from 2014 through 2017 is negligible, at approximately 5 percent during that time period.

In addition, the staff notes that the U.S. Department of Energy (DOE) report, FCRD-NFST-2015-000648, "Cost Implications of an Interim Storage Facility in Waste Management System," Revision 1, issued September 2016, assumes annual maintenance, security, and monitoring costs of $10 million for an ISFSI located at a shutdown reactor site.

Finally, for comparison, publicly available information from Connecticut Yankee, Maine Yankee, and Yankee Rowe Atomic Power Station decommissioned facilities indicates that the annual cost to operate each of the three ISFSIs was approximately $10 million (see http://www.connyankee.com, http://www.maineyankee.com, and http://www.yankeerowe.com).

Based on data and estimations from various publications and information reported by licensees, estimated annual costs to operate an ISFSI at a shutdown reactor are approximately $2.5 million to $10 million. Accordingly, the staff finds the estimated annual operating cost for the Rancho Seco ISFSI of $5 million, or $200 million in 2017 dollars over the requested period of extended operation, to be reasonable.

1.2.3 Independent Spent Fuel Storage Installation Operating Funds Availability

As stated by the applicant in the LRA, SMUD is an electric utility that generates or distributes electricity and recovers the cost of this electricity, either directly or indirectly, through rates established by the entity itself or by a separate regulatory authority. As previously stated, SMUD is the sixth largest community-owned electric utility in the Nation. In its submittal, the applicant stated that, as a municipal utility, SMUD, with an annual budget of over $1.7 billion, has the ability to recover costs of service, including ISFSI operating costs, through rates, if necessary.

1.2.4 Conclusion

Based on an analysis of the financial information as described in the ISFSI LRA, the NRC staff finds that SMUD has provided sufficient information for financial qualifications to engage in the proposed activities of the Rancho Seco ISFSI. The staff concludes that the applicant has demonstrated reasonable assurance that funding will remain available to cover the operating costs of the Rancho Seco ISFSI for the 40-year requested period of extended operation.

1.3 Decommissioning Funding Assurance

Under 10 CFR 72.30(c), each holder of, or applicant for, a license under 10 CFR Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste," must submit for NRC review and approval a decommissioning funding plan (DFP) containing information on how reasonable assurance will be provided that funds will be available to decommission its ISFSI. At the time of license renewal and at intervals not to exceed 3 years, the DFP must be resubmitted with adjustments as necessary to account for changes in decommissioning costs and the extent of contamination. The DFP must update the information submitted with the original or prior approved plan under 10 CFR 72.30(b). In addition, the DFP must specifically consider the effect of the following events on decommissioning costs:

- spills of radioactive material producing additional residual radioactivity in onsite subsurface material
- facility modifications
- changes in authorized possession limits
- actual remediation costs that exceed the previous cost estimate

The DFP must contain a detailed decommissioning cost estimate (DCE), in an amount reflecting the cost of an independent contractor to perform all decommissioning activities; an adequate contingency factor; and the cost of meeting 10 CFR 20.1402, "Radiological Criteria for Unrestricted Use" (or the cost of meeting 10 CFR 20.1403, "Criteria for License Termination under Restricted Conditions," provided the licensee can demonstrate its ability to meet these criteria). The licensee's DFP must also identify and justify using the key assumptions contained in the DCE. Further, the DFP must describe the method of assuring funds for ISFSI decommissioning, including the means for adjusting cost estimates and associated funding levels periodically over the life of the ISFSI. Finally, the DFP must specify the volume of onsite subsurface material containing residual radioactivity that will require remediation to meet the criteria for license termination and contain a certification that financial assurance for ISFSI decommissioning has been provided in the amount of the DCE.

According to the applicant, neither liquid spills of substances containing radioactive material, nor those that may come in contact with radioactive material, are considered credible at this stage of decommissioning, since the remaining radioactive material is in solid form and not dispersible. Because decommissioning tasks are only associated with dismantling any remaining facilities, no additional significant facility modifications are expected. In addition, no change in authorized possession limits for spent fuel is anticipated until the final 10 CFR Part 72 license termination, which occurs after the spent fuel is transferred to DOE and shipped off site for disposal.

Since SMUD provided the Rancho Seco 2015 DCE (SMUD, 2016a), the expected, remaining remediation costs have not changed. Further, no additional decommissioning costs are envisioned as a result of the implementation of aging management programs (AMPs) at the ISFSI, nor as a result of increases in the possession limits of byproduct strontium-90 source material (transferred from the license under 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to the 10 CFR Part 72 license). The only change in the decommissioning estimate is an inflationary adjustment for 2017. With available funds of

$8.41 million in the Decommissioning Trust Fund (DTF) as of December 31, 2017 (SMUD, 2018a), the remaining costs projected to complete decommissioning of $5.2 million can be paid from the available DTF balance. The staff finds that this information provides reasonable assurance of SMUD's ability to obtain the necessary funds to cover the remaining decommissioning costs for the Rancho Seco ISFSI.

1.4 Environmental Review

Regulations in 10 CFR 72.34, "Environmental Report," require that each application for an ISFSI license under this part must be accompanied by an environmental report that meets the requirements of Subpart A, "National Environmental Policy Act—Regulations Implementing Section 102(2)," of 10 CFR Part 51, "Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions." The applicant submitted an environmental report supplement as part of the LRA (SMUD, 2018b) that contained sufficient information to aid the staff in its independent analysis. In February 2020, the staff issued an environmental assessment (NRC, 2020) for the Rancho Seco ISFSI license renewal.

1.5 Safety Review

The objective of this safety review is to determine whether there is reasonable assurance that the ISFSI will continue to meet the requirements of 10 CFR Part 72 during the requested period of extended operation. Under 10 CFR 72.42(a), an application for ISFSI license renewal must include the following:

- time-limited aging analyses (TLAAs) that demonstrate structures, systems, and components (SSCs) important to safety will continue to perform their intended functions for the requested period of extended operation

- a description of the AMP for managing issues associated with aging that could adversely affect SSCs important to safety

The applicant stated that it prepared the LRA in accordance with the applicable provisions of 10 CFR Part 72 and NUREG-1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel," issued June 2016 (NRC, 2016). The applicant performed a scoping evaluation and aging management review (AMR) to identify all SSCs within the scope of the license renewal and pertinent aging mechanisms and effects, respectively. The applicant developed AMPs and evaluated TLAAs to ensure that the SSCs identified to be within the scope of renewal will continue to perform their intended functions during the period of extended operation. This review documents the staff's evaluation of the applicant's scoping analysis, AMR, and supporting AMPs and TLAAs.

1.6 Application Content

The applicant's LRA provided the following information:

- general and financial information
- scoping evaluation
- AMR
- AMPs
- TLAAs

- final safety analysis report (FSAR) supplement
- proposed license conditions
- environmental report supplement
- results of preapplication inspections and operating experience review
- ISFSI DFP

The FSAR supplement submitted by the applicant (in Revision 3 of the LRA, Appendix C) (SMUD, 2019c) proposed changes and additions to the Rancho Seco ISFSI FSAR to document the results of the scoping evaluation, AMR, TLAAs, and AMPs.

1.7 Evaluation Findings

The staff reviewed the information in the LRA, following the guidance in NUREG-1927, Revision 1. Based on its review, the staff determined that the applicant has provided sufficient information with adequate details to support the LRA, with the following findings:

F1.1 The information presented in the LRA satisfies the requirements of 10 CFR 72.22, “Contents of Application: General and Financial Information”; 10 CFR 72.30, “Financial Assurance and Recordkeeping for Decommissioning”; 10 CFR 72.34, “Environmental Report”; and 10 CFR 72.42, “Duration of License; Renewal.”

F1.2 The applicant has provided a tabulation of all supporting information and docketed material incorporated by reference in accordance with 10 CFR 72.42.

2 SCOPING EVALUATION

As described in NUREG-1927, Revision 1, a scoping evaluation is necessary to identify the SSCs subject to an AMR, where the effects of aging are assessed. The objective of this scoping evaluation is to identify SSCs meeting any of the following criteria:

(1) SSCs that are classified as important to safety, as they are relied on for one of the following functions:

- Maintain the conditions required by the regulations or the specific license to store spent fuel safely.
- Prevent damage to the spent fuel during handling and storage.
- Provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to public health and safety.

(2) SSCs that are classified as <u>not</u> important to safety but, according to the design bases, their failure could prevent the fulfillment of a function that is important to safety.

After determining the in-scope SSCs, the applicant screened the SSCs to identify and describe the subcomponents that support the SSCs' intended functions.

2.1 <u>Scoping and Screening Methodology</u>

In Section 2 of the LRA, the applicant performed a scoping evaluation and provided the following information:

- a description of the scoping and screening methodology for the inclusion of SSCs and SSC subcomponents in the scope of renewal review
- a list of sources of information used for the scoping evaluation
- descriptions of the SSCs
- a list of the SSCs identified to be within and outside the scope of renewal review and the basis for the scope determination

The staff reviewed the scoping process and results in the LRA. The following section discusses the staff's review and findings on the applicant's scoping evaluation.

2.1.1 Scoping Process

In Section 2.2.2 of the LRA, the applicant reviewed the following design-basis documents to identify SSCs with safety functions meeting either scoping criterion 1 or 2, as defined above:

- Rancho Seco ISFSI FSAR, Revision 6 (SMUD, 2016b)
- Special Nuclear Materials (SNM) License No. SNM-2510, Amendment No. 4, license and technical specifications (TS) (NRC, 2017a)

- Rancho Seco ISFSI SERs (NRC, 2017b)

Table 2.1-1 lists the SSCs included and excluded from the scope of renewal and identifies the scoping criterion met by each in-scope SSC.

Table 2.1-1 SSCs Within and Outside the Scope of License Renewal

SSCs	Criterion 1	Criterion 2	In-Scope
Dry shielded canister	Yes	N/A	Yes
Horizontal storage module	Yes	N/A	Yes
Transfer cask	Yes	N/A	Yes
Transfer cask lifting yoke and extensions	No	No	No
Spent fuel assemblies	Yes	N/A	Yes
ISFSI basemat	No	Yes[1]	Yes
ISFSI approach slab	No	No	No
Other transfer equipment	No	No	No
Auxiliary equipment	No	No	No
Miscellaneous equipment	No	No	No
Greater-than-Class-C waste	No	No	No

[1]The applicant stated that the basemat is not safety related; however, the applicant included the basemat within the scope of renewal because settlement of the basemat may affect the retrievability of the dry shielded canister.
N/A = Not applicable.

The staff reviewed the scoping results to determine whether the applicant's conclusions on the out-of-scope components accurately reflect the design-basis documentation in the Rancho Seco ISFSI FSAR.

The applicant stated that the fuel transfer and auxiliary equipment is not important to safety, and its failure would not prevent the fulfillment of any intended function supporting storage operations. The staff notes that, in Section 2.3.2.1 of the LRA, the applicant stated that the fuel transfer and auxiliary equipment will not be used for any loading operations, and it is subject to standard maintenance and repair before use for retrieving the dry shielded canister (DSC) from the horizontal storage module (HSM). As a result, the applicant did not include these SSCs in the scope of renewal. In response to a request for additional information (RAI), the applicant also stated that the lifting yoke and extensions were used to handle the transfer cask (TC) within the fuel/reactor building and were designed and procured as "safety-related" components as the licensee used them under 10 CFR Part 50 (SMUD, 2019a). The lifting yoke and extensions will not be used again since all spent fuel has been transferred into DSCs, and the Rancho Seco site has been decommissioned and its 10 CFR Part 50 license has been terminated. Therefore, the applicant did not include the lifting yoke and extensions within the scope of the renewal. The staff also notes that lifting yokes typically are procured and used under licensees' reactor operations, and thus they would not be expected to be tied to the design bases of the storage system. The staff reviewed Section 3.2 and Section 3.4.1 of the Rancho Seco ISFSI FSAR and confirmed that the fuel transfer and auxiliary equipment, including the lifting yoke and extensions, are considered not important to safety for storage purposes. The staff finds the applicant's determination that the fuel transfer and auxiliary equipment are not in the scope of renewal to be acceptable because this equipment has no safety function during storage operations and is not necessary for future operations at the Rancho Seco ISFSI. The staff confirmed that the applicant's determination is consistent with Section 2.4.3 of NUREG-1927, Revision 1, that SSCs associated with cask loading and unloading equipment are excluded from the scope of renewal.

The applicant stated that the ISFSI approach slab is not in the scope of renewal, as it is not important to safety and its failure would not affect the function of a component that is important to safety. The staff reviewed Section 3.4.1 of the Rancho Seco ISFSI FSAR and confirmed that the design bases of the ISFSI classifies the approach slab as not important to safety because it does not support the HSM. Therefore, its failure would not prevent the fulfillment of a safety function of the HSM loaded with a DSC. Based on its review of the applicant's design-basis documentation, the staff finds the applicant's determination that the approach slab is not in the scope of renewal to be acceptable because, consistent with the original safety class designation, it does not meet scoping criterion 2, as it is not relied upon for the fulfillment of an important safety function for an SSC that is important to safety.

The applicant stated that the ISFSI miscellaneous equipment, including security fences and gates, lighting, lightning protection, communications, and monitoring equipment, is not important to safety and its failure would not prevent the fulfillment of any intended function supporting storage operations. The staff reviewed Section 2.3.2.3 of the LRA and confirmed that the miscellaneous equipment is not important to safety. The staff finds the applicant's determination that the miscellaneous equipment is not in the scope of renewal to be acceptable because this equipment has no safety function during storage operations at the Rancho Seco ISFSI. The staff confirmed that the applicant's determination is consistent with Section 1.2 of NUREG-1927, Revision 1, that SSCs associated with the physical protection of the ISFSI are excluded from the scope of renewal.

The applicant stated that the greater-than-Class-C (GTCC) waste is not fuel-related material. The applicant stated that the GTCC waste is not important to safety and its failure (e.g., reconfiguration or deterioration) would not prevent the fulfillment of any intended function supporting storage operations. The staff reviewed Section 3.1.1 and Appendix C of the Rancho Seco ISFSI FSAR and Section 2.3.2.4 of the LRA and confirmed that the configuration of the GTCC waste is not considered in the ISFSI safety analyses. The staff also confirmed that the GTCC waste does not include any spent fuel assemblies (SFAs) or fissionable material. Therefore, the staff finds the applicant's determination that the GTCC waste is not in the scope of renewal to be acceptable because its failure would not impact any ISFSI safety function.

Based on its review, the staff finds that the applicant has identified the in-scope SSCs in a manner consistent with NUREG-1927, Revision 1, and, therefore, the staff finds the scoping results to be acceptable. The applicant screened the in-scope SSCs to identify and describe the subcomponents that support the SSCs' intended functions. Section 2.1.2 describes the SSC subcomponents within the scope of the renewal, and Section 2.1.3 covers those outside the scope of the renewal.

2.1.2 Structures, Systems, and Components Within the Scope of Renewal

Using the scoping process discussed in Section 2.2 of the LRA, the applicant identified four SSC groups to be within the scope of renewal, including four types of DSCs [fuel only (FO), fuel with control components (FC), failed fuel (FF), and GTCC], HSM, TC, and SFAs. Tables 2.1-2 through 2.1-5 tabulate the subcomponents that support the intended functions of the SSCs that are within the scope of renewal and thus subject to an AMR.

The staff reviewed the applicant's screening of the in-scope SSCs to identify subcomponents within the scope of renewal review. The staff also reviewed the applicant's responses to RAIs (SMUD, 2019a). The staff's review considered the intended function of the subcomponent, its safety classification or basis for inclusion in the scope of renewal review, and design-basis information in the Rancho Seco ISFSI FSAR. The staff notes that its review was informed by Chapter 4 of NUREG-2214, Revision 0, "Managing Aging Processes In Storage (MAPS) Report," issued July 2019 (NRC, 2019). Based on this review, the staff finds that the applicant screened the in-scope SSCs in a manner consistent with NUREG-1927, Revision 1, and, therefore, the staff finds the screening results for in-scope SSC subcomponents to be acceptable.

2.1.3 Structures, Systems, and Components Not Within the Scope of Renewal

The applicant reviewed the in-scope SSCs to identify and describe any subcomponents that do not support the SSCs' intended functions and thus do not require an AMR. The applicant provided a basis for the exclusion of these components in Section 2.3.2 of the LRA. Tables 2.1-5 and 2.1-6 through 2.1-8 tabulate these subcomponents.

The staff reviewed the applicant's screening of the out-of-scope SSCs. The staff also reviewed the applicant's responses to RAIs (SMUD, 2019a). The staff's review considered the intended function of the subcomponent, its safety classification or basis for exclusion in the scope of renewal, and design-basis information in the Rancho Seco ISFSI FSAR.

Based on this review, the staff finds that the applicant screened the out-of-scope SSCs in a manner consistent with NUREG-1927, Revision 1, and, therefore, the staff finds the screening results for out-of-scope SSC subcomponents to be acceptable.

2.2 <u>Evaluation Findings</u>

The NRC staff reviewed the scoping evaluation in the LRA, following the guidance in NUREG-1927, Revision 1. Based on its review, the staff finds the following:

F2.1 The applicant has identified all SSCs important to safety and SSCs the failure of which could prevent an SSC from fulfilling its safety function, in accordance with the requirements of 10 CFR 72.3, "Definitions"; 10 CFR 72.24, "Contents of application: Technical information"; 10 CFR 72.42; 10 CFR 72.120, "General considerations";10 CFR 72.122, "Overall requirements"; 10 CFR 72.124, "Criteria for nuclear criticality safety"; 10 CFR 72.126, "Criteria for radiological protection"; and 10 CFR 72.128, "Criteria for spent fuel, high-level radioactive waste, reactor-related greater than Class C waste, and other radioactive waste storage and handling," as applicable.

F2.2 The justification for any SSC determined not to be within the scope of the renewal is adequate and acceptable.

Table 2.1-2 DSC Subcomponents Within the Scope of Renewal

FO DSC	FC DSC	FF DSC	GTCC DSC
Cylindrical Shell	Cylindrical Shell	Cylindrical Shell	Cylindrical Shell
Outer Bottom Cover	Outer Bottom Cover	Outer Bottom Cover	Bottom Shield Plug
Grapple Ring	Lead Shielding	Key	Grapple Ring
Grapple Ring Support	Grapple Ring	Grapple Ring	Grapple Ring Support
Inner Bottom Cover	Grapple Ring Support	Grapple Ring Support	Top Shield Plug
Spacer Discs (Type "A," "B," and "C")	Inner Bottom Cover	Inner Bottom Cover	Outer Top Cover Plate
Guide Sleeve	Bottom Plug Post	Spacer Discs	Outer Bottom Cover Plate
Oversleeve	Spacer Discs (Type "A," "B," and "C")	Inner and Outer Support Plate	Siphon and Vent Port Cover Plate
Neutron Absorber Sheet	Guide Sleeve	Lead Shielding	Bottom Plate
Support Rod	Oversleeve	Bottom Plug Post	Basket Cylindrical Shell
Shear Key	Neutron Absorber Sheet	Siphon and Vent Block	
Extension Plate	Support Rod	Lifting Lug	
Key	Shear Key	Support Ring	
Siphon and Vent Block	Extension Plate	Inner Top Cover Plate	
Lifting Lug	Key	Outer Top Cover Plate	
Support Ring	Siphon and Vent Block	Siphon and Vent Port Cover Plate	
Top Shield Plug	Lifting Lug	Top Shield Plug Casing	
Bottom Shield Plug	Support Ring	Top Shield Plug Post	
Inner Top Cover Plate	Inner Top Cover Plate	Liner	
Outer Top Cover Plate	Outer Top Cover Plate	Flange Plate	
Siphon and Vent Port Cover Plate	Siphon and Vent Port Cover Plate	Shear Key	
Top and Bottom End Spacer Sleeve	Top Shield Plug Casing	Top Lid Cover Plate	
Spacer Sleeves (Type 1, 2, 3, 4, 5, 6)	Top Shield Plug Post	Bottom Lid Adapter Plate	
Stop Plate	Plate Stiffening	Top Lid Lifting Pintle	
Plate 0.085 Thick	Top and Bottom End Spacer Sleeve	Mesh, 6×6	
	Spacer Sleeves (Type 1, 2, 3, 4, 5, 6)	Washer Plate	
	Angle, 1-1/4 × 1-1/4 × ¼	Spacer Bar	
	Plate 1.25 × 1.25 × ¼	Cover Plate	
	Stop Plate	Side Lid Plate	
	Plate 0.085 Thick	Bottom Plug Top and Side Casing	
	Bottom Plug Top and Side Casing	Plate Stiffening	
	Plate Stiffening		

Table 2.1-3 HSM Subcomponents Within the Scope of Renewal

Subcomponent	Subcomponent Parts
Base Unit Assembly	HSM Base Walls and Floor Slab
Roof Slab Assembly	HSM Roof Slab
End and Rear Shield Walls	End and Rear Shield Walls
DSC Support Structure Assembly	Support Rail Beams and Cross Beams
	Support Rail Plate
	Support Structure Steel Rail Extension Plate, DSC Stop Plates, Stiffener Plates, Gussets, Mounting Plates, Base Plates, Support Plate, Wall Attachment Channel and Angles
	Tube Steel Leg Column
	Bolts
	Nuts
	Wall Attachment Hardware (Heavy Hex Bolt, Hardened Washer)
HSM Shielded Door Assembly	Steel Plates
	Encased Concrete Core
	Door Bolt
Canister Axial Retainer Assembly	Axial Retainer Rod, Mounting Plate, Bolts
Cask Docking Ring Assembly	Rings, Plates, Nelson Studs, Door Clamps, Hex Bolts
Heat Shield Assemblies	Roof and Side Wall Mounted Heat Shields
	ZEE Brackets
	Heat Shield Attachment Hardware (Rods, Nuts)
	Heat Shield Embedment Assemblies (Bolts, Sleeve Nuts)
Cask Restraint Embedment Assembly	Rods, Sleeve Nuts, Hexagonal Nuts
Wall and Floor Mounted Canister Support Structure Embedment Assembly	Bolt, Sleeve Nut
Roof Attachment Assembly	Roof Mounted, Wall Mounted Attachment Assemblies (Sleeve Nut)
End and Rear Shield Walls Attachment Hardware	Embedment Bolts, Sleeve Nuts
End Shield Wall Attachment Hardware	Embedment Bolts, Sleeve Nuts
End and Rear Shield Wall Attachment Hardware	Cast-In-Place Bolts, Nuts
	Tie Plate
HSM-to-HSM Spacer Channels	Spacer Channels
End and Rear Shield Wall Support Bolt Assembly	Shield Wall Support Bolt Assembly (Bolts, Nuts)
General Assembly	Hex Bolt

Table 2.1-4 TC Subcomponents Within the Scope of Renewal

Main Assembly	Onsite Transfer Arrangement/Accessories
Inner Shell	Castable Neutron Shielding Material
Bottom End Closure	Outer Plug Cover Plate
Bottom Structural Shell	Inner Plug Cover Plate
Top Structural Shell	Inner Plug Inside Sleeve
Top Flange	Bolt, 1-8UNC-2A
Gamma Shielding	Outer Plug Support Bracket
Upper Trunnion Plug Cover and Side Plate	Key Plug Cover Plate
Upper Trunnion Sleeve	Flat Hd Socket Cap Screw
Lower Trunnion Sleeve	Socket Hd Cap Screw
Pad Plate	Lower Trunnion
Bearing Block	Upper Trunnion
Tie Bar	Trunnion Back
NSP Top and Bottom Support Ring	Key Plug Side Plate
NSP Support Angle, Outer	Key Plug Bottom Plate
Rupture Plug	
Plugs	
Neutron Shield Shell	
Upper Trunnion Plug Bottom Plate	
Rails	
Castable Neutron Shielding Material	
Ram Closure Plate	
Top Closure Plate	
Screw, Cap Hd. Soc.	
Filler Plate	
Hardened Washer (3″ and 1.5″ outer diameter)	
Test Port Screw	
Test Vent and Drain Threaded Insert	
Vent and Drain Port Plug	
Test Port Plug	
Vent and Drain Port Screw	
Lower Trunnion Plug Cover Plate	
Lower Trunnion Plug Shield Block	
Screw, Flat Hd. Cap	
NSP Support Angle, Inner	
Screw Thread Insert (1″ and 2″)	
10 Gage Sheet	
Inner Shell	
Bottom End Closure	

Table 2.1-5 SFA Subcomponents Within and Not Within the Scope of Renewal

Within the Scope	Not Within the Scope
Fuel Cladding	Fuel Pellets
Spacer Grid Assemblies	Hold Down Spring and Upper End Plugs
Upper End Fitting and Nozzle	Control Components
Lower End Fitting and Nozzle	
Guide Tubes	

Table 2.1-6 DSC Subcomponents Not Within the Scope of Renewal

FO DSC	FC DSC	FF DSC	GTCC DSC
Siphon Tubing	Siphon Tubing	Siphon Tubing	Basket Alignment Key
Male Connector	Male Connector	Male Connector	Top Shield Plug Alignment Key
Quick Connect	Quick Connect	Quick Connect	O-Ring
			Backup O-Ring
			Swagelok Quick Connect
			Channel C12 × 20.7
			Channel Half
			Box Plate
			Washer Plate
			Screen Mesh
			Lifting Lug
			TS
			Angle
			Angle
			Plate

Table 2.1-7 HSM Subcomponents Not Within the Scope of Renewal

Subcomponent	Subcomponent Parts
Roof Slab Assembly	Tube, Coupling, Protective Plug
DSC Support Structure Assembly	Washer Plates, Hardened Washers
	Standard Washer, Hardened Washer
HSM Shielded Door Assembly	Steel Plate and Bar Anchors
Canister Axial Retainer Assembly	Axial Retainer Sleeve
	Hardened Washer
Cask Docking Ring Assembly	Standard Washer
Heat Shield Assemblies	Heat Shield Attachment Hardware (Washers)
Roof Attachment Assembly	Roof Attachment Angle, Stiffener Plate
	Roof Mounted, Wall Mounted Attachment Assemblies (Bolt)
	Roof and Wall Mounted Attachment Hardware and Embedment (Washers, Bolts, Studs, Nuts)
End and Rear Shield Walls Attachment Hardware	Pipe Sleeve, Angle, Studs
Inlet and Outlet Vents	Bird Screens Strip, Wirecloth
Shield Wall Support Assembly	Plates, Pipe, Tube
End and Rear Shield Wall Support Bolt Assembly	Standard Washer
General Assembly	Washer

Table 2.1-8 TC Subcomponents Not Within the Scope of Renewal

Main Assembly	Onsite Transfer Arrangement/Accessories
O-Rings, Top Closure Plate	Pipe, 1″ Sch. 40
O-Rings, Ram Closure Plate	Lifting Eye, Drop Forged
O-Ring (2″ inner diameter)	Plate, 1/2″ Thick
Impact Limiter Attachment Block	Hardened Washer
Vent and Drain Port Seal	Outer Plug Sleeve
Test Port Seal	Nut, 1/2-13 UNC-2B
Tapered Pin	Inner Plug Sleeve
Spacer Washer	
Tube, 1-1/2″ Sch. 40	

3 AGING MANAGEMENT REVIEW

3.1 Review Objective

The objective of the staff's evaluation of the applicant's AMR is to determine whether the applicant has adequately reviewed applicable materials, environments, and aging mechanisms and effects and proposed adequate aging management activities for in-scope SSCs. The AMR addresses aging mechanisms and effects that could adversely affect the ability of the SSCs and associated subcomponents to perform their intended functions during the period of extended operation.

3.2 Aging Management Review Process

The applicant described its AMR process as consisting of four steps:

(1) identification of materials and environments
(2) identification of aging mechanisms and effects requiring management
(3) determination of the activities required to manage the effects of aging
(4) evaluation of spent fuel (canister) retrievability during the period of extended operation

The applicant stated that the AMR provides an assessment of the aging effects that could adversely affect the ability of the SSCs to perform their intended functions during the period of extended operation. For each SSC, the applicant identified the material of construction and the environment to which each SSC is exposed. Then, the applicant identified applicable aging effects and associated aging mechanisms based on a review of engineering literature, related research and industry information, and existing operating experience. Finally, for each aging effect requiring management, the applicant identified an AMP or TLAA to ensure that the intended function of the SSC would be maintained during the period of extended operation.

The staff reviewed the applicant's AMR process, including a description of the review process and the design-basis references. Based on its review, the staff finds that the applicant's AMR process is acceptable because it is consistent with the methodology recommended in NUREG-1927, Revision 1, and adequate for identifying credible aging effects for the SSCs within the scope of renewal.

3.3 Aging Management Review Results: Materials, Service Environment, Aging Effects, and Aging Management Activities

Tables 3.3-1 through 3.3-6 show the results of the applicant's AMR and identify the AMPs or TLAAs that are credited to manage the aging mechanisms and effects for SSC subcomponents within the scope of renewal, as provided in its LRA.

Table 3.3-1 AMR Results—FO DSC

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Cylindrical Shell	1	Stainless Steel	Inert Gas/ Sheltered	Loss of Material, Cracking	3.3.1	3.4.1 3.5.1
Outer Bottom Cover	1	Stainless Steel	Sheltered/ Embedded	Loss of Material, Cracking	3.3.1	3.5.1
Grapple Ring	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Grapple Ring Support	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Inner Bottom Cover	1	Stainless Steel	Inert Gas/ Embedded	Cracking	3.3.1	3.4.1
Spacer Discs (Type “A,” “B,” and “C”)	1	Carbon Steel	Inert Gas	None Identified	N/A	N/A
Guide Sleeve	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Oversleeve	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Neutron Absorber Sheet	1	Boral®	Inert Gas	Loss of Criticality Control	3.3.1	3.4.3
Support Rod	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Shear Key	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Extension Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Key	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Siphon and Vent Block	1	Stainless Steel	Inert Gas/ Embedded	Cracking	3.3.1	3.4.1
Lifting Lug	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Support Ring	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Top Shield Plug	1	Carbon Steel	Inert Gas	None Identified	N/A	N/A
Bottom Shield Plug	1	Carbon Steel	Embedded	None Identified	N/A	N/A
Inner Top Cover Plate	1	Stainless Steel	Embedded/ Inert Gas	Cracking	3.3.1	3.4.1
Outer Top Cover Plate	1	Stainless Steel	Embedded/ Sheltered	Loss of Material, Cracking	3.3.1	3.4.1 3.5.1

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Siphon and Vent Port Cover Plate	1	Stainless Steel	Embedded	Cracking	3.3.1	3.4.1
Top and Bottom End Spacer Sleeve	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Spacer Sleeves (Type 1, 2, 3, 4, 5, 6)	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Stop Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Plate 0.085 Thick	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A

Table 3.3-2 AMR Results—FC DSC

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Cylindrical Shell	1	Stainless Steel	Inert Gas/ Sheltered	Loss of Material, Cracking	3.3.1	3.4.1 3.5.1
Outer Bottom Cover	1	Stainless Steel	Sheltered/ Embedded	Loss of Material, Cracking	3.3.1	3.5.1
Lead Shielding	1	Lead	Embedded/ Encased	None Identified	N/A	N/A
Grapple Ring	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Grapple Ring Support	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Inner Bottom Cover	1	Stainless Steel	Inert Gas/ Embedded	Cracking	3.3.1	3.4.1
Bottom Plug Post	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Spacer Discs (Type “A,” “B,” and “C”)	1	Carbon Steel	Inert Gas	None Identified	N/A	N/A
Guide Sleeve	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Oversleeve	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Neutron Absorber Sheet	1	Boral®	Inert Gas	Loss of Criticality Control	3.3.1	3.4.3
Support Rod	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Shear Key	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Extension Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Key	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Siphon and Vent Block	1	Stainless Steel	Inert Gas/ Embedded	Cracking	3.3.1	3.4.1
Lifting Lug	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Support Ring	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Inner Top Cover Plate	1	Stainless Steel	Embedded/ Inert Gas	Cracking	3.3.1	3.4.1
Outer Top Cover Plate	1	Stainless Steel	Embedded/ Sheltered	Loss of Material, Cracking	3.3.1	3.4.1 3.5.1

Subcomponent	**In-Scope Classification Criterion 1 or 2**	**Materials**	**Environment**	**Aging Effect**	**AMR SER Section**	**TLAA/ AMP SER Section**
Siphon and Vent Port Cover Plate	1	Stainless Steel	Embedded	Cracking	3.3.1	3.4.1
Top Shield Plug Casing	1	Carbon Steel	Inert Gas/ Embedded	None Identified	N/A	N/A
Top Shield Plug Post	1	Carbon Steel	Embedded	None Identified	N/A	N/A
Plate Stiffening	1	Carbon Steel	Embedded	None Identified	N/A	N/A
Top and Bottom End Spacer Sleeve	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Spacer Sleeves (Type 1, 2, 3, 4, 5, 6)	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Angle, 1-1/4 × 1-1/4 × ¼	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Plate 1.25 × 1.25 × ¼	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Stop Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Plate 0.085 Thick	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Bottom Plug Top and Side Casing	1	Stainless Steel	Embedded	None Identified	N/A	N/A
Plate Stiffening	1	Stainless Steel	Embedded	None Identified	N/A	N/A

Table 3.3-3 AMR Results—FF DSC

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Cylindrical Shell	1	Stainless Steel	Inert Gas/ Sheltered	Loss of Material, Cracking	3.3.1	3.4.1 3.5.1
Outer Bottom Cover	1	Stainless Steel	Sheltered/ Embedded	Loss of Material, Cracking	3.3.1	3.5.1
Key	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Grapple Ring	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Grapple Ring Support	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Inner Bottom Cover	1	Stainless Steel	Inert Gas/ Embedded	Cracking	3.3.1	3.4.1
Spacer Discs	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Inner and Outer Support Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Lead Shielding	1	Lead	Embedded/ Encased	None Identified	N/A	N/A
Bottom Plug Post	1	Stainless Steel	Embedded	None Identified	N/A	N/A
Siphon and Vent Block	1	Stainless Steel	Inert Gas/ Embedded	Cracking	3.3.1	3.4.1
Lifting Lug	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Support Ring	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Inner Top Cover Plate	1	Stainless Steel	Embedded/ Inert Gas	Cracking	3.3.1	3.4.1
Outer Top Cover Plate	1	Stainless Steel	Embedded/ Sheltered	Loss of Material, Cracking	3.3.1	3.4.1 3.5.1
Siphon and Vent Port Cover Plate	1	Stainless Steel	Embedded	Cracking	3.3.1	3.4.1
Top Shield Plug Casing	1	Carbon Steel	Inert Gas/ Embedded	None Identified	N/A	N/A
Top Shield Plug Post	1	Carbon Steel	Embedded	None Identified	N/A	N/A
Liner	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Flange Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Shear Key	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1

Subcomponent	**In-Scope Classification Criterion 1 or 2**	**Materials**	**Environment**	**Aging Effect**	**AMR SER Section**	**TLAA/ AMP SER Section**
Top Lid Cover Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Bottom Lid Adapter Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Top Lid Lifting Pintle	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Mesh, 6×6	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Washer Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Spacer Bar	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Cover Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Side Lid Plate	1	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Bottom Plug Top and Side Casing	1	Stainless Steel	Embedded	None Identified	N/A	N/A
Plate Stiffening	1	Stainless Steel	Embedded	None Identified	N/A	N/A
Plate Stiffening	1	Carbon Steel	Embedded	None Identified	N/A	N/A

Table 3.3-4 AMR Results—GTCC Waste DSC

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Cylindrical Shell	1	Stainless Steel	Inert Gas/ Sheltered	Loss of Material, Cracking	3.3.1	3.4.1 3.5.1
Bottom Shield Plug	1	Stainless Steel	Embedded/ Inert Gas/ Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Grapple Ring	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Grapple Ring Support	1	Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Top Shield Plug	1	Carbon Steel	Embedded/ Inert Gas	None Identified	N/A	N/A
Outer Top Cover Plate	1	Stainless Steel	Embedded/ Sheltered	Loss of Material, Cracking	3.3.1	3.4.1 3.5.1
Outer Bottom Cover Plate	1	Stainless Steel	Embedded/ Sheltered	Loss of Material, Cracking	3.3.1	3.5.1
Siphon and Vent Port Cover Plate	1	Carbon Steel	Embedded	Cracking	3.3.1	3.4.1
Bottom Plate	2	Stainless Steel	Inert Gas	None Identified	N/A	N/A
Basket Cylindrical Shell	2	Stainless Steel	Inert Gas	None Identified	N/A	N/A

Table 3.3-5 AMR Results—HSM and ISFSI Basemat

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
HSM Base Walls and Floor Slab	1	Reinforced Concrete	External/ Sheltered	Loss of Material, Cracking, Change in Material Properties	3.3.2	3.5.2
HSM Roof Slab	1	Reinforced Concrete	External/ Sheltered	Loss of Material, Cracking, Change in Material Properties	3.3.2	3.5.2
End and Rear Shield Walls	1	Reinforced Concrete	External/ Sheltered	Loss of Material, Cracking, Change in Material Properties	3.3.2	3.5.2
Support Rail Beams and Cross Beams	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2
Support Rail Plate	2	Nitronic® 60 Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.2	3.5.2
Support Structure Steel Rail Extension Plate, DSC Stop Plates, Stiffener Plates, Gussets, Mounting Plates, Base Plates, Support Plate, Wall Attachment Channel and Angles	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2
Tube Steel Leg Column	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2
Support Structure Bolts	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2
Support Structure Nuts	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2
Wall Attachment Hardware (Heavy Hex Bolt, Hardened Washer)	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Steel Plates	1	Carbon Steel	External	Loss of Material	3.3.2	3.5.2
Encased Concrete Core	1	Non-Shrink Grout	Embedded/ Encased	None Identified	N/A	N/A
Door Bolt	1	Carbon Steel	External	Loss of Material	3.3.2	3.5.2
Axial Retainer Rod, Mounting Plate, Bolts	1	Carbon Steel	External/ Sheltered	Loss of Material	3.3.2	3.5.2
Rings, Plates, Nelson Studs, Door Clamps, Hex Bolts	1	Carbon Steel	Embedded/ Encased/ External	Loss of Material	3.3.2	3.5.2
Roof and Side Wall Mounted Heat Shields	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2
ZEE Brackets	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2
Heat Shield Attachment Hardware (Rods, Nuts)	1	Carbon Steel	Sheltered	Loss of Material	3.3.2	3.5.2
Heat Shield Embedment Assemblies (Bolts, Sleeve Nuts)	1	Carbon Steel	Embedded/ Encased/ Sheltered	Loss of Material	3.3.2	3.5.2
Rods, Sleeve Nuts, Hexagonal Nuts	1	Carbon Steel	Embedded/ Encased/ Sheltered	Loss of Material	3.3.2	3.5.2
Bolt, Sleeve Nut	1	Carbon Steel	Embedded/ Encased/ Sheltered	Loss of Material	3.3.2	3.5.2
Roof Mounted, Wall Mounted Attachment Assemblies (Sleeve Nut)	1	Carbon Steel	Embedded/ Encased/ Sheltered	Loss of Material	3.3.2	3.5.2
Embedment Bolts, Sleeve Nuts	1	Carbon Steel	Embedded/ Encased/ Sheltered	Loss of Material	3.3.2	3.5.2
Cast-In-Place Bolts, Nuts	1	Carbon Steel	Embedded/ Encased/ External	Loss of Material	3.3.2	3.5.2
Tie Plate	1	Carbon Steel	External	Loss of Material	3.3.2	3.5.2
Spacer Channels	2	Carbon Steel	External	Loss of Material	3.3.2	3.5.2

Subcomponent	**In-Scope Classification Criterion 1 or 2**	**Materials**	**Environment**	**Aging Effect**	**AMR SER Section**	**TLAA/ AMP SER Section**
Shield Wall Support Bolt Assembly (Bolts, Nuts)	1	Carbon Steel	External	Loss of Material	3.3.2	3.5.2
Basemat	2	Reinforced Concrete	External/ Sheltered/ Embedded/ Soil	Loss of Material, Cracking, Change in Material Properties	3.3.3	3.5.4

Table 3.3-6 AMR Results—TC

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Inner Shell	1	Stainless Steel	Encased/ Sheltered	Loss of Material, Cracking	3.3.4	3.5.3
Bottom End Closure	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Bottom Structural Shell	1	Stainless Steel	Encased/ Sheltered	Loss of Material, Cracking	3.3.4	3.5.3
Top Structural Shell	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Top Flange	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Gamma Shielding	1	Lead	Encased	None Identified	N/A	N/A
Upper Trunnion Plug Cover and Side Plate	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Upper Trunnion Sleeve	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Lower Trunnion Sleeve	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Pad Plate	1	Stainless Steel	Encased	None Identified	N/A	N/A
Bearing Block	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Tie Bar	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
NSP Top and Bottom Support Ring	1	Stainless Steel	Encased/ Sheltered	Loss of Material, Cracking	3.3.4	3.5.3
NSP Support Angle, Outer	1	Stainless Steel	Encased	None Identified	N/A	N/A
Rupture Plug	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Plugs	1	Stainless Steel	Encased/ Sheltered	Loss of Material, Cracking	3.3.4	3.5.3
Neutron Shield Shell	1	Stainless Steel	Encased/ Sheltered	Loss of Material, Cracking	3.3.4	3.5.3
Upper Trunnion Plug Bottom Plate	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Rails	1	Nitronic® 60 Stainless Steel	Sheltered	Loss of Material, Cracking	3.3.4	3.5.3

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Castable Neutron Shielding Material	1	NS-3	Encased	Loss of material/ gas generation	3.3.4	3.4.5
Ram Closure Plate	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Top Closure Plate	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Screw, Cap Hd. Soc.	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Filler Plate	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Hardened Washer (3″ and 1.5″ outer diameter)	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Test Port Screw	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Vent/Drain Port Screw	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Threaded Insert and Port Plugs	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Lower Trunnion Plug Cover Plate	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Lower Trunnion Plug Shield Block	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Screw, Flat Hd. Cap	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
NSP Support Angle, Inner	1	Aluminum	Embedded	None Identified	N/A	N/A
Screw Thread Insert (1″ and 2″)	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
10-Gage Sheet	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Outer Plug Cover Plate	2	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Inner Plug Cover Plate	2	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Inner Plug Inside Sleeve	2	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Bolt, 1-8UNC-2A	2	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Outer Plug Support Bracket	2	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Key Plug Cover Plate	2	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Flat Hd Socket Cap Screw	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3

Subcomponent	In-Scope Classification Criterion 1 or 2	Materials	Environment	Aging Effect	AMR SER Section	TLAA/ AMP SER Section
Socket Hd Cap Screw	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Lower Trunnion	1	Stainless Steel	Sheltered	Loss of Material	3.3.4	3.5.3
Upper Trunnion	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Trunnion Back	1	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Key Plug Side Plate	2	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3
Key Plug Bottom Plate	2	Stainless Steel	Encased/ Sheltered	Loss of Material	3.3.4	3.5.3

3.3.1 Dry Shielded Canisters

The applicant described four types of DSCs in Section 2.4.1 of the LRA: (1) FO, (2) FC, (3) FF, and (4) GTCC waste. As specified in the application, each FO, FC, and FF DSC is a welded pressure vessel consisting of a stainless steel alloy cylindrical shell and top and bottom end assemblies, which form the pressure retaining confinement boundary for the spent fuel. Each FO, FC, and FF DSC is capable of holding multiple SFAs in an internal basket assembly in an inert helium atmosphere. The GTCC waste DSC uses a similar design and is constructed from a stainless steel alloy as specified in the application. The GTCC DSC has a modified internal basket assembly to accommodate the solid reactor-related waste form.

3.3.1.1 Materials and Environments

The applicant described the materials in LRA Sections 2.4.1 and 3.4.2 and Table 3-2, as well as the external and internal environments for the DSCs in LRA Section 3.4.3.

<u>Dry Shielded Canister Shell Assembly and External Environment</u>

The DSC shell assembly subcomponents are constructed of several materials, including stainless steel, carbon steel or coated carbon steel with electroless nickel, and lead. For FO, FC, and FF DSCs, the cylindrical shell, inner and outer top cover plates, inner and outer bottom cover plates, grapple ring, grapple ring support, siphon and vent block, siphon and vent cover plates, support ring, lifting lugs, reinforcing pads, and other miscellaneous parts are constructed of stainless steel. The applicant indicated that the top and bottom shield plugs for the FO DSC are constructed of carbon steel coated with electroless nickel unless it is encased. For the FC DSC, a lead shield plug with carbon steel casing forms the top shield plug assembly, and a lead shield plug with stainless steel casing forms the bottom shield plug assembly. For GTCC DSC shell assembly subcomponents, the cylindrical shell, outer top and bottom cover plate, bottom shield plug, grapple ring, and grapple ring support are constructed of stainless steel, and the siphon and vent port cover plates and top shield plug are constructed of carbon steel coated with electroless nickel.

After loading, each DSC is positioned for storage inside an HSM. The applicant stated that the external surfaces of the DSC (shell, top and bottom outer cover plates, grapple assembly, and associated welds) are exposed to a sheltered environment, which is a protected environment with no direct exposure to sun, wind, or precipitation but which may contain moisture and salts or other contaminants from the external ambient air. Using 21 degrees Celsius (C) (70 degrees Fahrenheit [F]) as the long-term normal ambient temperature, the applicant calculated the FO and FC DSC shell assembly temperatures for normal and off-normal conditions of storage inside the HSM to be 152 degrees C (305 degrees F) and 217 degrees C (423 degrees F), respectively, at the beginning of storage. The applicant reported that the DSC shell temperatures decrease continuously during the period of extended operation. The applicant also noted that the DSC shell assembly components are exposed to neutron and gamma radiation. The renewal application includes analyses that evaluate the effects of neutron fluence and gamma radiation on the mechanical properties of DSC shell assembly subcomponents. The evaluation takes credit for source strength decay over an assumed storage duration of 100 years, and the energy deposition is integrated over the same period. The applicant reported that the calculated neutron fluence on the DSC shell assembly is well below the level of concern for embrittlement of the stainless steel of 1×10^{18} neutrons per square centimeter (n/cm^2) (EPRI, 2007).

<u>Dry Shielded Canister Basket Assembly and Internal Environment</u>

Internal basket assembly subcomponents for FO, FC, and FF DSCs are constructed of three materials: (1) carbon steel coated with electroless nickel and stainless steel for spacer discs, support rods, and support plates, (2) stainless steel for guide sleeves, oversleeves, spacer sleeves, and FF Can, and (3) Boral® for neutron absorber plates, except in the FF DSC. The GTCC DSC basket assembly subcomponents, including the perforated basket shell, bottom plate, and lifting lugs, are constructed of stainless steel.

The applicant stated that the internal components of the DSC, including the basket assembly, other shell assembly components (such as inner top and bottom cover plates, shield plugs, siphon and vent block and cover plates), and associated welds are exposed to the inert gas (helium) environment inside the DSC cavity. The applicant analyzed the hypothetical impact of helium leaking from the DSC welded canister pressure boundary at the TS 3.1.2 DSC helium leakage rate of 10^{-5} std-cc/sec through the requested period of extended operation. The evaluation determined that the helium loss and potential ingress of air into the canister would be a small percentage of the initial helium inventory. The applicant concluded that these small changes will have a negligible impact on the inert environment in the DSC during the period of extended operation. The staff reviewed the applicant's analysis and determines the applicant's conclusion that leakage of helium over the requested period of extended operation will have a negligible impact on the inert environment of the DSC is reasonable, and therefore it concludes that the inert environment inside the DSC is maintained.

The applicant calculated that, at the beginning of storage under normal conditions, the average helium temperature inside the DSC cavity reaches 273 degrees C (524 degrees F) and the helium gas pressure is 26.9 kilopascals (3.9 pounds per square inch, gauge). Furthermore, the applicant stated that the helium gas temperature and pressure decrease over the period of extended operation. The applicant noted that the DSC internal components are exposed to neutron and gamma radiation, but the calculated maximum neutron fluence in the basket assembly center fuel compartments is well below the level of concern for embrittlement of the stainless steel of 1×10^{18} n/cm^2 (EPRI, 2007).

NRC Staff Review

The staff reviewed the applicant's description of the materials and environment for the FO, FC, FF, and GTCC DSCs and the DSC subcomponents to confirm that the description is consistent with the descriptions and engineering drawings in the Rancho Seco ISFSI FSAR. The staff evaluated the information provided by the applicant with respect to the neutron fluence interacting with the DSC components as part of the operating environment review. The staff finds the applicant's identification and description of the materials and environments for the DSCs to be acceptable because it is consistent with the descriptions and engineering drawings in the Rancho Seco ISFSI FSAR.

3.3.1.2 Aging Effects and Mechanisms for Dry Shielded Canister during the Period of Extended Operation

The applicant evaluated the materials of construction for DSC subcomponents, including stainless steel and carbon steel, that are subject to further AMR. The applicant considered aging effects of these materials that could, if left unmanaged, cause degradation of DSC subcomponents and result in loss of the component's intended function(s). The applicant's assessment of the aging effects that could cause loss of intended function(s) if left unmanaged include the following:

- loss of material
- cracking
- change in material properties
- change in dimensions due to creep
- loss of subcriticality control due to boron depletion

Using technical publications, NUREG/CRs, and Electric Power Research Institute (EPRI) reports, the applicant then determined the range of possible aging mechanisms that could lead to aging effects for stainless steel and carbon steel. The applicant evaluated each aging mechanism to determine whether it could lead to an aging effect requiring management.

For loss of material, the applicant evaluated the following aging mechanism and material combinations:

- loss of material due to general corrosion—carbon steel
- loss of material due to crevice and pitting corrosion—carbon steel and stainless steel
- loss of material due to galvanic corrosion—dissimilar metals

For cracking, the applicant evaluated the following aging mechanism and material combinations:

- cracking due to stress-corrosion cracking (SCC)—stainless steel
- cracking due to thermal fatigue—carbon steel and stainless steel

For change in material properties, the applicant evaluated the following aging mechanism and material combinations:

- change in material properties due to thermal aging—carbon steel and stainless steel

- change in material properties due to irradiation embrittlement—carbon steel and stainless steel

For change in dimensions due to creep, the applicant evaluated carbon steel, stainless steel, and Boral®.

For loss of subcriticality control due to boron depletion, the applicant evaluated Boral®.

In addition to the degradation effects and mechanisms described above, the applicant also considered the following for the DSC:

- coating degradation
- effects of temporary attachments during fabrication on SCC

Loss of Material Due to General Corrosion

The applicant indicated that general corrosion can occur when carbon steel surfaces are in contact with moist air or water. The applicant noted that several factors govern the rate of general corrosion, such as the moisture of the air, the salinity level of the air, and the temperature of the metal surface. The applicant stated that the carbon steel subcomponents of the DSC, including the top and bottom shield plugs and other top shield plug assembly parts, are completely enclosed within a DSC cavity that has been dried and backfilled with helium. Based on the inert nature of the helium environment inside the DSC, the applicant concluded that loss of material due to general corrosion of carbon steel is not an aging effect requiring management for DSCs.

The staff reviewed the applicant's assessment of general corrosion of carbon steel used in internal components of the DSC, as well as the description of the internal environment. Freely exposed carbon steel surfaces in contact with moist air or water are subject to general corrosion. The corrosion rate depends on humidity, time of wetness, solution composition, pH, and temperature. The iron Pourbaix diagram shows that iron undergoes active corrosion forming iron $(Fe)^{2+}$ or Fe^{3+} ions at pH values lower than 8.5 to 9 as long as water is present (Kodama, 2005). However, very little residual water is present in internal environments following drying and refilling with inert gas, and thus corrosion of carbon steel will be limited. Jung et al. (2013) showed that the relative humidity inside the system after drying is no more than 5 percent at the beginning of storage and is less than 0.5 percent in 60 years. Furthermore, some steel subcomponents are coated by electroless nickel, which is more corrosion resistant than carbon steel. Therefore, the staff finds the applicant's assessment that no aging management activity is required for general corrosion of carbon steel inside the DSC to be acceptable.

Loss of Material Due to Crevice Corrosion and Pitting Corrosion

The applicant stated that crevice corrosion is a form of localized corrosion that occurs in shielded spaces or crevices created by component or part connections such as lap joints, splice plates, bolt threads, the underside of bolt heads, or points of contact between metals and nonmetals. A number of factors influence crevice corrosion, including electrolyte composition and flow, the geometry of the occluded region, and the concentration of dissolved oxygen within the occluded region. The applicant stated that, while atmospheric pollutants and contaminants are typically insufficient to promote crevice corrosion, alternating wetting and drying is particularly harmful because this leads to concentrated atmospheric pollutants and contaminants.

The applicant described pitting corrosion as another localized corrosive attack in aqueous environments containing dissolved oxygen and halides such as chlorides and bromides. As with crevice corrosion, the applicant noted that areas in which aggressive species can concentrate—that is, locations of frequent or prolonged wetting or of alternate wetting and drying—are particularly susceptible to pitting. The applicant noted that pitting corrosion is more common with passive materials, such as 300 Series austenitic stainless steels, than with nonpassive materials, such as carbon steels.

The applicant stated that the DSC is located within the sheltered environment of the HSM interior, and that air heating as a result of decay within the DSC will prevent the accumulation or condensation of moisture inside the HSM. The applicant stated, however, that because the DSC decay heat will decrease during the period of extended operation, the presence of moist air that is potentially aggressive in contact with the DSC needs to be considered. Furthermore, the applicant stated that the interaction of gamma radiation with water, which could be present on the canister surfaces as a result of deliquescence of deposited salts, can generate radiolytic oxidizing products, such as hydrogen peroxide and nitric acid. The applicant provided references (Glass et al., 1986; Marsh et al., 1986; Worthington et al., 1989; Furuya et al., 1984) to indicate that gamma radiation dose rates on the order of 10^5 and 10^6 rad per hour (rad/h) generated radiolytic oxidizing products to initiate pitting corrosion and crevice corrosion for unstressed specimens. However, the applicant showed that the maximum dose rate on the DSC shell is much less than the cited gamma radiation dose rates. Because of lack of data to demonstrate the effect of these radiation levels on corrosion, the applicant stated that a conservative approach is taken to assume that the radiolytic products could contribute to the corrosive environment that initiates corrosion, including pitting corrosion and crevice corrosion.

Considering the possible presence of a corrosive environment, the applicant concluded that crevice corrosion and pitting corrosion are aging mechanisms potentially operative on the external surface of the stainless steel DSC shell assembly, and loss of material due to crevice corrosion and pitting corrosion in stainless steel is an aging effect requiring management. The applicant stated that the DSC steel subcomponents located inside the DSC cavity are in an inert environment and are not subject to crevice corrosion and pitting corrosion.

The staff reviewed the applicant’s assessment of crevice corrosion and pitting corrosion of the external surface of the stainless steel DSC shell assembly, as well as the description of the external environment inside the HSM. Electrolytes conducive to crevice corrosion and pitting corrosion exist in the sheltered environment. Although crevice corrosion and pitting corrosion of stainless steel may not lead to significant material loss to DSC subcomponents because of limited penetration rates (Davison et al., 1987; NRC, 2014a; Morrison, 1972), crevice corrosion and pitting corrosion are known to be precursors to SCC (He et al., 2014), which can lead to significant damage. Therefore, the staff finds the applicant’s assessment that aging management activity is required for crevice corrosion and pitting corrosion of the external surface of the stainless steel DSC shell assembly to be acceptable.

The staff also reviewed the applicant’s assessment of crevice corrosion and pitting corrosion of steel used in internal components of the DSC, as well as the description of the internal environment. Because very little residual water is inside the DSC following drying, steel corrosion will be limited. Jung et al. (2013) showed that the relative humidity inside the system is no more than 5 percent at the beginning of storage and less than 0.5 percent in 60 years. Furthermore, there is a lack of oxygen and halides inside the DSC to promote crevice corrosion or pitting corrosion. Therefore, the staff finds the applicant’s assessment that no aging management activity is required for crevice corrosion and pitting corrosion of steels inside the DSC to be acceptable.

Loss of Material Due to Galvanic Corrosion

The applicant identified the potential for galvanic corrosion of the DSC shell as a result of contact with the graphite lubricant used on the DSC support structure rail faces because the graphite lubricant has relatively noble (electropositive) potential compared to the stainless steel materials (i.e., when graphite and stainless steel are coupled, the less reactive graphite will become the cathode to drive corrosion of the more reactive stainless steel). While the applicant stated that, even though there is no reported operating experience of stainless steel degradation where a graphite lubricant film causes galvanic corrosion, loss of material due to galvanic corrosion of the DSC shell is an aging effect requiring management for a stainless steel DSC shell in a sheltered environment.

The staff reviewed the materials of construction for the DSC shell and the DSC support rails with a graphite lubricant. Electrolytes formed by the deliquescence of atmospheric deposits or by the formation of condensation and dissolution of deposits that may induce corrosion could exist in a sheltered environment. For the stainless steel and graphite galvanic couple, stainless steel is expected to be anodic, and corrosion of the stainless steel could occur. Therefore, the staff finds the applicant's assessment that aging management activity is required for galvanic corrosion of the external surface of the stainless steel DSC shell assembly to be acceptable.

Cracking Due to Stress-Corrosion Cracking

The applicant stated that SCC is a localized nonductile cracking failure resulting from an unfavorable combination of applied or residual tensile stresses, material condition, and the presence of a corrosive environment. The applicant acknowledged that SCC can be a significant phenomenon that occurs in austenitic stainless steels if tensile stress and a corrosion environment exist. The applicant referenced the work of EPRI (2007) to show that dissolved oxygen, sulfates, fluorides, and chlorides can provide the necessary environment for SCC to occur. However, based on the NUHOMS system operating experience history and other generic industry experience, the applicant stated that SCC has not been caused by factors other than chloride. (The applicant evaluated chloride-induced SCC following the evaluation of SCC and concluded that the chloride concentration is not sufficiently high to cause chloride-induced SCC.) The applicant provided references (Glass et al., 1986; Marsh et al., 1986; Worthington et al., 1989; Furuya et al., 1984) to indicate that gamma radiation dose rates in the order of 10^5 and 10^6 rad/h generated radiolytic oxidizing products to initiate SCC for stressed specimens. However, the applicant showed that the maximum dose rate on the DSC shell is only in the order of 10^4 rad/h and decreases to 10^3 rad/h (SMUD, 2019a, c). Because of lack of data to demonstrate the effect of these radiation levels on SCC, the applicant stated that a conservative approach is taken to assume that the radiolytic products could contribute to the corrosive environment that initiates SCC of the stainless steel canister during storage. Considering uncertainties of some factors that could cause SCC, the applicant concluded that SCC is an aging mechanism potentially operative on the external surface of the stainless steel DSC shell assembly and that cracking due to SCC is an aging effect requiring management.

The staff reviewed the materials and the welds for the DSC shell assembly, as well as the description of the sheltered environment inside the HSM. All austenitic grades, especially Types 304 and 304L, have long been reported to be susceptible to SCC, especially in the presence of chloride in the normal wrought condition (Grubb et al., 2005; Morgan, 1980;

Kain, 1990). This susceptibility increases when the material is sensitized (He et al., 2014). In the welded condition, the heat-affected zone (HAZ), which is a thin band located adjacent to the weld, can be sensitized by the precipitation of carbides that extract chromium out of the metal matrix. For DSC subcomponents, the shell is welded, and welds also exist in other subcomponents. Analyses by EPRI (2013) concluded that the driving stress for SCC of a welded canister is expected to be weld residual stress. Weld residual stress modeling conducted by the NRC (2013) also indicated that through-wall tensile stresses of sufficient magnitude to support SCC are likely to exist in the weld HAZ. Because sufficient weld residual stresses and more susceptible material conditions are present near the welds, and aqueous electrolytes conducive to SCC are present in a sheltered environment, the staff finds that the potential for SCC of the welds in the canister shell and other stainless steel subcomponents is present, and it finds the applicant's assessment that aging management activity is required for SCC of the external surface of the stainless steel DSC shell assembly to be acceptable.

Cracking Due to Chloride-Induced Stress-Corrosion Cracking

The applicant referenced operating experience documented in NRC Information Notice 2012-20, "Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters," issued November 2012 (NRC, 2012b), to show that austenitic stainless steels are susceptible to SCC in chloride-containing environments under tensile stresses. The applicant understood that, as the DSC surface temperature decreases, airborne chloride salts from seawater or salted roads deposited on DSC surfaces via the HSM external vents could deliquesce at high relative humidity and form chloride-rich deliquescent brines to initiate chloride-induced stress-corrosion cracking (CISCC). The applicant stated that initiation of CISCC is affected by three major environmental parameters: (1) canister surface temperature, (2) relative humidity, and (3) accumulated chloride salt concentration on the DSC surface. The applicant conducted a susceptibility evaluation based on the criteria developed by EPRI (2015) and showed that the Rancho Seco ISFSI has a CISCC susceptibility ranking of "2" on the scale of 1–10, with "10" being the highest susceptibility. The applicant stated that the low CISCC susceptibility is supported by low chloride concentrations analyzed from surrogate chloride samples and the lack of aging-related degradation during preapplication inspection. Based on the results of the susceptibility evaluation, the applicant determined that cracking due to CISCC is not an aging effect requiring management for the external surface of the stainless steel DSC shell assembly.

The staff reviewed the CISCC susceptibility evaluation based on the criteria developed by EPRI (2015). The ranking factor ranging from 1 to 10 is calculated from three parameters: (1) the initial ranking based on the distance from the ISFSI to a marine shore, (2) an adjustment factor to account for local sources of chloride, such as a cooling tower and salted roads, and the elevation of the ISFSI pad, and (3) another adjustment factor accounting for annual average absolute humidity. The staff verified the calculation of the CISCC susceptibility ranking based on detailed information in Enclosure 5 of the applicant's response to a request for supplemental information (SMUD, 2018c).

The staff also reviewed the results of analyses of samples obtained from several ISFSI locations. Bryan and Schindelholz (2017) reported that concentrations of chloride, the aggressive component in the salts, were 50 milligrams per square meter or less, representing only a tiny fraction of the total solutes present in samples from Calvert Cliffs Nuclear Power Plant collected in 2017. Bryan and Schindelholz (2017) reported that, in seawater (and in initially formed sea salt aerosols), the molar ratio of chloride to sodium is equal to 1.16. The samples collected from Calvert Cliffs in 2017 had chloride-to-sodium ratios ranging from 0.15 to

0.25. Bryan and Schindelholz (2017) concluded that, if the chloride were deposited as sea salts, then the salt particles would have partially undergone particle-gas conversion reactions before or after deposition. These reactions convert chloride-rich sea salts to nitrate and sulfate minerals and, by reducing the chloride load on the canister surface, these reactions reduce the risk of canister SCC. In addition to the nitrate-to-chloride ratios, the collected samples had large amounts of sulfate and phosphate salts. Similar results were obtained on samples collected at Diablo Canyon Power Plant and Hope Creek Generating Station (Bryan and Enos, 2014) and more recently in samples collected from Maine Yankee (Plante, 2018).

The staff notes that, in chloride-containing solutions, nitrate is an effective inhibitor of localized corrosion on stainless steels (Cragnolino and Sridhar, 1991; King et al., 2016; Cook et al., 2017). The summary compiled by King et al. (2016) suggests that nitrate can be an effective inhibitor even at much lower nitrate-to-chloride ratios observed in the samples collected at Calvert Cliffs, Hope Creek, Diablo Canyon, and Maine Yankee. Recent results reported by Cook et al. (2017) show that nitrate can be an effective inhibitor for localized corrosion of stainless steels even in the presence of magnesium chloride. Very low magnesium concentrations were observed in the samples from Calvert Cliffs, Hope Creek, and Diablo Canyon. No magnesium results were reported from the samples obtained from Maine Yankee but, given that the site is 12 miles from the open ocean, the concentration of magnesium salts is expected to be similar to the other samples.

The staff notes that deposits collected at other ISFSI locations with a higher susceptibility ranking do not show the presence of high chloride concentrations, and the chemistry of the deposits analyzed were generally not consistent with a marine salt composition that has been shown to induce CISCC in stainless steels. Although the analysis of the low CISCC susceptibility of the Rancho Seco site using the EPRI susceptibility assessment criteria (EPRI, 2015) is in agreement with the results of samples obtained at other ISFSIs, including some that are much closer to the open ocean, the staff notes that there is insufficient data to use the EPRI susceptibility assessment criteria to determine that cracking due to CISCC is not a credible aging effect requiring management. The staff expects that the operating experience will increase with additional renewals and allow such an assessment in the future. Therefore, the staff does not agree with the applicant's exclusion of cracking due to CISCC as a credible aging effect requiring management. However, the staff determined that the DSC External Surfaces AMP (reviewed in SER Section 3.5.1) is sufficient to address aging effects on the DSC external surfaces, including the potential for cracking due to CISCC.

Cracking Due to Thermal Fatigue

The applicant stated that thermal fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading associated with thermal cycling. The applicant stated that the only source of potential thermal fatigue of the DSC is ambient seasonal and daily temperature fluctuation. However, the applicant stated that, because of its large mass, the DSC does not experience the full amplitude of ambient temperature cycles, and a gradual, long-term temperature decrease occurs during the course of storage. The applicant evaluated the DSC pressure and temperature fluctuations in accordance with the provisions of NB-3222.4(d) of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) (ASME, 1992), considering bounding conditions. The applicant stated that, as provided by NB-3222.4(d), fatigue effects need not be specifically evaluated provided the six criteria in NB-3222.4(d) are met. Because the six criteria in NB-3222.4(d) are met, the applicant concluded that cracking due to thermal fatigue is an aging effect managed through a TLAA.

The staff reviewed the applicant's calculations and the criteria from NB-3222.4(d) and finds that, because the criteria are met, thermal fatigue is not an aging effect requiring management for DSC subcomponents. SER Section 3.4.1 documents the staff's review of the TLAA on the thermal fatigue of DSC subcomponents.

Change in Material Properties Due to Thermal Aging

The applicant stated that the maximum DSC internal temperatures are limited by the cladding temperature limit at the beginning of storage. In addition, the applicant stated that the DSC subcomponent initial temperatures for normal conditions of storage were within the temperature limits allowed by the ASME Code or were evaluated to show they can perform their safety function. The applicant analyzed the potential for thermally induced degradation of the DSC welds by embrittlement of the delta ferrite phase. The applicant cited the results published in NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds," issued May 1996 (Gavendra et al., 1996), to show that full embrittlement of the delta ferrite phase does not lead to significant embrittlement of the entire weld, as confirmed by measured values of fracture toughness. The applicant also cited Chandra et al. (2012b) to show that, even though the fracture toughness of the stainless steel welds in the highest temperature zone near the center of the basket may have decreased, it is still in the range required by the ASME Code. The applicant also cited the spent fuel storage demonstration tests using a Castor V/21 (Bare et al., 2001), which showed no evidence of stainless steel basket weld degradation due to thermal aging after 14 years with fuel in the dry storage cask. The applicant therefore concluded that thermal aging embrittlement of the fuel basket weld is not expected to affect the safety function or performance of the basket structure.

The applicant stated that the temperatures of the DSC shell at normal and off-normal conditions of storage inside the HSM were 152 degrees C (305 degrees F) and 217 degrees C (423 degrees F), respectively, at the beginning of storage. The applicant cited Gavendra et al. (1996) and Chandra et al. (2012a, b) to show that this DSC shell temperature is below the embrittlement saturation temperature of 400 degrees C (752 degrees F) and the thermal aging embrittlement temperature of 335 degrees C (635 degrees F). The applicant also cited NUREG-1801, "Generic Aging Lessons Learned (GALL) Report," issued December 2010 (NRC, 2010a), which states that austenitic steels with service temperatures below 250 degrees C (482 degrees F) are not susceptible to thermal aging embrittlement. Because the maximum DSC shell temperature is below the thermal aging embrittlement temperatures reported in the literature, the applicant concluded that, for the external surface of the stainless steel DSC shell assembly, a change in material properties due to thermal aging is not an aging effect requiring management during the period of extended operation.

The applicant acknowledges that thermal aging of carbon steel could affect the mechanical properties, depending on the time exposed at the temperature and the microstructure and carbon content of the steel subcomponents. However, the applicant stated that any tempering effect that could occur after storage for 20 years would not be dominant compared to what occurred during manufacturing and the initial 20-year storage period. Therefore, the applicant concluded that a change in material properties of carbon steel due to thermal aging is not an aging effect requiring management during the period of extended operation.

The staff reviewed the applicant's assessment of the potential for thermal aging embrittlement of stainless steel and carbon steel. The ferrite present in austenitic stainless steel welds can transform by spinodal decomposition to form Fe-rich alpha and chromium-rich alpha prime phases, and further aging can produce an intermetallic G-phase. These phase transformations take place during extended exposure to temperatures between 300 and 400 degrees C (572 and 752 degrees F) (Alexander and Nanstad, 1995; Chandra et al., 2012a) and increase the hardness and reduce the toughness of the ferrite phase. However, they do not alter the mechanical properties of the austenite phase. Subcomponents located inside the canister and near the fuel could be above the 300 degrees C (572 degrees F) minimum temperature required for these phase changes. Based on Charpy impact toughness testing of cast duplex stainless steels, Kim and Kim (1998) concluded that ferrite levels above 15 percent are required for significant embrittlement, because ferrite resides in discrete islands below this level and does not provide a continuous low-toughness fracture path. Because most welds contain around 4- to 15-percent ferrite (Gavendra et al., 1996), substantial embrittlement of austenitic stainless steel welds is not expected. Gavendra et al. (1996) concluded that thermal aging produced moderate decreases (no more than 25 percent) in the upper shelf Charpy impact energy and relatively small decreases in the fracture toughness of a wide range of austenitic welds. The DSC shell temperatures are too low to be adversely affected during storage. For carbon steel, the staff determines that the steel tempering that occurs during manufacture and the higher temperatures present during the initial storage period would dominate any effects of tempering at the lower temperatures during the period of extended operation. Therefore, the staff finds the applicant's assessment that no aging management activity is required for thermal aging of steel components to be acceptable.

Change in Material Properties Due to Irradiation Embrittlement

The applicant stated that high neutron radiation could cause loss of fracture toughness in steel. The applicant provided analyses to evaluate the maximum neutron fluence and gamma radiation levels the DSC subcomponents could experience over an assumed storage duration of 100 years. The applicant demonstrated that these calculated values are well below the level of concern for embrittlement of stainless steel, reported to be 1×10^{18} n/cm^2 (EPRI, 2007); therefore, the applicant concluded that a change in material properties due to irradiation embrittlement is not an aging effect requiring management for DSC subcomponents. The applicant further stated that gamma radiation does not have any significant impact on the properties of steel.

The staff recognizes that neutron irradiation has the potential to increase the tensile and yield strength and decrease the toughness of carbon and alloy steels (NRC, 2019). Neutron fluence levels greater than 10^{19} n/cm^2 are required to produce a measurable degradation of the carbon steel mechanical properties (Nikolaev et al., 2002; Odette and Lucas, 2001). Cracking of stainless steel has been observed in boiling-water reactor oxygenated water at fluences above 2×10^{20} n/cm^2 to 5×10^{20} n/cm^2 (Was et al., 2006). Gamble (2006) found that neutron fluence levels greater than 1×10^{20} n/cm^2 are required to produce measurable degradation of the mechanical properties of stainless steel. Caskey et al. (1990) also indicated that neutron fluence levels of up to 2×10^{21} n/cm^2 were not found to enhance SCC susceptibility of stainless steel.

For a dry storage system (DSS), a neutron flux of 10^4–10^6 n/cm^2-s is typical (Sindelar et al., 2011). At these flux levels, the accumulated neutron fluence after 60 years is about 10^{13}–10^{15} n/cm^2. To verify the conservatism of this estimate, the staff independently calculated the maximum potential accumulated neutron fluence on DSS components. The staff considered components most directly exposed to the radiation source (middle of the fuel basket) and assumed fuel is loaded immediately after it is removed from the reactor vessel and stored for 100 years. To further provide a bounding estimate, the staff assumed a DSS design that uses 40 Westinghouse 17×17 pressurized-water reactor fuel assembles with an average burnup of

70 gigawatt-days per metric ton of uranium (GWd/MTU) and 4.0 fuel enrichment. The staff calculated the neutron source term for neutrons with energy at or greater than 1 million electron volts using the Origen/Arp computer code of the SCALE 6.1 computer code system. At this location, the total accumulated neutron fluence after 100 years of storage was calculated to be 2.63×10^{16} n/cm^2. This worst-case estimate is greater than that calculated using the flux levels reported in Sindelar et al. (2011); however, the staff's estimated fluence level is still three to four orders of magnitude below the levels reported to degrade the fracture resistance of carbon and alloy steels and stainless steels. Furthermore, as discussed in SER Section 3.4.4, the staff independently verified the applicant's calculation of maximum neutron fluence and gamma radiation levels and found the applicant's calculation and results to be conservative. Therefore, the staff finds the applicant's assessment that aging management activity is not required for radiation embrittlement of steel subcomponents exposed to sheltered and helium environments to be acceptable.

Change in Dimensions Due to Creep

The applicant stated that creep is a time-dependent strain (deformation) in metals occurring under constant load at constant temperature. The applicant stated that an increase in either stress or temperature accelerates creep and if stress or temperature is increased beyond certain levels, the increased deformation can eventually result in failure. The applicant stated that metallic materials are generally considered to be subject to creep under conditions of extended exposure to stress and temperature in excess of a homologous temperature of $0.4T_m$, where T_m is the melting point in degrees Kelvin (K). Based on the melting points of 1,516 degrees C (1,789 K or 2,760 degrees F) and 1,425 degrees C (1,698 K or 2,597 degrees F) for carbon steel and stainless steel, respectively, the applicant calculated that at least 443 degrees C (716 K or 829 degrees F) and 406 degrees C (679 K or 763 degrees F), respectively, are required to initiate creep. The applicant stated that the maximum DSC internal temperatures are limited by the cladding temperature limit at the beginning of storage, which is 379 degrees C (714 degrees F). As a result, the applicant concluded that creep is not an aging mechanism requiring management for the carbon steel and stainless steel subcomponents in the DSC.

The applicant stated that the DSC assembly does not contain any aluminum subcomponents, other than Boral® as neutron absorber plates. The applicant cited the analysis in NUREG-2214 to show that creep is not an aging mechanism requiring management for Boral® because the absorber plates do not serve a structural function and are not expected to be under loads other than their own weight.

The staff verified the results from the general rule of thumb of $0.4T_m$ (Cadek, 1988) used by the applicant to assess the potential for creep. The staff recognizes that the highest temperatures within the DSCs are at locations close to the fuel rods, and the maximum expected temperature of fuel cladding has been estimated to be 400 degrees C (752 degrees F) at the beginning of storage (Jung et. al., 2013). This cladding temperature is expected to decrease to around 266 degrees C (510 degrees F) after 20 years and to approximately 127 degrees C (261 degrees F) after 60 years. These estimates depend on many factors, such as the initial heat load of the canister and thermal decay constant. Because the fuel rods are the only heat source within the system, these temperatures provide upper temperature limits for all subcomponents. It is apparent from these temperatures that internal subcomponents will not approach the minimum 443 degrees C (829 degrees F) and 406 degrees C (763 degrees F) that have been found to be required for significant creep to occur in carbon steel and stainless steel, respectively. Furthermore, temperatures above 500 degrees C (932 degrees F) have been reported to be required for creep in carbon steels (Samuels, 1988). Because steel subcomponents experience significantly lower temperatures, creep is not expected to occur. The staff verified that Boral®, as applied in the DSCs as a neutron absorber, only provides criticality control of the spent fuel and does not serve any structural function. Therefore, the staff finds the applicant's assessment that no aging management activity is required for creep of the DSC to be acceptable.

Loss of Criticality Control Due to Boron Depletion

The applicant stated that only Boral® neutron absorber plates are used for criticality control in the DSC basket, and the performance is ensured only by the presence of boron-10 and the uniformity of its distribution. The applicant provided a bounding evaluation of boron-10 depletion of the neutron absorber plates, based on the maximum neutron source terms and minimum boron-10 areal density during 100 years of storage. The TLAA determines that the amount of boron-10 depleted in the poison plates due to neutron irradiation is negligible. Based on the results, the applicant stated that the ability of the poison plates to maintain subcriticality remains unaffected over the desired duration of storage. Therefore, the applicant concluded that loss of criticality due to boron depletion is an aging effect managed through a TLAA.

The staff verified by independent analysis that a negligible amount of boron-10 is depleted in the poison plates due to neutron irradiation and concluded that the applicant's analysis for boron-10 depletion is acceptable. SER Section 3.4.3 documents the staff's review of the TLAA on boron depletion.

Coating Degradation

The applicant stated that electroless nickel is applied to carbon steel subcomponents that are part of the DSC shell assembly as protective coatings to mitigate corrosion during loading and unloading operations when the DSC is in the spent fuel pool. However, the applicant stated that the coating does not perform any important-to-safety dry storage function, and it is not subject to corrosion in the internal helium environment during storage. Therefore, the applicant concluded that no aging management is required for the electroless nickel coating within the DSC.

The staff reviewed the design of the DSC with respect to the use of coating and determined that all components that use protective coatings are on the interior of the DSC and are not exposed to environmental conditions where coating degradation would need to be considered. Therefore, the staff finds the applicant's assessment that no aging management activity is required for coating degradation to be acceptable.

Effects of Temporary Attachments during Fabrication on SCC

The applicant stated that the DSC fabrication is in accordance with ASME Code Section III, NB-4000 (ASME, 1992). The applicant also stated that the welding and removal of temporary attachments result in a shallow HAZ, but through-wall detrimental microstructural alteration does not occur. Because the microstructural effects are limited and because the welding and removal of temporary attachments are done in accordance with the provisions of the ASME Code that includes required examinations, the applicant determined there is a low risk that SCC will occur at these locations, and any local cracking would be limited at a very shallow depth, even if crack initiation could occur. Therefore, the applicant concluded that it is not necessary to search for temporary attachment locations specifically for inspection during the period of extended operation.

The staff reviewed the applicant's assessment of temporary attachments and the relevant portions of the ASME Code, including NB-4435, related to the welding of nonstructural attachments and their removal (ASME, 1992). The staff determined that the process of welding and removing temporary fixtures to the DSC shell is unlikely to result in through-wall tensile residual stresses necessary for the initiation and propagation of SCC. Therefore, the staff concludes that the applicant's assessment of temporary attachments is acceptable.

3.3.1.3 Proposed Aging Management Activities

The applicant provided a TLAA for the following potential aging effects for the DSC:

- fatigue analysis of the NUHOMS® DSC shell assembly
- boron depletion, gamma irradiation, and neutron fluence analysis

The applicant evaluated the DSCs for fatigue according to the requirements of ASME Code NB-3222.4(d). SER Section 3.4.1 includes the staff's review of the applicant's TLAA on the fatigue of the DSC materials. The boron depletion analysis is based upon a bounding analysis of the highest neutron fluence. SER Section 3.4.3 includes the staff's review of the applicant's boron depletion TLAA. The applicant's evaluation of neutron fluence and gamma radiation on storage system structural materials relies on knowledge of the radiation effects on material and is supported by an analysis of the neutron fluence and gamma radiation. SER Section 3.4.4 includes the staff's review of the applicant's TLAA of neutron fluence and gamma radiation effects. The staff reviewed the applicant's assessment and determined that its analysis of these potential aging effects using a TLAA is acceptable.

The applicant has also developed the DSC External Services AMP to address potential aging effects and mechanisms. The applicant has concluded that the welded stainless steel DSC shell exposed to the sheltered environment inside the HSM may be susceptible to pitting corrosion, crevice corrosion, and galvanic corrosion and SCC, and these are aging effects that require management in the period of extended operation. The staff finds that the applicant's assessment is consistent with the results of NUREG-2214 showing that pitting corrosion, crevice corrosion, and galvanic corrosion and SCC are credible aging mechanisms over the period of extended operation. Therefore, the staff finds the applicant's assessment that aging management is required to manage the effects of DSC shell assembly to be acceptable. SER Section 3.5.1 includes the staff's review of this AMP.

3.3.2 Horizontal Storage Module

The applicant described the HSM in Section 2.4.2 of the LRA. The applicant stated that the Rancho Seco HSM design is similar to the Standardized NUHOMS® HSM, which is a low-profile, modular, reinforced concrete structure that provides radiation shielding protection, a means for passively removing spent fuel decay heat, structural support, and environmental protection to the loaded DSC.

3.3.2.1 Materials and Environments

The applicant described the materials and environments of the HSM in Sections 3.5.2 and 3.5.3 of the LRA. The applicant identified the materials of construction for the HSM subcomponents, as provided in SER Table 3.3-5. The applicant stated that HSMs are located outdoors; thus, the exterior surfaces of the HSM are exposed to all weather conditions, including insolation, wind, rain, snow, plant-specific ambient temperature, humidity, and airborne contamination. The applicant also stated that the Rancho Seco ISFSI is located in an environment that is considered to be dry and remote from chloride sources, industrial areas, or areas near the seashore. The applicant further stated that the temperature environment for the exterior surfaces of the HSM is bounded by the temperatures due to a design-basis heat load of 13.5 kilowatts at the normal ambient temperature range of −17.8 degrees C (0 degrees F) to 38.3 degrees C (101 degrees F) and the off-normal ambient temperature range of −28.9 degrees C (−20 degrees F) to 47.2 degrees C (117 degrees F).

The applicant stated that the HSM interior subcomponents (interior side of the HSM walls, HSM steel, and DSC external shell assembly subcomponents) are considered to be in a sheltered environment because they are located in a protected environment in uncontrolled air with no direct exposure to sun, wind, or precipitation. However, the applicant explained that a sheltered environment may contain moisture and salts or other contaminants from the external ambient air. The applicant also stated that the maximum predicted temperatures in the HSM concrete at the beginning of storage are 73.3 degrees C (164 degrees F) and 116.1 degrees C (241 degrees F) for normal and off-normal conditions, respectively. The applicant further stated that the HSM interior subcomponents are exposed to neutron fluence and gamma radiation. The applicant provided a TLAA in Appendix A to the LRA that analyzes the effects of neutron fluence and gamma exposure on the mechanical properties of reinforced concrete components. SER Section 3.4.4 includes the staff’s evaluation of the TLAA.

The staff reviewed the applicant’s description of the materials and environments for the HSM to confirm that the description is consistent with the descriptions and engineering drawings in the Rancho Seco ISFSI FSAR. Based on its confirmation of consistency with these design-basis documents, the staff finds the applicant’s identification of the materials and environments for the HSM to be acceptable.

3.3.2.2 Aging Effects and Mechanisms for the Horizontal Storage Module

3.3.2.2.1 Reinforced Concrete Structure

In Section 3.5.4.1 of the LRA, the applicant stated that loss of material, cracking, and change in material properties are the aging effects that could lead to a loss of intended functions of the HSM concrete subcomponents. The applicant identified various aging mechanisms that could lead to aging effects for these subcomponents and evaluated them to determine whether the aging mechanisms could lead to an aging effect requiring management.

The applicant described the aging effect of loss of material as manifesting in concrete subcomponents as scaling, spalling, pitting, and erosion, as described in American Concrete Institute (ACI) 201.1R, “Guide for Conducting a Visual Inspection of Concrete in Service” (ACI, 2008a). The applicant evaluated the following aging mechanisms:

- freeze-thaw
- salt scaling
- abrasion and cavitation
- thermal aging
- microbiological chemical attack
- aggressive chemical attack
- corrosion of embedded steel (embedments, rebar)
- delayed ettringite formation (DEF)

The applicant described the aging effect of cracking as manifesting in concrete subcomponents as a complete or incomplete separation of the concrete in two or more parts, as depicted in ACI 201.1R. The applicant evaluated the following aging mechanisms:

- freeze-thaw
- reactions with aggregates
- shrinkage
- thermal aging
- fatigue
- irradiation embrittlement
- creep
- corrosion of embedded steel (embedments, rebar)

The change in material properties aging effect manifests in concrete subcomponents as increased permeability, increased porosity, reduction in pH, reduction in tensile strength, reduction in compressive strength, reduction in modulus of elasticity, and reduction in bond strength. The applicant evaluated the following aging mechanisms:

- leaching of calcium hydroxide
- thermal aging
- aggressive chemical attack, including microbiological chemical attack
- irradiation embrittlement
- fatigue

The staff reviewed the applicant's identification of aging mechanisms and effects for the HSM concrete. In its review, the staff considered NRC guidance, the technical literature, and operating experience from nuclear and nonnuclear applications. A summary of the staff's evaluation of the aging mechanisms and effects follows.

<u>Loss of Material and Cracking Due to Freeze-Thaw</u>

The applicant stated that repeated freezing and thawing can cause degradation of concrete as a result of hydraulic pressure due to water freezing within the pores of the concrete. Freeze-thaw degradation manifests as scaling, cracking, and spalling. The applicant also stated that the weathering index for the Northern California region is less than 50 day-in./yr, in accordance with American Society for Testing and Materials International (ASTM) ASTM C216, "Standard Specification for Facing Brick (Solid Masonry Units Made from Clay or Shale)," issued in 2016 (ASTM, 2016). The applicant further stated that the Rancho Seco preapplication inspection results indicate that the HSM concrete surfaces do not show any visible degradation from freeze-thaw aging effects after 15 years of service. Therefore, the applicant concluded that loss of material and cracking due to freeze-thaw of HSM concrete are not aging effects requiring management.

The staff reviewed the applicant's evaluation of the potential for freeze-thaw degradation of the HSM concrete. The staff notes that concretes that are nearly or fully saturated with water can be damaged by repeated freezing and thawing cycles in environments with weathering indexes on the order of 100 day-in./yr or greater (NRC, 2010a). For environments with weathering indexes less than 100 day-in./yr, freeze and thaw degradation is not considered to be significant. Therefore, the staff finds the applicant's determination that aging management activity is not required for freeze-thaw of the HSM concrete to be acceptable because the weathering index for the Northern California region is less than 50 day-in./yr.

Loss of Material Due to Abrasion and Cavitation

The applicant stated that the HSM concrete is not subjected to flowing water. Therefore, the applicant concluded that loss of material due to abrasion and cavitation is not an aging effect requiring management.

The staff notes that loss of material due to abrasion and cavitation is caused by flowing water. Therefore, the staff finds the applicant's determination that aging management activity is not required for abrasion and cavitation of the HSM concrete to be acceptable because the HSM concrete is not subjected to flowing water.

Loss of Material and Change in Material Properties Due to Thermal Aging

The applicant stated that the maximum HSM concrete temperatures at the beginning of storage are 73.3 degrees C (164 degrees F) and 116.1 degrees C (241 degrees F) for normal and off-normal conditions, respectively. The applicant also stated that, under normal conditions of storage, the concrete temperatures will experience a gradual decrease over the service life of the HSM with only daily or seasonal fluctuations due to the ambient conditions. The applicant cited NUREG-1536, Revision 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility—Final Report," issued July 2010 (NRC, 2010b), to show that the HSM concrete temperatures are consistent with the temperature limits for storage of concrete structures. The applicant stated that the following temperature criteria are used for the HSM concrete:

- If concrete temperatures of general or local areas do not exceed 93.3 degrees C (200 degrees F) in normal or off-normal conditions or occurrences, no tests or reduction of concrete strength in the design analysis are required.

- If concrete temperatures of general or local areas exceed 93.3 degrees C (200 degrees F) but would not exceed 149 degrees C (300 degrees F), no tests or reduction of concrete strength are required if Type II cement is used and aggregates are selected that are acceptable for concrete in this temperature range. The following criteria for fine and coarse aggregates are considered suitable:

  - Satisfy ASTM C33 requirements and other requirements as referenced in ACI 349.3R, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," issued 2010 (ACI, 2010), for aggregates.

  - Demonstrate a coefficient of thermal expansion (tangent in temperature range of 21 degrees C to 37.8 degrees C [70 degrees F to 100 degrees F]) no greater than 1×10^{-5} cm/cm/degree C (6×10^{-6} in./in./degree F) or be one of the following minerals: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite.

The applicant concluded that loss of material and change in material properties due to thermal aging are not aging effects requiring management because the above temperature criteria are consistent with the criteria described in NUREG-1536, Revision 1.

The staff reviewed the applicant's evaluation of the potential for thermal aging of the HSM concrete. The staff notes that the applicant's temperature criteria are consistent with the guidance in NUREG-1536, Revision 1, for acceptable temperature limits during operation of DSS concrete structures. Therefore, the staff finds the applicant's determination that aging management activity is not required for thermal aging of the HSM concrete to be acceptable because the maximum HSM concrete temperatures are below the temperature limits.

Loss of Material and Change in Material Properties Due to Aggressive Chemical Attack

The applicant stated that, because the HSMs are installed above grade on a concrete basemat, the HSM concrete is not normally subject to aggressive chemical attack due to prolonged wetting. The applicant also stated that ISFSIs may be located in areas where the exposed surfaces of HSM can be subject to sulfur-based acid-rain degradation; however, no explicit acid rain data are available for the Rancho Seco ISFSI. Because the ISFSI is potentially located in an area where acid-rain could occur, the applicant concluded that loss of material and change in material properties due to aggressive chemical attack are aging effects requiring management.

The staff reviewed the applicant's evaluation of the potential for aggressive chemical attack of the HSM concrete. The staff notes that, when aggressive ions or acids intrude into the pore network of the concrete, the consequent chemical attack can cause several degradation phenomena. Depending on the type of aggressive chemical, the degradation of concrete can manifest in the form of cracking, loss of strength, concrete spalling and scaling, and reduction of pH. The staff also notes that, although there is a lack of explicit acid rain data for the Rancho Seco ISFSI, the data from the National Oceanic and Atmospheric Administration's National Atmospheric Deposition Program (NADP) indicate that the general area does not show evidence for acid rain. The NADP data are publicly available on the NADP Web site at https://nadp.slh.wisc.edu/NTN/annualmapsByYear.aspx#2017. Nevertheless, because the applicant stated that the ISFSI is potentially located in an area where acid-rain could occur, the staff finds the applicant's determination that aging management activity is required for aggressive chemical attack of the HSM concrete to be acceptable.

Loss of Material and Cracking Due to Corrosion of Reinforcing Steel

The applicant stated that the oxide products resulting from corrosion of the reinforcing steel embedded in concrete can result in tensile stresses and eventually cause hairline cracking, followed by rust staining, spalling, and more cracking in the concrete surrounding the embedded steel. The applicant referenced operating experience documented in Section 3.2 of the LRA, which contains cases where spalling of concrete cover with exposed rebar was observed. The applicant further stated that, although all observed cases were repaired, environmental degradation of the HSM concrete due to rebar corrosion is considered an aging effect requiring management for the HSM. Therefore, the applicant concluded that corrosion of reinforcing steel is a credible concrete aging mechanism.

The staff reviewed the applicant's evaluation of the potential for corrosion of the reinforcing steel in the HSM concrete. The staff notes that corrosion of the reinforcing steel embedded in the concrete is mainly caused by the presence of chloride ions in the concrete pore solution and carbonation of the concrete. The presence of corrosion products at the steel surface can generate internal stresses within the concrete matrix, causing cracks and spalling of the concrete cover with consequent structural damage. Based on the reported operating experience for corrosion of the reinforcing steel, the staff finds the applicant's determination that aging management activity is required for corrosion of the reinforcing steel in the HSM concrete to be acceptable.

Cracking Due to Reaction with Aggregates

The applicant stated that an alkali-silica reaction (ASR) can occur when aggregate containing silica is exposed to alkaline solutions, causing expansion and cracking of concrete structures. The applicant also stated that HSM concrete aggregate reactions during the period of extended operation are not likely, based on the HSM operating experience history without ASR-type cracking ever observed; however, reactions with aggregates, should they occur, have the potential to adversely affect the structural (strength) properties of the concrete. Therefore, the applicant concluded that cracking due to reaction with aggregates is an aging effect requiring management.

The staff reviewed the applicant's evaluation of the potential effects of reaction with aggregates for the HSM concrete. The staff notes that reactions can occur between the concrete aggregate and alkaline components within the cement or from outside sources such as deicing salts and groundwater. The reaction products (e.g., alkali-silica gel in the case of alkali-silica reactions) can swell with the absorption of water, exerting expansive pressures within the concrete and leading to cracking (ACI, 2008c). Such degradation has been identified in nuclear power plant concrete structures. The applicant also identified reaction with aggregates as an applicable aging mechanism. Therefore, the staff finds the applicant's determination that aging management activity is required for reaction with aggregates to be acceptable.

Cracking Due to Shrinkage

The applicant stated that, according to ACI 209R, "Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures (Reapproved 2008)" (ACI, 2008b), most of the concrete shrinkage has already occurred (91 percent in the first year, 98 percent in 5 years, and 100 percent in 20 years). Since 20 years will have passed by the end of the initial approved storage term, the applicant concluded that cracking due to shrinkage is not an aging effect requiring management during the period of extended operation.

The staff reviewed the applicant's evaluation of the potential for shrinkage of the HSM concrete. The staff notes that the applicant's assessment is based on ACI 209R and is consistent with the assessment of concrete shrinkage in NUREG-2214. Therefore, the staff finds the applicant's determination that aging management activity is not required for shrinkage to be acceptable because most of the shrinkage will take place early in the life of the concrete and is not expected to influence concrete performance after the initial licensing period.

Change in Material Properties Due to Leaching of Calcium Hydroxide

The applicant stated that leaching of calcium hydroxide due to water penetration can result in loss of concrete material, converting the cement into gels that have no strength. The applicant also stated that leaching over long periods of time can not only increase the permeability of concrete but also lower the pH of the concrete and affect the integrity of the protective oxide film of reinforcement steel. Since there is operating experience indicating occurrences of leaching in the HSM concrete, the applicant concluded that change in material properties due to leaching of calcium hydroxide is an aging effect requiring management.

The staff reviewed the applicant's evaluation of the potential effects of leaching of calcium hydroxide on the HSM concrete. The staff notes that a flux of water through a concrete surface can result in the removal, or leaching, of calcium hydroxide (Hanson et al., 2012). This can cause a loss of concrete strength, an increase in the concrete porosity and permeability, and a reduction in pH. The applicant also identified leaching of calcium hydroxide as a potential aging mechanism and noted this could lead to a change in the concrete material properties. Therefore, the staff finds the applicant's determination that aging management activity is required for leaching of calcium hydroxide in the HSM concrete to be acceptable.

Cracking Due to Irradiation Embrittlement

The applicant stated that the level of irradiation over the extended operation is not expected to reach a level that is sufficient to cause a reduction in concrete strength. Therefore, the applicant concluded that cracking due to irradiation effect is not an aging effect requiring management.

The applicant also stated that the HSM interior subcomponents are exposed to neutron fluence and gamma radiation. The applicant provided a TLAA in Appendix A to the LRA that analyzes the effects of neutron fluence and gamma exposure on the mechanical properties of reinforced concrete components. SER Section 3.4.4 includes the staff's evaluation of the TLAA.

The staff reviewed the applicant's evaluation of the potential for irradiation embrittlement. The staff notes that the maximum potential accumulated neutron fluence on DSS basket components after 100 years is three orders of magnitude below the level that would lead to a reduction of concrete strength and elastic modulus (NRC, 2019). The gamma dose is also expected to be several orders of magnitude less than the limits defined in the above references, according to the specific DSS design bases. Therefore, the staff finds the applicant's determination that aging management activity is not required for irradiation embrittlement to be acceptable because the level of irradiation over the period of extended operation is expected to be below the critical radiation levels.

Cracking Due to Creep

The applicant explained that creep-induced concrete cracks are typically not large enough to result in concrete deterioration or in exposure of the reinforcing steel to environmental stressors, and that cracks of this magnitude do not reduce the concrete's compressive strength. The applicant stated that creep is significant when new concrete is subjected to load; however, creep decreases exponentially with time. The applicant also stated that, according to ACI 209R, 78 percent of creep occurs within the first few years, 93 percent within 10 years, 95 percent within 20 years, and 96 percent within 30 years. Because 20 years will have passed by the end of the initial approved storage term, the applicant concluded that cracking due to creep is not an aging effect requiring management during the period of extended operation.

The staff reviewed the applicant's evaluation of the potential for creep of the HSM concrete. The staff notes that the age of concrete and the magnitude and duration of sustained loading are the primary factors that determine the magnitude of the creep of concrete (Neville and Dilger, 1970). Therefore, the staff finds the applicant's determination that aging management activity is not required for creep to be acceptable because the initial sustained load is normally low, and no significant change of load is expected after the initial licensing period.

Cracking and Change in Material Properties Due to Fatigue

The applicant stated that the only source of thermal fatigue is daily and seasonal environmental temperature fluctuations, and the maximum average daily fluctuation at the Rancho Seco ISFSI is 8.3 degrees C (47 degrees F). The applicant also stated that the high thermal mass of the HSM and low conductivity of the concrete material limit the magnitude of the thermal forces that could be developed due to this temperature difference. Therefore, the applicant concluded that cracking and change in material properties due to fatigue are not aging effects requiring management.

The staff reviewed the applicant's evaluation of the potential for fatigue of the HSM concrete. The staff notes that concrete fatigue in the DSS reinforced concrete may be caused by diurnal and seasonal temperature gradients through the wall of the DSS assembly. The staff evaluated the effects of cyclic stresses due to seasonal and daily temperature variations (NRC, 2019). The results showed that the calculated ratio of the concrete compressive stress to its design strength is lower than the lowest stress/cycles-to-failure (S–N) curve for concrete reported in ACI 215R, "Considerations for Design of Concrete Structures Subjected to Fatigue Loading" (ACI, 1997). Therefore, the staff finds the applicant's determination that aging management activity is not required for fatigue of the HSM concrete to be acceptable because the thermal stresses due to seasonal and daily temperature variations are not sufficient for fatigue-induced failure in the concrete.

Loss of Material Due to Delayed Ettringite Formation

The applicant stated that DEF is the potential deleterious reformation of ettringite in moist concrete after destruction of primary ettringite by high early-age temperatures from heat treatment, and heat treatment temperatures above approximately 70 degrees C (158 degrees F) are most often cited to cause deleterious volume expansion due to DEF. The applicant stated that the HSM fabrication specification requires the concrete to be cured and protected in accordance with the curing provisions of ACI 318, "Building Code Requirements for Structural Concrete and Commentary" (ACI, 2005). Therefore, the applicant concluded that loss of material due to DEF of the HSM concrete is not an aging effect requiring management.

The staff reviewed the applicant's evaluation of the potential for DEF of the HSM concrete. The staff notes that the conditions necessary for the occurrence of DEF are excessive temperatures during concrete placement and curing, the presence of internal sulfates, and a moist environment. NUREG-1536, Revision 1, cites ACI 318 as an applicable code for the design and construction of concrete structures of the DSSs, which effectively limits the concrete temperature to below 70 degrees C (158 degrees F), preventing the development of DEF. Therefore, the staff finds the applicant's determination that aging management activity is not required for DEF of the HSM concrete to be acceptable because the HSMs are fabricated in accordance with ACI 318.

Loss of Material and Change in Material Properties Due to Microbiological Degradation

The applicant stated that microbiological degradation of concrete structures is a potential aging mechanism caused by live organisms that grow in environments that offer favorable conditions (e.g., available moisture, neutral pH, presence of nutrients), which facilitate the colonization of microbes on concrete surfaces. The applicant also stated that microorganisms can affect the concrete mainly by contributing to the deterioration of the exposed concrete surface, reducing the protective cover depth, increasing concrete porosity, and increasing the transport of degrading substances into the concrete that can accelerate cracking and spalling. In response to an RAI, the applicant stated that, in an external or sheltered environment, favorable conditions for microbiological degradation mechanisms may exist because of the potential presence of moisture (SMUD, 2019a). However, the applicant also stated that the conditions would be intermittent, and there is no evidence that actual concrete subcomponents in these environments microbiologically degrade. The applicant concluded that, consistent with the guidance in NUREG-2214, loss of material and change in material properties due to microbiological degradation in an external or sheltered environment are not aging effects requiring management.

The staff reviewed the applicant's evaluation of the potential for microbiological degradation of concrete structures. The staff notes that, while microbiological degradation of concretes exposed to groundwater or soil (below-grade) environments is considered credible in NUREG-2214, the degradation mode is not considered credible in an outdoor air environment. The staff finds the applicant's determination that aging management activity is not required for microbiological degradation of concrete structures to be acceptable because the HSM concrete is not exposed to a groundwater or soil environment.

<u>Aging Effects Due to Salt Scaling</u>

In response to an RAI, the applicant stated that salt scaling is defined as superficial damage caused by freezing a saline solution on the surface of a concrete body (SMUD, 2019a). The damage is progressive and consists of the removal of small chips or flakes of material. The applicant also stated that, similar to freeze-thaw damage, salt scaling takes place when concrete is exposed to freezing temperatures, moisture, and dissolved salts. The degradation is maximized at a moderate concentration of salt (e.g., from deicing salts). The applicant further stated that the weathering index for the Northern California region is less than 50 day-in./yr, and freezing degradation in this environment is not considered to be significant. Therefore, the applicant concluded that salt scaling of HSM concrete is not an applicable aging mechanism.

The staff reviewed the applicant's evaluation of the potential for salt-scaling degradation of the HSM concrete. The staff notes that, because salt scaling is closely related to freezing and thawing damage, the HSM concrete is not susceptible to salt-scaling degradation in environments with weathering indexes less than 100 day-in./yr (NRC, 2010a). Therefore, the staff finds the applicant's determination that aging management activity is not required for salt scaling of the HSM concrete to be acceptable because the weathering index for the Northern California region is less than 50 day-in./yr.

3.3.2.2.2 Metallic Subcomponents

In Section 3.5.4.3 of the LRA, the applicant stated that loss of material, cracking, and change in material properties are the aging effects that could lead to a loss of intended functions of the carbon steel and stainless steel HSM subcomponents. The applicant identified various aging mechanisms that could lead to aging effects for these subcomponents and evaluated them to determine whether they could lead to an aging effect requiring management.

For loss of material, the applicant evaluated the following aging mechanism and material combinations:

- loss of material due to general corrosion—carbon steel
- loss of material due to crevice and pitting corrosion—carbon steel and stainless steel

- loss of material due to galvanic corrosion—dissimilar metals

For cracking, the applicant evaluated the following aging mechanism and material combinations:

- cracking due to SCC—stainless steel
- cracking due to SCC—bolting
- cracking due to thermal fatigue—carbon steel and stainless steel

For change in material properties, the applicant evaluated the following aging mechanism and material combinations:

- change in material properties due to thermal aging—carbon steel and stainless steel

- change in material properties due to irradiation embrittlement—carbon steel and stainless steel

In addition to the degradation effects and mechanisms described above, the applicant also considered the potential for coating degradation of the DSC support structure inside the HSM.

Loss of Material Due to General Corrosion

The applicant stated that the atmosphere inside the HSM will depend on the site location and environment but is typically benign in terms of corrosion. The applicant further stated that, although DSC decay heat will heat the air, preventing the accumulation or condensation of moisture inside the HSM, the decay heat will decline during the period of extended operation. Thus, the presence of moist air cannot be ruled out. Therefore, the applicant concluded that loss of material due to general corrosion is an aging effect requiring management for carbon and low-alloy steel in outdoor and sheltered environments. The applicant stated that stainless steel is not susceptible to general corrosion.

The staff reviewed the applicant's assessment of general corrosion of carbon steel and stainless steel, as well as the description of the outdoor and sheltered environments. In outdoor conditions, rain, fog, and dew condensation can generate moisture layers on the steel surface that cause general corrosion. Atmospheric corrosion rates can vary from 0 to 0.2 millimeter per year, depending on relative humidity, temperature, and levels of chloride and pollutants in the atmosphere (NACE, 2002). Rates can be more significant in industrial and marine environments (McCuen and Albrecht, 1994). In a sheltered environment, deliquescence of airborne salts below the dew point could also generate an aqueous electrolyte, which is conducive to general corrosion. The staff also notes that stainless steels form an oxidized protective film on their surfaces that leads to negligible general corrosion (Grubb et al., 2005). Therefore, the staff finds the applicant's evaluation of loss of material due to general corrosion to be acceptable because the applicant has identified and will manage this well-known mechanism and effect for carbon and low-alloy steels and has appropriately excluded the consideration of general corrosion for stainless steels, given the passivity of these alloys in the outdoor and sheltered environments.

Loss of Material Due to Crevice Corrosion and Pitting Corrosion

The applicant stated that crevice corrosion is a form of localized corrosion that occurs in shielded spaces or crevices created by component or part connections such as lap joints, splice plates, bolt threads, under bolt heads, or points of contact between metals and nonmetals. A number of factors influence crevice corrosion, including electrolyte composition and flow, the geometry of the occluded region, and the concentration of dissolved oxygen within the occluded region. The applicant stated that, while atmospheric pollutants and contaminants are typically insufficient to promote crevice corrosion, alternating wetting and drying is particularly harmful because this leads to a concentration of atmospheric pollutants and contaminants.

The applicant described pitting corrosion as another localized corrosive attack in aqueous environments containing dissolved oxygen and halides such as chlorides and bromides. As with crevice corrosion, the applicant noted that areas in which aggressive species can concentrate (i.e., locations of frequent or prolonged wetting or of alternate wetting and drying) are particularly susceptible to pitting.

The applicant stated that DSC decay heat will heat the air, preventing the accumulation or condensation of moisture inside the HSM. However, because the DSC decay heat will decrease during the period of extended operation, the applicant recognized that the presence of moist air cannot be ruled out, and HSM metals are all susceptible to crevice corrosion and pitting corrosion. Therefore, the applicant concluded that crevice corrosion and pitting corrosion are aging effects requiring management for HSM metals.

The staff reviewed the applicant's evaluation of the potential effects of pitting and crevice corrosion on HSM materials. The staff notes that, in addition to moisture, the outside atmosphere can transport contaminants to the HSM subcomponent surfaces. Thus, the conditions necessary for pitting and crevice corrosion may be present for carbon and low-alloy steels (Revie, 2000) and stainless steels (Grubb et al., 2005). Therefore, the staff finds the applicant's evaluation of loss of material due to pitting corrosion and crevice corrosion to be acceptable because the applicant's management of these aging mechanisms and their effects for all metallic components in the outdoor and sheltered environments is consistent with their observed occurrence in potentially contaminated and moist environments.

<u>Loss of Material Due to Galvanic Corrosion</u>

The applicant identified the potential for galvanic corrosion of DSC support structure rail plate in contact with graphite lubricant because the graphite lubricant is noble relative to the rail face. Therefore, the applicant concluded that loss of material due to galvanic corrosion of the DSC support structure rail plate is an aging effect requiring management.

The staff reviewed the materials of construction for the DSC support rails with a graphite lubricant. Because graphite is strongly cathodic and the contact is close, the galvanic coupling effect between stainless steel and graphite is expected to be strong. Therefore, the staff finds the applicant's assessment that aging management activity is required for galvanic corrosion of the DSC support structure rail plate to be acceptable.

<u>Cracking Due to Stress-Corrosion Cracking</u>

The applicant stated that SCC is a localized, nonductile cracking failure resulting from the combination of applied or residual tensile stresses, material condition, and the presence of a corrosive environment. The applicant referenced the work of EPRI (2007) to show that dissolved oxygen, sulfates, fluorides, and chlorides can provide the necessary environment for SCC to occur. The applicant acknowledged that austenitic stainless steels are susceptible to SCC, and residual stresses at the welds are likely to be sufficient to initiate SCC. Therefore, the applicant concluded that cracking due to SCC at the welds and HAZs of the stainless steel rail face to carbon steel support structures is an aging effect requiring management.

The applicant also recognized that bolting fabricated from high-strength, low-alloy steel is susceptible to SCC, and bolted connections, including bolted joints and threaded connections, exist in the HSM. However, the applicant stated that HSM structural component anchorages are installed snug-tight according to installation specifications. As a result, the tensile stress is not sufficiently high to initiate SCC. Therefore, the applicant concluded that cracking due to SCC of high-strength bolting is not an aging effect requiring management.

The staff reviewed the materials of construction, the stress level, and the environments of HSM subcomponents. All austenitic grades have long been reported in the literature to be susceptible to SCC, especially in the presence of chloride in the normal wrought condition (Grubb et al., 2005; Morgan, 1980; Kain, 1990). This susceptibility increases when the material is sensitized (He et al., 2014). In the welded condition, the HAZ, which is a thin band located adjacent to the weld, can be sensitized by the precipitation of carbides that extract chromium out of the metal matrix. Welds exist in the HSM subcomponents, such as at the stainless steel rail face to carbon steel support structures. Because sufficient weld residual stresses and susceptible material conditions are present near the welds, and aqueous electrolytes conducive to SCC are present in outdoor and sheltered environments, the staff finds that the potential for SCC of the welds is present and the applicant's assessment that aging management activity is required for SCC of HSM subcomponents to be acceptable.

The staff also reviewed the materials to construct the bolts and the stress level for the HSM subcomponents and acknowledged that SCC also requires the presence of a sufficient tensile stress. Calculations using the approach proposed by Baggerly (1999) show that the stress threshold to initiate SCC of steel bolts is usually larger than 70 percent of the bolting material's minimum yield strength, while EPRI (2007) states that stresses near the yield strength are required to initiate SCC. The high-strength structural bolts in the HSM are installed "snug-tight" and are not loaded close to critical stresses. Because of the low applied stresses, the staff finds the applicant's assessment that aging management activity is not required for SCC of steel bolts exposed to sheltered and outdoor environments to be acceptable.

<u>Cracking Due to Thermal Fatigue</u>

The applicant stated that the only source of thermal fatigue is environmental temperature fluctuation. For HSM steel subcomponents located inside the HSM (i.e., in a sheltered environment), the applicant stated that the thermal fluctuations due to external ambient temperature fluctuations are significantly dampened by the HSM walls and roof and the DSC decay heat. Therefore, the applicant concluded that thermal cycling fatigue due to fluctuations in the ambient conditions is not an aging effect requiring management for HSM subcomponents. The staff reviewed the applicant's analysis and concludes that cracking due to thermal fatigue is not an aging effect requiring management for the HSM subcomponents.

<u>Change in Material Properties Due to Thermal Aging</u>

The applicant stated that the maximum temperature of the HSM steel subcomponents at the beginning of storage is 116 degrees C (241 degrees F) during off-normal conditions of storage, and this temperature is well within allowed temperature limits for the structural metal components of the HSMs (371 degrees C [700 degrees F] for carbon steel and 427 degrees C [800 degrees F] for stainless steel), in accordance with the ASME Code. Furthermore, the applicant stated that this temperature is well below the embrittlement saturation temperature of 400 degrees C (752 degrees F) in NUREG/CR-6428 and is also below the lower temperature (335 degrees C [635 degrees F]) (Chandra et al., 2012a, b) at which embrittlement was observed for stainless steel welds. The applicant also cited NUREG-1801, which states that austenitic stainless steels with service temperature below 250 degrees C (482 degrees F) are not susceptible to thermal aging embrittlement. Therefore, the applicant concluded that changes in material properties due to thermal aging is not an aging effect requiring management for HSM metal subcomponents.

The staff recognizes that the ferrite present in austenitic stainless steel welds can transform by spinodal decomposition to form Fe-rich alpha and chromium-rich alpha prime phases, and further aging can produce an intermetallic G-phase (NRC, 2019). The spinodal decomposition and the formation of the intermetallic G-phase takes place during extended exposure to temperatures between 300 and 400 degrees C (572 and 752 degrees F) (Alexander and Nanstad, 1995; Chandra et al., 2012a). The possible significant thermal aging in nonwelded austenitic stainless steels would be a decrease in strength due to a decrease in dislocation density, recrystallization, and an increase in grain size. These processes occur during annealing at temperatures above 1,000 degrees C (1,832 degrees F). The staff also recognizes that undesired material property changes due to tempering of hardened carbon steel and low-alloy steels could occur at temperatures greater than 200 degrees C (392 degrees F) (Krauss, 2005). The staff notes that the temperatures of metal subcomponents exposed to outdoor and sheltered environments are bounded by the DSC shell temperature because these subcomponents are located farther away from the fuel. Time-temperature profiles calculated for the DSC shell estimate that the peak temperature is below 200 degrees C (392 degrees F) (EPRI, 2006; Meyer et al., 2013). Because the peak temperatures are below the temperature required to cause changes in material properties, the staff finds the applicant’s assessment that aging management activity is not required for thermal aging of steel subcomponents exposed to sheltered and outdoor environments to be acceptable.

<u>Change in Material Properties Due to Irradiation Embrittlement</u>

The applicant stated that high neutron radiation can cause loss of fracture toughness in steel. According to the calculations in the renewal application, the applicant showed that the neutron radiation experienced by the HSM steel subcomponents is orders of magnitude lower than that required to produce any effect. Therefore, the applicant concluded that neutron radiation is not a significant aging effect for the HSM steel subcomponents. The applicant further stated that gamma radiation does not have any significant impact on the properties of steel.

As described in NUREG-2214 and discussed in SER Section 3.3.1, the staff recognizes that neutron irradiation has the potential to alter the mechanical properties of materials, including a resultant increase in the tensile and yield strength and decrease in the toughness of carbon and alloy steels as well as an increase in the cracking susceptibility of stainless steels in some aqueous environments. SER Section 3.4.4 summarizes the staff’s independent verification of the applicant's calculation of maximum neutron fluence and gamma radiation levels. The staff determined the applicant’s calculation and results were conservative. Therefore, the staff finds the applicant’s assessment that aging management activity is not required for radiation embrittlement of steel subcomponents exposed to sheltered and outdoor environments to be acceptable.

Coating Degradation

The applicant stated that carbon steel subcomponents in the HSM are coated with inorganic coatings or are galvanized for corrosion protection, and the sliding surfaces of the DSC support rails of the HSM are coated with a dry film graphite lubricant to minimize friction during insertion and retrieval of the DSC. However, the applicant stated that no credit is taken for coatings or lubricants for the performance of intended functions. Therefore, the applicant concluded that the deterioration of coatings and dry lubricant is not an aging effect requiring aging management.

The staff reviewed the design of the HSM with respect to the use of coating on carbon steel and lubricant on DSC support rails and verified that they are not credited as supporting an important-to-safety function. Therefore, the staff finds the applicant's assessment that no aging management activity is required for coating degradation to be acceptable.

3.3.2.3 Proposed Aging Management Activities

The applicant credited the HSM AMP to manage aging mechanisms and effects for the HSM concrete and steel subcomponents. Based on its review of the information in the LRA, the staff concluded that an AMP is an acceptable means of ensuring that the identified aging effects will not result in a loss of intended functions. SER Section 3.5.2 includes the staff's evaluation of the HSM AMP.

3.3.3 Concrete Basemat

The applicant described the basemat in Section 2.4.5 of the LRA. The applicant stated that the basemat was built in accordance with applicable commercial-grade codes and standards. The applicant also stated that the HSMs are installed on a reinforced concrete basemat; however, there are no structural connections to transfer shear between the HSM base unit module and the concrete basemat.

3.3.3.1 Materials and Environments

The applicant described the materials and environments of the basemat in Sections 3.6.2 and 3.6.3 of the LRA. The applicant identified the construction material of the basemat as reinforced concrete. The applicant stated that the exterior surfaces of the basemat are exposed to all weather conditions, including insolation, wind, rain, snow, and plant-specific ambient temperature, humidity, and airborne contamination. The areas of the basemat within the footprint of the installed HSM array are in a sheltered environment. These areas are protected from outdoor effects (e.g., direct sunlight, precipitation) and experience temperatures and radiation exposure levels that are bounded by those of the HSM concrete subcomponents. In response to an RAI, the applicant further explained that below-grade portions of the basemat are in an underground environment exposed to soil (SMUD, 2019a). The reinforcing bar is embedded in the concrete and thus is exposed to an embedded environment.

The staff reviewed the applicant's description of the materials and environments for the basemat to confirm that the description is consistent with the descriptions and engineering drawings in the Rancho Seco ISFSI FSAR. Based on its confirmation of consistency with these design-basis documents, the staff finds the applicant's identification of the materials and environments for the basemat to be acceptable.

3.3.3.2 Aging Effects and Mechanisms for the Concrete Basemat

In Section 3.6.5 of the LRA, the applicant identified the aging effects of loss of material, cracking, and change in material properties for the basemat that require aging management, based on its evaluations in Sections 3.5.4.1 and 3.5.4.2 of the LRA. The applicant stated that, since the basemat is not specified to be fabricated in accordance with ACI 318, DEF is an applicable aging mechanism for the basemat in the sheltered, external, and underground environments. The applicant also identified settlement as an aging mechanism associated with the aging effect of cracking in the sheltered, external, and underground environments.

For the aging effect of loss of material, the applicant identified the following aging mechanisms:

- aggressive chemical attack
- corrosion of embedded steel (embedments, rebar)
- DEF

For the aging effect of cracking, the applicant identified the following aging mechanisms:

- reactions with aggregates
- corrosion of embedded steel (embedments, rebar)
- settlement

For the aging effect of change in material properties, the applicant identified the following aging mechanisms:

- leaching of calcium hydroxide
- aggressive chemical attack

The applicant further identified the following aging effect and mechanisms for the carbon steel rebar that require aging management, based on its evaluations in Sections 3.5.4.3 and 3.5.4.4 of the LRA:

- loss of material due to general corrosion
- loss of material due to pitting corrosion
- loss of material due to crevice corrosion

The staff reviewed the applicant's identification of aging mechanisms and effects for the basemat. In its review, the staff considered NRC guidance, the technical literature, and operating experience from nuclear and nonnuclear applications. With the exception of DEF and differential settlement, the staff's evaluation of the aging mechanisms and effects for the HSM included in SER Section 3.3.2.2 is also applicable to the concrete basemat. A summary of the staff's evaluation of DEF and differential settlement follows.

<u>Loss of Material Due to Delayed Ettringite Formation</u>

The staff's evaluation of the DEF aging mechanism is discussed in SER Section 3.3.2.2, which concludes that DEF is not a credible aging effect for concrete that is fabricated in accordance with ACI 318. Since the concrete basemat was not fabricated in accordance with ACI 318, the staff concludes that DEF is a credible aging mechanism for the basemat.

Cracking Due to Differential Settlement

The applicant stated that settlement of a structure may be due to changes in the site conditions (e.g., water table, soil). The applicant cited NUREG-2214, which states that differential settlement of concrete structures involves a combination of immediate settlement and progressive long-term settlement. Because of the potential aging effects that long-term settlement of the basemat may have on the HSM intended functions, the applicant concluded that cracking due to differential settlement of the basemat is an aging effect requiring management.

The staff reviewed the applicant's evaluation of the potential effects of differential settlement for the concrete basemat. The staff notes that differential settlement is the uneven deformation of the supporting foundation soil (Das, 1999; NAVFAC, 1986). The occurrence of differential settlement depends on the type of soil, thickness of soil layers, water-table level, depth of the foundation mat below the ground surface, liquefaction during seismic events, and mechanical loading. Operating experience has shown that differential settlement has occurred in nuclear power plant concrete structures (NRC, 1995). Because of this, the staff finds the applicant's determination that aging management activity is required for differential settlement of the basemat to be acceptable.

3.3.3.3 Proposed Aging Management Activities

The applicant credited the Basemat AMP to manage aging mechanisms and effects for the concrete basemat. Based on its review of the information in the LRA, the staff concluded that an AMP is an acceptable means of ensuring that the identified aging effects will not result in a loss of intended functions. SER Section 3.5.4 describes the staff's evaluation of the Basemat AMP.

3.3.4 Transfer Cask

The applicant described the TC in Sections 2.4.3 and 3.7.1 of the LRA. The applicant stated that the Rancho Seco ISFSI has one onsite TC that is designated as the NUHOMS® MP187 TC. The MP187 TC is a cylindrical vessel with a welded bottom assembly and a bolted top cover plate.

3.3.4.1 Materials and Environments

The applicant listed the construction materials for the TC in Section 3.7.2, Table 3-9, and the environments in Section 3.7.3 in the LRA. The cask is fabricated primarily of stainless steel. Nonstainless steel materials include the lead gamma shielding material between the containment boundary inner shell and the structural shell, the cementitious BISCO NS-3 neutron shielding material, the aluminum inner neutron shield plugs, the O-ring seals, and the carbon steel closure bolts. The applicant identified two normal operating environments: (1) sheltered until the DSCs are to be retrieved from the HSMs for inspection or offsite shipment and (2) embedded or encased for the shielding materials and the inner side of the metals encasing the shielding materials.

The staff reviewed the design bases of the TC and design drawings and confirmed that the application appropriately identified the materials of construction and environments. The staff concludes that the applicant adequately identified the materials of construction and environments for the TC subcomponents.

3.3.4.2 Aging Effects and Mechanisms for the Transfer Cask

The applicant evaluated the materials of construction for TC subcomponents, including stainless steel and carbon steel, that are subject to further AMR. The applicant considered aging effects of these materials that could, if left unmanaged, cause degradation of TC subcomponents and result in loss of the component's intended function(s). The applicant's assessment of the aging effects that could cause loss of intended function(s) if left unmanaged include the following:

For the loss of material aging effect, the applicant evaluated the following aging mechanism and material combinations:

- Loss of material due to general corrosion – carbon steel
- Loss of material due to crevice and pitting corrosion – carbon steel and stainless steel
- Loss of material due to galvanic corrosion – dissimilar metals
- Loss of material due to wear – rails, inner shell, and trunnions

For the cracking aging effect, the applicant evaluated the following aging mechanism and material combinations:

- Cracking due to stress corrosion cracking – stainless steel
- Cracking due to chloride induced stress corrosion cracking – stainless steel
- Cracking due to thermal fatigue – carbon steel and stainless steel

For the change in material properties aging effect, the applicant evaluated the following aging mechanism and material combinations:

- Change in material properties due to thermal aging – carbon steel and stainless steel
- Change in material properties due to irradiation embrittlement – carbon steel and stainless steel

The applicant included a supplemental evaluation for the lead, BISCO NS-3, and aluminum shielding materials in the encased or embedded environment in the TC. The applicant also evaluated the degradation of the dry film lubricant used to minimize friction during DSC transfer operations.

<u>Loss of Material Due to General Corrosion</u>

The applicant stated that carbon steel surfaces in contact with moist air or water are subject to general corrosion. Therefore, the applicant concluded that loss of material due to general corrosion is an aging effect requiring management for carbon and low-alloy steel in sheltered environments. However, the applicant stated that stainless steel is not susceptible to general corrosion. Furthermore, the applicant stated that, in an embedded or encased environment, general corrosion of carbon or stainless steel is not a mechanism that requires management, due to the lack of moisture.

The staff reviewed the applicant's assessment of general corrosion of carbon steel and stainless steel, as well as the description of the sheltered environment. The staff notes that general corrosion of carbon and low-alloy steels in moisture-bearing atmospheres is a well-known aging mechanism. The rate of material loss depends on a number of factors, including humidity, time of wetness, atmospheric contaminants, and oxidizing species (Fontana, 1986). The staff also notes that stainless steels form an oxidized protective film on their surfaces that leads to negligible general corrosion (Grubb et al., 2005). Therefore, the staff finds the applicant's evaluation of loss of material due to general corrosion to be acceptable because the applicant has identified and will manage this well-known mechanism and effect for carbon and low-alloy steels and has appropriately excluded the consideration of general corrosion for stainless steels, given the passivity of these alloys in the sheltered environment.

The staff also reviewed the applicant's assessment of general corrosion of carbon steel and stainless steel used in an embedded or encased environment. The staff notes that general corrosion of steels embedded in neutron-shielding materials is not likely to occur because the embedded side of the steels has limited exposure to water and oxygen. Therefore, the staff finds the applicant's assessment that no aging management activity is required for general corrosion of steels in an embedded or encased environment to be acceptable.

Loss of Material Due to Crevice Corrosion and Pitting Corrosion

The applicant stated that crevice corrosion is a form of localized corrosion that occurs in shielded spaces or crevices created by component or part connections such as lap joints, splice plates, bolt threads, under bolt heads, or points of contact between metals and nonmetals. The applicant stated that, while atmospheric pollutants and contaminants are typically insufficient to promote crevice corrosion, alternating wetting and drying is particularly harmful because this leads to concentrated atmospheric pollutants and contaminants. The applicant described pitting corrosion as another localized corrosive attack in aqueous environments containing dissolved oxygen and halides such as chlorides and bromides. As with crevice corrosion, the applicant noted that areas in which aggressive species can concentrate (i.e., locations of frequent or prolonged wetting or of alternate wetting and drying) are particularly susceptible to pitting. The applicant noted that pitting corrosion is more common with passive materials, such as 300 Series austenitic stainless steels, than with nonpassive materials, such as carbon steels. The applicant recognized that the presence of moist air cannot be ruled out, and some steels are susceptible to crevice corrosion and pitting corrosion. Therefore, the applicant concluded that crevice corrosion and pitting corrosion are aging effects requiring management for carbon steel, low-alloy steel, and stainless steel in a sheltered environment. Furthermore, the applicant stated that, in an embedded or encased environment, crevice corrosion and pitting corrosion of carbon or stainless steel are not mechanisms that require management, due to the lack of moisture.

The staff reviewed the applicant's evaluation of the potential effects of pitting and crevice corrosion for carbon steel, low-alloy steel, and stainless steel. The staff notes that, in addition to moisture, the outside atmosphere can transport contaminants to a cask's external surfaces. Thus, the conditions necessary for pitting and crevice corrosion may be present for carbon and low-alloy steels (Revie, 2000) and stainless steels (Grubb et al., 2005). Therefore, the staff finds the applicant's evaluation of loss of material due to pitting and crevice corrosion to be acceptable because the applicant's management of these aging mechanisms and effects for all metallic components in the sheltered environment is consistent with their observed occurrence in potentially contaminated and moist environments.

The staff also reviewed the applicant's assessment of crevice corrosion and pitting corrosion of carbon steel and stainless steel used in an embedded or encased environment. The staff notes that crevice corrosion and pitting corrosion of steels embedded in neutron-shielding materials are not likely to occur because the embedded side of the steels has limited exposure to water and oxygen. Therefore, the staff finds the applicant's assessment that no aging management activity is required for crevice corrosion and pitting corrosion of steels in an embedded or encased environment to be acceptable.

Loss of Material Due to Galvanic Corrosion

The applicant stated that there is potential for galvanic corrosion of the TC rails, inner shell, and bottom end closure that are in contact with graphite lubricant because the graphite lubricant is noble relative to stainless steel used to construct these TC subcomponents. The applicant's assessment for the TC rails is the same as that identified for the DSC shell and the DSC support structure rail plate in the HSM. Therefore, the applicant concluded that loss of material due to galvanic corrosion of the rail face, inner shell, and bottom end closure is an aging effect requiring management in a sheltered environment. The applicant stated that, in an embedded or encased environment, galvanic corrosion of carbon or stainless steel is not a mechanism that requires management, due to the lack of moisture.

The staff reviewed the materials of construction for some of the TC subcomponents with a graphite lubricant. Because graphite is strongly cathodic and the contact is close, the galvanic coupling effect between stainless steel and graphite is expected to be strong. Therefore, the staff finds the applicant's assessment that aging management activity is required for galvanic corrosion of the rail face, inner shell, and bottom end closure to be acceptable.

The staff also reviewed the applicant's assessment of galvanic corrosion of carbon steel and stainless steel used in an embedded or encased environment. The staff notes that galvanic corrosion of steels embedded in neutron-shielding materials is not likely to occur because the embedded side of the steels has limited exposure to water and oxygen. Therefore, the staff finds the applicant's assessment that no aging management activity is required for galvanic corrosion of steels in an embedded or encased environment to be acceptable.

Loss of Material Due to Wear

The applicant identified the TC subcomponents with materials in sliding contact and that thus may be subject to loss of material due to wear as the Nitronic® 60 austenitic stainless steel rails, the TC inner shell, and the trunnions. The applicant acknowledges that operating experience showed scratching of the cask liner and trunnions, although the DSC is designed to slide from the TC into the HSM and back without damage to the sliding surfaces by adding the rails. Therefore, the applicant concluded that loss of material due to wear is an aging effect requiring management for the Nitronic® 60 rails, the TC inner shell, and the trunnions.

The staff reviewed the applicant's evaluation of the potential effects of wear and verified the applicant's assessment of locations where there may be sliding contact. The staff recognizes that a TC may be used to transfer the DSCs from the HSMs for inspection or offsite shipment during period of extended operation, although it is currently in storage. Therefore, the staff finds the applicant's assessment that aging management activity is required for wear of the rail face, inner shell, and bottom end closure to be acceptable.

Cracking Due to Stress-Corrosion Cracking

The applicant stated that SCC is a localized, nonductile cracking failure resulting from the combination of tensile stress, material condition, and the presence of a corrosive environment. The applicant referenced the work of EPRI (2007) to show that dissolved oxygen, sulfates, fluorides, and chlorides can provide the necessary environment for SCC to occur. However, the applicant stated that the sheltered environment where the TC is currently stored is not conducive to SCC because this environment is different from an external ambient environment, which subjects the TC to rain, wind, snow, or dissolved oxygen. The applicant stated further that, due to the lack of moisture in an embedded or encased environment, SCC is not a mechanism that requires management. Therefore, cracking due to SCC is not an aging effect requiring management.

The staff reviewed the applicant's assessment of SCC as well as the description of the sheltered and embedded or encased environments. The staff reviewed the system design and verified that there is either lack of electrolytes in the sheltered environment or there is lack of moisture in the embedded or encased environment that is inductive to SCC. Therefore, the staff finds the applicant's assessment that no aging management activity is required for SCC of the TC to be acceptable.

Cracking Due to Chloride-Induced Stress-Corrosion Cracking

The applicant referenced operating experience documented in NRC Information Notice 2012-20 to show that austenitic stainless steels are susceptible to SCC in a chloride-containing environment under tensile stresses. The applicant recognized that airborne chloride salts deposited on a material surface could deliquesce at high relative humidity to initiate CISCC. However, the applicant stated that it is highly unlikely for CISCC to occur while the TC is currently stored in a sheltered environment at the site, which is far away from salt water bodies. Therefore, the applicant concluded that cracking due to CISCC is not an aging effect requiring management for TC subcomponents.

The staff reviewed the applicant's evaluation of conditions for CISCC to initiate and the environmental conditions during storage. The staff reviewed the applicant's assessment of the potential for CISCC and determined that the level of chloride-containing salt deposition on the TC in the sheltered environment where the TC is in storage is likely to be too low to initiate CISCC. Therefore, the staff finds the applicant's assessment that aging management activity is not required for CISCC of TC subcomponents to be acceptable.

Cracking Due to Thermal Fatigue

The applicant stated that thermal fatigue is the progressive and localized structural damage that occurs when a material is subjected to cyclic loading associated with thermal cycling. The applicant stated that the source of potential thermal fatigue of the TC may be caused by loading the DSC from the spent fuel pool and unloading the DSC into the HSM. The applicant evaluated the TC for pressure and temperature fluctuations in accordance with the provisions of NB-3222.4(d) of the ASME Code based on an average of 20 uses per year of a TC for a service life of 60 years. The applicant stated that, as provided by NB-3222.4(d), fatigue effects need not be specifically evaluated, provided the six criteria in NB-3222.4(d) are met. Because the six criteria of NB-3222.4(d) are met, the applicant concluded that cracking due to thermal fatigue is an aging effect managed through a TLAA.

The staff reviewed the applicant's calculations and the criteria from NB-3222.4(d) and finds that, because the criteria are met, thermal fatigue is not an aging effect requiring management for TC subcomponents. SER Section 3.4.2 documents the staff's review of the TLAA on the thermal fatigue of TC subcomponents.

Change in Material Properties Due to Thermal Aging

The applicant stated that steels in the TC are used in the annealed condition except for the precipitation-hardened trunnions and optionally cold-worked Nitronic® 60 rails, and the maximum TC temperatures are well below the levels that permanently affect the trunnions' mechanical properties. Therefore, the applicant concluded that thermal aging is not an aging effect requiring management for TC subcomponents.

As described in NUREG-2214 and discussed in SER Section 3.3.2.2.2, the staff recognizes that the exposure of austenitic stainless steel welds and carbon steels to elevated temperatures can result in metallurgical changes that can affect the mechanical properties of these materials. Because the peak temperatures for these materials in the TC subcomponents are below the temperature required to cause changes in material properties, the staff finds the applicant's assessment that aging management activity is not required for thermal aging of the TC subcomponents exposed to a sheltered environment to be acceptable.

Change in Material Properties Due to Irradiation Embrittlement

The applicant stated that high-neutron radiation can cause a loss of fracture toughness in steel. However, the applicant stated that the neutron radiation experienced by the steel components is orders of magnitude lower than that required to produce any significant effect. Furthermore, the applicant stated that the exposure of the TC to neutron fluence and gamma exposure was only limited to a short duration during loading and transfer operations, and it should be bound by the exposure on DSC materials discussed in SER Section 3.3.1.2. Therefore, the applicant concluded that irradiation embrittlement is not an aging effect requiring management for TC subcomponents.

The staff recognizes that neutron irradiation has the potential to increase the tensile and yield strength and decrease the toughness of carbon and alloy steels (Nikolaev et al., 2002). As discussed in SER Section 3.3.1.2 for the DSC subcomponents, the staff estimated that the neutron fluence level is three to four orders of magnitude below the levels reported to degrade the fracture resistance of carbon and alloy steels and stainless steels. The neutron fluence level that the TC is exposed to is expected to be even lower than that of the DSC subcomponents because the TC is currently in storage without the neutron exposure. Therefore, the staff finds the applicant's assessment that aging management activity is not required for radiation embrittlement of TC steel subcomponents exposed to a sheltered environment to be acceptable.

Change in Material Properties of Lead, NS-3, and Aluminum in Encased or Embedded Environment

The applicant did not identify a change in material properties as an aging effect requiring management for the lead, NS-3, and aluminum shielding materials because of the dry encased or embedded environment and insignificant changes induced by heat and radiation during fuel loading and transfer operations, based on the applicant's analysis.

The staff reviewed the TC drawings and confirmed that the lead is fully encased in metal and thus is not exposed to water or atmospheric contaminants. The staff notes that lead is well known to be resistant to corrosion in a variety of environments (Alhasan, 2005), and that lead is not susceptible to thermal or irradiation-induced material property changes under the exposures in the TC application. For the NS-3 material, the staff notes that the accumulated radiation dose in storage systems after 60 years (10^{13}–10^{15} n/cm^2) is several orders of magnitude below the 1.5×10^{19} n/cm^2 thermal neutron limit cited in the NS-3 product specification sheet (BISCO, 1986; NRC, 2014b). The staff also notes that the shielding materials are subject to elevated temperatures and radiation when the DSC is being transported from the spent fuel pool to the storage pad, and the staff recognizes that the cementitious BISCO NS-3 shielding material may experience some loss of hydrogen (neutron moderator). However, the time of thermal and radiation exposure is minimal during the brief period compared to the continuous exposures in other NRC-approved applications (e.g., NS-3 material in the MC-10 metal storage cask (NRC, 2005). The staff also reviewed the TC drawings and confirmed that the aluminum inner neutron shield plug (NSP) support angles are embedded in the NS-3 neutron shield material and thus are not exposed to a moist environment. The staff notes that corrosion of aluminum in an embedded environment is not considered to be credible because of limited exposure to water and oxygen. The staff also notes that the time the aluminum inner NSP support angles are subject to elevated temperatures and radiation exposure during the period of extended operation is negligible, and thus creep, thermal aging, radiation embrittlement, and fatigue are not credible. Therefore, because the temperature and radiation exposures experienced by the TC shielding materials are significantly lower than those that would be expected to alter shielding properties, the staff finds the applicant's assessment that change in material properties is not an aging effect requiring management to be acceptable.

Coating Degradation

The applicant stated that the DSC support rails of the TC are coated with a dry film graphite lubricant to minimize friction during insertion and retrieval of the DSC. However, the applicant stated that no credit is taken for lubricants for the performance of intended functions. Therefore, the applicant concluded that deterioration of dry lubricant is not an aging effect requiring aging management.

The staff reviewed the design of the TC with respect to the use of lubricant on DSC support rails and verified that they are not credited as supporting an important-to-safety function. Therefore, the staff finds the applicant's assessment that no aging management activity is required for coating degradation to be acceptable.

3.3.4.3 Proposed Aging Management Activities

The applicant provided a TLAA for the following potential aging effect for the TC:

- fatigue analysis of the NUHOMS® MP187 TC

The applicant evaluated TC fatigue according to the requirements of NB-3222.4(d) of the ASME Code. SER Section 3.4.2 includes the staff's review of the applicant's TLAA on fatigue of the TC materials. The staff reviewed the applicant's assessment and determined that analysis of this potential aging effect using a TLAA is acceptable.

The applicant provided a supplemental analysis for the following potential aging effect for the TC:

- Combustible gas generation in the neutron shield shell of the NUHOMS® MP187 TC

The applicant provided a supporting analysis, to determine the amount of combustible gases generated as a result of irradiation of neutron shield material for the MP187 TC during its function as a TC. SER Section 3.4.5 includes the staff's review of the applicant's supplemental analysis on combustible gas generation in the neutron shield shell for the TC. The staff reviewed the applicant's assessment and determined that analysis of this potential aging effect using a supplemental analysis is acceptable.

The applicant has also developed the TC AMP to address loss of material due to general, pitting, crevice, and galvanic corrosion, and wear. NUREG-1927, Revision 1 states that the use of an AMP is an acceptable approach to address aging degradation issues, and, therefore, the staff finds the applicant's use of an AMP to manage the effects of loss of material of TC subcomponents to be acceptable. SER Section 3.5 documents the staff's review of the TC AMP.

3.3.5 Spent Fuel Assemblies

The applicant described the SFAs in Section 2.4.4 of the LRA. The applicant stated that the FO/FC DSCs are designed to store 24 intact Babcock & Wilcox (B&W) 15×15 SFAs, while the FF DSC is designed to store 13 intact or damaged B&W 15×15 SFAs. The subcomponents of the assemblies include the fuel cladding, guide tubes, instrumentation tube assembly, spacer grid assembly, upper and lower end fitting, and related subcomponents. The maximum fuel burnup is 38.268 GWd/MTU, and the DSC design-basis heat load is less than 13.5 kilowatts.

3.3.5.1 Materials and Environments

Sections 2.4.4 and 3.8 of the LRA describe the materials and environments of the SFAs. The assemblies are constructed of zirconium-based alloys (fuel cladding, end plugs, guide tubes, instrument tubes, retainers, spacer sleeves), nickel-based alloys (spacer grid assemblies), and stainless steel (upper end fitting, lower end fitting, connectors).

The applicant identified external and internal environments that the SFAs experience normally and continuously. The applicant stated that the external environment of the SFAs refers to the internal DSC atmosphere that is an inert helium environment with trace amounts of water vapor and air. The applicant also stated that boric acid residue could be present on the surfaces of pressurized-water reactor SFAs, since they were exposed to a borated water environment in the spent fuel pool before storage. Any boric acid residue remaining on the SFAs will have no deleterious effects because of the absence of water and the materials used to construct the SFAs.

The applicant stated that the maximum cladding temperature at the beginning of storage for SFAs loaded in the FO/FC DSCs is 397 degrees C (746 degrees F) for off-normal conditions, which is greater than the normal temperature of 379 degrees C (714 degrees F) but less than the fuel cladding acceptance criteria of 570 degrees C (1,058 degrees F). The applicant also stated that the FF and GTCC DSC thermal analysis results are bounded by the FO/FC DSC thermal analysis results.

The applicant stated that the internal environment of the SFAs refers to the fuel rod interior. The applicant also stated that the fuel rod internal environment is assumed to be a combination of pressurized helium added during the manufacturing process and fission products produced during reactor operation.

The staff reviewed the applicant's description of the materials and environments for the SFAs to confirm that the description is consistent with the descriptions and engineering drawings in the

Rancho Seco ISFSI FSAR. Based on its confirmation of consistency with these design-basis documents, the staff finds the applicant's identification of the materials and environments for the SFAs to be acceptable.

3.3.5.2 Aging Effects and Mechanisms for the Spent Fuel Assemblies

In Section 3.8.5 of the LRA, the applicant stated that there are no aging effects requiring management for low-to-moderate burnup fuel (less than or equal to 45 GWd/MTU) that is stored in an inert environment during the period of extended operation. The applicant also stated that the EPRI Dry Cask Storage Characterization Project (EPRI, 2002) provides the basis for the assertion that the SFAs will not degrade to unacceptable levels during the period of extended operation. The applicant concluded that no AMPs or activities are credited during the period of extended operation for the Rancho Seco low-burnup SFAs and associated subcomponents.

The staff reviewed the applicant's conclusion that identified no aging effects requiring management for the SFAs. The staff notes that the demonstration project involved the visual inspection and material analysis of SFAs that were in dry storage for approximately 15 years. The staff based its review on NUREG-1927, Revision 1, and Interim Staff Guidance (ISG)-11, Revision 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel," issued November 2003 (NRC, 2003), which state that low-burnup fuel (less than or equal to 45 GWd/MTU) is expected to maintain its integrity for the period of extended operation, provided that the maximum cladding temperature limits cited in ISG-11 are followed. Based on its review of the LRA and the references cited above, the staff finds the applicant's aging management results for the SFAs to be acceptable.

3.3.6 Evaluation Findings

The staff reviewed the AMR in the LRA to verify that it adequately identified the materials, environments, and aging effects of the in-scope SSCs. The staff performed its review following the guidance in NUREG-1927, Revision 1, and NUREG-2214. Based on its review, the staff finds the following:

F3.1 The applicant's AMR process is comprehensive in identifying the materials of construction and associated operating environmental conditions for those SSCs within the scope of renewal, and the applicant has provided a summary of the information in the LRA and the FSAR supplement.

F3.2 The applicant's AMR process is comprehensive in identifying all pertinent aging mechanisms and effects applicable to the in-scope SSCs, and the applicant has provided a summary of the information in the LRA and the FSAR supplement.

3.4 Time-Limited Aging Analyses and Other Supporting Analyses

As discussed in Appendix A to the LRA, the applicant identified four TLAAs for SSCs within the scope of license renewal and provided one supporting analysis:

(1) fatigue analysis of the DSCs
(2) fatigue analysis of the TC
(3) DSC poison plates boron depletion analysis
(4) neutron fluence and gamma irradiation analysis
(5) combustible gas generation analysis

The staff reviewed the applicant's analyses in support of conclusions on potential aging effects for SSCs and SSC subcomponents within the scope of renewal. The staff reviewed the applicant's analyses to determine those meeting all six criteria under 10 CFR 72.3 for valid TLAAs. The staff also reviewed the applicant's supplemental analysis in support of the proposed AMPs.

Based on its review of the design-basis documents, the staff confirmed that the applicant identified all calculations and analyses meeting all six criteria in 10 CFR 72.3 and therefore concludes that the applicant adequately identified all TLAAs.

3.4.1 Fatigue Analysis of the Dry Shielded Canisters

Appendix A.2.1 to the LRA summarizes the results of the applicant's TLAA on fatigue evaluation of the DSCs. The applicant provided an analysis in accordance with the provisions of NB-3222.4(d) (ASME, 1992) and indicated that fatigue effects need not be evaluated specifically, as long as the six criteria contained in NB-3222.4(d), if applicable, are met. These six criteria are based on a comparison of peak stresses with strain cycling fatigue data and include cyclic stresses generated as a result of (1) atmospheric-to-service pressure cycles, (2) normal service pressure fluctuation, (3) temperature difference between startup and shutdown, (4) temperature difference in normal service, (5) temperature difference between dissimilar metals, and (6) mechanical loads. The applicant's evaluation used these six conditions to show that the ASME Code fatigue exemption requirements are satisfied for the DSCs.

The applicant demonstrated that the requirements mentioned in the six criteria listed in NB-3222.4(d) are satisfied for all three DCS designs (i.e., FO, FC, and FF). As a result, the applicant stated that a specific fatigue analysis for the DSCs is not warranted. The staff reviewed this information and verified that the six ASME criteria were satisfied and thus a more specific fatigue analysis is not warranted.

3.4.2 Fatigue Analysis of the Transfer Cask

Appendix A.2.2 to the LRA summarizes the results of the applicant's TLAA on the fatigue evaluation of the TC. The applicant evaluated the fatigue effects on TC components due to normal operating cycles using the criteria contained in NB-3222.4(d). The criteria state that fatigue effects need not be evaluated specifically, provided the six criteria in NB-3222.4(d) are met. These six criteria are based on a comparison of peak stresses with strain cycling fatigue data and include cyclic stresses generated as a result of (1) atmospheric-to-service pressure cycles, (2) normal service pressure fluctuation, (3) temperature difference between startup and shutdown, (4) temperature difference in normal service, (5) temperature difference between dissimilar metals, and (6) mechanical loads. The applicant's evaluation of these six criteria shows that the ASME Code fatigue exemption requirements are satisfied for the TC.

The applicant demonstrated that the requirements mentioned in the six criteria in NB-3222.4(d) of the ASME Code are satisfied for the TC, and those in NB-3232.3 for the TC bolts. As a result, the applicant stated that the TC and TC bolts need not be examined explicitly for fatigue. The staff reviewed this information and verified that the six ASME criteria were satisfied and thus a more specific fatigue analysis is not warranted.

3.4.3 Dry Shielded Canister Poison Plates Boron Depletion Analysis

The DSS at the Rancho Seco ISFSI uses Boral® as neutron poison plate material to assure that the system can perform its criticality safety function in accordance with the regulatory requirements of 10 CFR 72.124. The applicant determined, through its AMR, that the neutron poison plates, which are important to safety, are subject to aging management. The applicant performed a TLAA to evaluate the boron-10 depletion due to neutron irradiation for different thicknesses of the neutron poison plates. Appendix A.2.3.1 to the LRA summarized the results of the applicant's TLAA for the loss of boron-10 in the neutron poison plates due to neutron irradiation over 100 years. The applicant's calculation showed that the amount of boron-10 to be lost due to neutron irradiation from the spent fuel in the storage system is negligible compared to the initial concentration of boron-10 in the poison plate for any poison plate thickness that is used in the storage system design.

The staff reviewed the applicant's TLAA for the loss of boron-10 in the neutron poison plates of the spent fuel storage system for the extended period of operation following the guidance in NUREG-1927, Revision 1. The staff noted that the applicant's analyses for neutron radiation effects on materials are consistent with the analyses in NUREG-2214. By its own independent analysis, the staff also verified and confirmed that a negligible amount of boron-10 is depleted in the poison plates due to neutron irradiation.

Based upon a review of the applicant's information and the results of the staff's confirmatory calculations, the staff concludes that the loss of boron-10 due to irradiation during the period of the extended operation is negligible. Therefore, the staff finds there is no need for an AMP to manage the loss of the effectiveness of the neutron poison plates, and there is reasonable assurance that the spent fuel storage system will continue to remain subcritical during the period of extended operation of the Rancho Seco ISFSI.

3.4.4 Neutron Fluence and Gamma Irradiation Analysis

Appendix A.2.3.2 to the LRA summarized the results of the applicant's TLAA to determine the effect on ISFSI storage equipment (e.g., concrete structures, steel components) from neutron fluence and gamma exposure. The applicant generated neutron and gamma source terms using SCALE 5.0. The applicant conservatively calculated both the neutron fluence and the gamma exposure over 100 years of storage using a generic 32-assembly DSC containing B&W 15x15 fuel assemblies versus a Rancho Seco 24-assembly DSC loaded with B&W 15x15 fuel assemblies. In the analysis, the applicant also modeled a shorter distance between the HSM and the DSC compared to the actual distance at the Rancho Seco facility. The staff determined this to be acceptable because it is a conservative approximation. The applicant's calculations showed no credible mechanical degradation to the compressive strength and tensile strength of the DSC shell, shield plug, or HSM components.

The staff reviewed the applicant's TLAA for the impact of neutron fluence and gamma exposure on ISFSI storage equipment for the extended period of operation, following the guidance in NUREG-1927, Revision 1. The staff determined that the applicant's results showed good agreement with the analysis for concrete and steel components included in NUREG-2214. In addition, the staff independently performed confirmatory calculations to assess whether the source terms used by the applicant were reasonable. The neutron and gamma source terms generated by the staff showed good agreement with the applicant's source terms. Using its independently generated neutron source terms, the staff calculated a neutron fluence below the level of concern for steel embrittlement, which is reported to be 1×10^{18} n/cm^2, as well as below the level reported to result in mechanical degradation of concrete, which is 1×10^{19} n/cm^2. To evaluate the amount of energy deposited in the steel and concrete storage system components, the staff developed a model of the DSC inside the HSM using the MAVRIC module of SCALE 6.2.3. Using its independently generated neutron and gamma source terms in the MAVRIC model, the staff calculated a dose both to the DSC and to the concrete. Based upon the results of its confirmatory analyses, the staff determined the applicant's results to be conservative.

Based upon its review of the applicant's information and the results of its confirmatory calculations, the staff concludes that the neutron fluence and gamma irradiation during the period of extended operation will not adversely impact the steel and concrete storage system components to any measurable level. Therefore, the staff finds there is no need for an AMP for managing the loss of the effectiveness of the steel and concrete components, and there is reasonable assurance that the storage system will continue to perform its structural, shielding, and confinement functions during the period of extended operation of the Rancho Seco ISFSI.

3.4.5 Combustible Gas Generation Analysis

The DSS at the Rancho Seco ISFSI uses an MP187 TC with a neutron shield. The applicant determined through its AMR that the neutron shield, which is important to safety, is subject to aging management. Appendix A.2.4 to the LRA summarized the applicant's results for combustible gas generation in the TC neutron shield material due to radiolysis and the resulting loss of hydrogen in the neutron shield.

Using the neutron and gamma source terms generated with the SCALE computer codes, the applicant determined the amount of energy deposited in the neutron shield using the MCNP computer code. Based upon the amount of energy deposited in the neutron shield, the applicant stated that the hydrogen liberated from the neutron shield material over the service period of the cask due to irradiation is statistically insignificant when compared to the total hydrogen mass originally present in the neutron shield material. The applicant also explained that the neutron shield material would reabsorb some of the hydrogen generated by radiolysis. The applicant reached this conclusion by calculating the hydrogen solubility of the NS-3 neutron shield material using assumed temperature and pressure values, which the staff found reasonable, as well as a hydrogen solubility value for polymers obtained from a literature search. The applicant also hand-calculated the amount of hydrogen, total combustible gases, and all gases generated due to radiolysis of NS-3, using its analysis of the energy deposited in the TC neutron shield. Based upon the total amount of gases it calculated, the applicant concluded that the pressure inside the TC neutron shield would not exceed the burst pressure of the TC neutron shield rupture disk.

The staff reviewed the applicant's combustible gas generation analysis and found the assumptions and modeling techniques employed by the applicant in generating the neutron and gamma source terms, as well as in determining the amount of energy deposited in the TC neutron shield, to be both reasonable and conservative. The staff independently calculated neutron and gamma source terms using SCALE 6.2.3, and in general, the source terms generated by the staff showed good agreement with the applicant's source terms. The staff also developed a TC model using the MAVRIC module in SCALE 6.2.3 to evaluate the amount of energy deposited in the TC neutron shield. Using the results from SCALE 6.2.3, the staff hand-calculated both the amount of hydrogen and the total amount of gases generated by radiolysis in the TC neutron shield. Because hydrogen dominated the total combustible gases value presented by the applicant, the staff only evaluated the amount of hydrogen generated by radiolysis within the TC neutron shield.

Using a hydrogen G value of four molecules per 100 electron volts (eV) from Table 3.1 in NUREG/CR-6673, “Hydrogen Generation in TRU Waste Transportation Packages,” issued May 2000 (Anderson et.al., 2000), the staff calculated the amount of hydrogen generated within the TC neutron shield by radiolysis to be roughly three times the amount of hydrogen calculated by the applicant. The staff compared this amount to the ability of the NS-3 neutron shield to reabsorb hydrogen as calculated by the applicant and determined that the NS-3 neutron shield has the ability to absorb twice the amount of hydrogen calculated by the staff. Therefore, the staff determined that neither a combustible nor an explosive environment will be generated during the period of extended operation. Using parameters provided in the applicant’s calculation in equation 2.2 of NUREG/CR-6673, the staff also calculated the total amount of gases generated due to radiolysis using a G value of 1.77 molecules per 100 eV. The staff determined that the total amount of gas generated will not exceed the burst pressure of the TC neutron shield rupture disk.

Based upon a review of the applicant’s information and the results of staff’s confirmatory calculations, the staff concludes that gas generation due to radiolysis will not result in a combustible gas mixture. Therefore, the staff finds there is no need for an AMP for managing gas buildup within the TC neutron shield, and there is reasonable assurance that the TC neutron shield will continue to perform its shielding function during the period of extended operation of the ISFSI at the Rancho Seco site.

3.4.6 Evaluation Findings

The staff reviewed the TLAAs in the LRA following the guidance in NUREG-1927, Revision 1, and NUREG-2214. The staff verified that the TLAA assumptions, calculations, and analyses were adequate and bound the environment, and aging mechanisms or aging effects for the pertinent SSCs. Based on its review, the staff finds the following:

F3.3 The applicant identified all pertinent aging mechanisms and effects pertinent to SSCs within the scope of renewal that involve TLAAs. The methods and values of the input parameters for the applicant’s TLAAs are adequate. Therefore, the applicant’s evaluation provides reasonable assurance that the SSCs will maintain their intended functions for the period of extended operation, require no further aging management activities, and meet the requirements in 10 CFR 72.42(a)(1).

3.5 Aging Management Programs

Under 10 CFR 72.42(a)(2), the applicant must describe AMPs for managing issues associated with aging that could adversely affect SSCs important to safety. The applicant proposed four AMPs:

(1) DSC External Surfaces AMP
(2) HSM AMP
(3) TC AMP
(4) Basemat AMP

Appendix B to the LRA described these AMPs (SMUD, 2019c). Appendix C to the LRA proposed changes to the Rancho Seco ISFSI FSAR (referred to as the FSAR supplement), which summarized these AMPs. As specified in the new license condition in SER Section 4, the AMPs summarized in the FSAR supplement will be incorporated in the Rancho Seco ISFSI FSAR after issuance of the license renewal.

3.5.1 Dry Shielded Cask External Surfaces Aging Management Program

The applicant credited the DSC External Surfaces AMP described in Appendix B.3 to the LRA for managing the aging effects for the external surfaces of the DSCs. The applicant stated that the purpose of the AMP is to manage the aging effects on the external surfaces of the DSC shell assembly. The applicant identified the materials and environments included in this AMP as the DSC shell assembly components, including the shell, external surfaces of the outer bottom cover plate, grapple ring, and outer top cover plate. The assembly is constructed of stainless steel exposed to a sheltered environment inside the HSM. Aging effects requiring management identified by the applicant include (1) loss of material due to crevice corrosion and pitting corrosion for stainless steel components, (2) loss of material due to galvanic corrosion for the DSC shell contacting graphite lubricant at the sliding rail surface, and (3) cracking due to SCC for stainless steel components.

The staff reviewed the adequacy of the DSC external surfaces AMP to address the identified aging mechanisms and effects for the storage cask against the criteria in Section 3.6.1 of NUREG-1927, Revision 1. The staff's evaluation of each of the AMP program elements follows.

(1) Scope of Program

The applicant's description of the scope of the program includes external surfaces of the DSC shell assembly that may be subject to loss of material and cracking. The areas of inspection include (1) fabrication welds and the associated HAZ, (2) crevice locations, (3) upper surface of the DSC shell where atmospheric particulates could settle, (4) the top and bottom ends of the cylinder, which are cooler, (5) outer bottom cover plate, grapple assembly, shear key, their welds, and HAZ, and (6) outer top cover plate, welds, and HAZ. The applicant stated that the inspections of the HAZ will be at least 2 inches on either side of the welds.

The staff reviewed the scope of the program to verify that the applicant adequately described the components covered under the program, as recommended in NUREG-1927, Revision 1. Based on the staff's confirmation that the applicant accurately and clearly specified the details of the components addressed under the program, the staff finds the scope of the program to be acceptable.

(2) Preventive Actions

The applicant stated that the program does not include any preventive actions and is a condition-monitoring program.

The staff reviewed the preventive actions program element and confirmed that the program does not rely on preventive actions to manage the effects of aging. The staff notes that the program uses visual inspections and augmented examinations to manage loss of material and cracking. The staff finds the applicant's preventive actions program element to be acceptable because, consistent with the recommendations in NUREG-1927, Revision 1, preventive actions are not needed for condition-monitoring programs.

(3) Parameters Monitored or Inspected

The applicant stated that it will perform remote visual inspections of the following normally non-accessible areas for loss of material and cracking, including: (1) DSC surfaces, welds and HAZs, and crevice locations near the DSC support rails; (2) portions of the outer top cover plate, closure weld and HAZ; (3) outer bottom cover plate, grapple ring assembly, shear key, their welds and HAZs; and (4) portions of the DSC shell bottom surface. The applicant stated that there are some inaccessible areas that will not be inspected by remote visual inspection, including: (1) the upper surface of the DSC shell, (2) the majority of outer top cover plate, welds and HAZs, (3) DSC shell crevice locations.

The staff reviewed the parameters monitored or inspected program element of the normally non-accessible areas to confirm that the parameters will be capable of identifying degradation before a loss of intended function and provide a clear link to the aging effects identified in the scope of the program, as recommended in NUREG-1927, Revision 1. The staff notes that the normally non-accessible areas inspected are sufficient to address possible aging effects in the inaccessible location because (1) crevice locations near the DSC support rails may be used as an indication for the inaccessible areas and (2) the acceptance criteria are conservative (for example, a pit identified by visual inspection is an indication of localized corrosion and is viewed as a likely precursor to SCC; however, the pit is not expected to penetrate the canister wall). Furthermore, if needed, a limited portion of the top of the DSC may be accessed by removing the HSM door and installing temporary shielding in a similar manner as was done at Calvert Cliffs in 2012. In addition, inspection tools continue to be developed, and the areas listed as inaccessible may be accessible with tools to be developed. These approaches can be used if the results of future inspections indicate that additional inspection is necessary. As such, the staff finds the applicant's parameters monitored or inspected program element to be acceptable because loss of material and cracking due to corrosion will lead to discontinuities and imperfections, and visual inspections are capable of identifying the initiation or progression of loss of material or cracking of the DSC external surfaces.

(4) Detection of Aging Effects

The applicant stated that the detection of the loss of material and cracking aging effects relies on visual inspection procedures consistent with the ASME Code Section XI, IWA-2200 (ASME, 2013).

Selection of DSC(s) for Inspection

The applicant selected the same DSC inspected during the preapplication inspection for the baseline and subsequent inspections. The applicant stated that this DSC has the longest time in service, low initial heat load, identified welds, and the potential for ambient contaminant accumulation on the surface. The applicant considered that this DSC is bounding, and the selection criteria will be updated as necessary to incorporate new information.

Inspection Methods

The applicant stated that standard visual exams and augmented examination will be the inspection methods. Visual inspections follow procedures consistent with ASME Code Section XI, IWA-2200. VT-3 visual examinations can be initially performed remotely to detect discontinuities and imperfections on the surface, including corrosion, by camera. Additional VT-1 visual examinations are performed when indicated by the assessment of the VT-3 results.

The applicant stated that it will detect the aging effects for inaccessible areas indirectly by monitoring the inspection findings in normally non-accessible areas, and the inspections are only conducted if SMUD's Corrective Action Program determines it is necessary to ensure the component's intended function is maintained. The applicant stated that it will conduct an augmented examination on a DSC when the visual examinations indicate the presence of major or minor corrosion using qualified personnel and qualified examination procedures.

Inspection Timing and Frequency

The applicant stated that the baseline visual inspection will occur no later than 2 years after entering the period of extended operation, and the follow-on inspections are at a 10±2-year interval, which is consistent with ASME Code Section XI, IWA-2430. The applicant also stated that the inspection interval is decreased to 5±1 years if the preceding inspection identified a major corrosion indication or if an engineering evaluation calculation shows that the crack will reach 75 percent through-wall.

Staff Evaluation

For the selected DSC for preapplication, baseline, and any subsequent inspections, the staff considers that the selected DSC is reasonably bounding for the expected extent of aging degradation due to heat load and time in service. For the normally non-accessible areas, the staff considers visual inspections performed in accordance with ASME Code Section XI, VT-1 and VT-3, to be a proven and widely accepted approach to identifying degradation of nuclear components. The staff notes that the results of the applicant's preapplication inspection showed that no degradation was found after 15 years. In the evaluation of the timing of the inspections, the staff reviewed the results and details of the preapplication inspections to ensure that they were sufficiently rigorous to provide a valid baseline of the condition of the DSC. The staff also notes that the follow-on inspection interval of 10±2 years is consistent with the ASME Code and industry practice. EPRI (2014) shows that, even under aggressive chemical conditions, the time to grow a 50-percent through-wall CISCC crack is more than 10 years; however, the environmental conditions at Rancho Seco ISFSI are not aggressive. As discussed in SER Section 3.3.1.2, the Rancho Seco ISFSI only has a CISCC susceptibility ranking of "2" on the scale of 1–10, with "10" being the highest susceptibility based on the criteria developed by EPRI (2015).

In summary, the staff finds the detection of the aging effects program element to be acceptable because the AMP uses techniques capable of identifying degradation, and the applicant will conduct inspections at a frequency that supports timely identification of degraded conditions and implementation of corrective actions, consistent with the guidance in NUREG-1927, Revision 1.

(5) Monitoring and Trending

The applicant stated that the inspections and monitoring activities are performed periodically to identify areas of degradation. The applicant further stated that the SMUD Corrective Action Program collects information on conditions adverse to quality noted during the inspection and monitoring activities. Visual inspections consider cumulative operating experience from previous inspections and assessments to monitor and trend the progression of aging effects over time.

The staff reviewed the applicant's monitoring and trending activities to ensure that they provide for an evaluation of the extent of aging and the need for timely corrective or mitigative actions.

The staff notes that the baseline and periodic inspections offer sufficient opportunity to identify adverse trends so that corrective actions can be implemented before a loss of functions. The staff also notes that the preapplication visual inspections and trending future inspection results against that baseline can effectively evaluate and respond to any identified effects of aging. On these bases, the staff finds the monitoring and trending program element to be acceptable because activities will be in place to ensure that the applicant adequately evaluates the rate of degradation so that it will conduct future inspections or repair components before a loss of functions, consistent with the guidance in NUREG-1927, Revision 1.

(6) Acceptance Criteria

The applicant established the acceptance criteria following the aging management guidance in EPRI Report 3002008193, "Aging Management Guidance to Address Potential Chloride-Induced Stress Corrosion Cracking of Welded Stainless Steel Canisters," issued 2017 (EPRI, 2017). Although the focus is on CISCC, the applicant stated that other atmospheric corrosion mechanisms, such as pitting corrosion, are conservatively addressed by the three tiers of acceptance criteria: (1) visual examination criteria, (2) augmented examination criteria, and (3) flaw evaluation criteria.

<u>Visual Examination Criteria</u>

The applicant stated that, according to the criteria, the indication is classified as a major, minor, or insignificant corrosion indication. The presence of a major corrosion indication anywhere on the DSC, such as cracking of any size, corrosion products with a linear or branching appearance, or a minor corrosion indication within 2 inches of a weld, will result in a supplemental surface or volumetric examination for the presence of cracking. The applicant will enter the indication in SMUD's Corrective Action Program. Furthermore, the applicant stated that a minor corrosion indication more than 2 inches from a weld will receive a supplemental VT-1 examination to show that there is no attack under the corrosion indication. If there is, the applicant will also enter it in SMUD's Corrective Action Program. According to the criteria, the applicant stated that insignificant corrosion indications that do not meet the criteria for being either major or minor are acceptable without further action.

<u>Augmented Examination Criteria</u>

The applicant stated that, if a surface examination is performed under an augmented examination, no further actions are required if any of the following apply: (1) there is a confirmed absence of flaws, (2) a detected flaw is a rounded indication, and if no corrosion products or masking deposits are present, or (3) a detected flaw is a linear indication, if no corrosion products or masking deposits are present, and if the linear indication is determined not to have a crack-like morphology. Furthermore, the applicant stated that, if a volumetric examination is performed, no further actions are required if any of the following apply: (1) there is a confirmed absence of planar flaws, (2) a detected indication is determined not to be connected to the exterior of the DSC, (3) the entirety of the detected flaw is in an area that is confirmed to have no corrosion products present and to have no crack-like morphology, or (4) the detected indication was recorded before being mitigated or remediated and there has been no measurable increase in the flaw size after being remediated.

The applicant stated that, if none of the above volumetric examination conditions apply, the detected indication is associated with the outside surface connected planar flaw and is considered material cracking. In this case, an engineering evaluation will be performed to demonstrate the acceptability of the cracking indication using the flaw evaluation criteria described below.

Flaw Evaluation Criteria

In accordance with the criteria, the applicant stated that, if a crack is identified, an engineering evaluation will be performed to determine when the flaw will reach 75 percent of through-wall thickness. If the flaw is measured to be greater than 75 percent, or if it is not feasible to perform a supplemental inspection before the engineering evaluation can determine the flaw will reach 75 percent through-wall, the applicant will address the condition in accordance with SMUD's Corrective Action Program. In addition, the applicant referred to a review of industry operating experience, repair experience, or generic industry analyses relating to the consequences of through-wall flaws.

Staff Evaluation

The staff reviewed the applicant's acceptance criteria following the guidance in EPRI Report 3002008193 to verify that they provide specific benchmarks to prompt corrective actions before a loss of intended functions. The staff notes that the criteria are consistent with the parameters monitored, and they can be detected using the inspection methods detailed in the detection of aging effects program element. Therefore, the staff finds the acceptance criteria program element to be acceptable because it provides clear criteria against which to evaluate the need for corrective actions, consistent with the recommendations in NUREG-1927, Revision 1.

(7) Corrective Actions

The applicant stated that corrective actions are in accordance with SMUD quality assurance (QA) procedures and review and approval processes and that administrative controls are implemented according to the requirements of the Rancho Seco Quality Manual. The applicant further stated that SMUD's Corrective Action Program ensures that conditions adverse to quality are either corrected or are evaluated as acceptable for continued service through engineering analysis using the same methodology used in the licensing and design-basis calculations. The applicant recognized that an extent-of-condition investigation may trigger additional inspections using a different method, increased inspection frequency, or expanded inspection sample size, and the identification of major corrosion requires an expansion of the sample size to determine the extent of condition at the site. Furthermore, the applicant stated that a subject DSC with major corrosion that does not meet the prescribed evaluation criteria must be evaluated for continued service.

The staff reviewed the corrective actions element following the guidance in NUREG-1927, Revision 1. The staff finds the element to be acceptable because, consistent with NUREG-1927, Revision 1, inspection and monitoring results that do not meet the acceptance criteria will be entered in the SMUD Corrective Action Program, and the QA program provides reasonable assurance that corrective actions will be adequate to manage the aging of the DSC.

(8) Confirmation Process

The applicant stated that procedural controls are in place to ensure that it reviews the responses to corrective action assignments entered into SMUD's Corrective Action Program and verifies the adequacy of the response. The applicant stated that it will review condition reports for trending purposes and will establish a tollgate to assess the effectiveness of corrective actions and update the AMP, as necessary, on a periodic basis.

The staff notes that NUREG-1927, Revision 1, states that NRC-approved QA programs are an accepted approach to ensuring that the effectiveness of corrective actions are verified and that adverse trends are monitored to address potential degradation before a loss of function. The staff also notes that performing periodic AMP effectiveness reviews is consistent with the guidance in NUREG-1927, Revision 1, which recommends that programs incorporate future reviews of operating experience to maintain effectiveness. Therefore, the staff finds the confirmation process program element to be acceptable.

(9) Administrative Controls

The applicant stated that the administrative controls are conducted in accordance with the requirements of a QA program under 10 CFR Part 72, Subpart G, “Quality Assurance,” and will continue for the period of extended operation. The applicant stated further that the Rancho Seco Quality Manual meets this requirement.

The staff notes that NUREG-1927, Revision 1, states that NRC-approved programs under 10 CFR Part 72, Subpart G, are an accepted approach to providing adequate review, approval, and fulfillment of activities that ensure SSCs will continue to perform satisfactorily in service. On this basis, the staff finds the administrative controls program element to be acceptable.

(10) Operating Experience

The applicant stated that Section 3.2 of the LRA evaluates operating experience for DSCs and supports a conclusion that no other actions will be necessary to adequately detect aging effects and mechanisms other than those prescribed above.

The staff reviewed the operating experience program element to ensure that the applicant considered past operating experience appropriately and that the program provides for future reviews of operating experience to confirm the program’s continued effectiveness. The staff notes that no degradation of the DSC was identified in the applicant’s preapplication inspection. The staff also notes that the applicant’s proposal to share operating experience with the industry and conduct periodic AMP effectiveness reviews is consistent with the guidance in NUREG-1927, Revision 1, which recommends that programs incorporate future reviews of operating experience to ensure continued effectiveness. Based on the applicant’s commitments, the staff finds the operating experience program element to be acceptable because the applicant provided sufficient prior operating experience to support the effectiveness of AMP activities and included a framework for future operating experience reviews to ensure that AMPs will be adjusted as knowledge becomes available from new analyses, experiments, and inspection activities.

The staff concludes that (1) the applicant adequately addressed the 10 program elements of an AMP described in NUREG-1927, Revision 1, and (2) there is reasonable assurance that the AMP is adequate for managing the aging mechanisms and effects of the in-scope SSCs identified by the AMR, such that the in-scope SSCs will continue to perform their intended functions during the requested period of extended operation.

3.5.2 Horizontal Storage Module Aging Management Program

The applicant credited the HSM AMP described in Appendix B.4 to the LRA for managing the aging effects for the HSM concrete and steel subcomponents. The applicant stated that the purpose of the AMP is to manage the aging effects on the internal and external surfaces of the HSMs. Aging effects requiring management identified by the applicant include (1) cracking due to corrosion of embedded steel and ASR for reinforced concrete structures, (2) loss of material due to aggressive chemical attack and corrosion of embedded steel, (3) increase in porosity and permeability due to aggressive chemical attack and leaching of calcium hydroxide, (4) cracking of the welds on the Nitronic® 60 rail face due to SCC, and (5) loss of material due to general, pitting, and crevice corrosion of steel subcomponents.

The staff reviewed the AMP in accordance with the guidance in Section 3.6.1 of NUREG-1927, Revision 1. The staff's evaluation of each of the program elements follows.

(1) Scope of Program

The applicant stated that the scope of the program includes visual inspection of accessible and normally non-accessible concrete and steel components, including HSM walls, roof, and floor slab; HSM access door; DSC support structure and rail assembly; heat shields; embedments; and anchorages (such as bolts and mounting hardware). The applicant also stated that the program consists of periodic visual inspections by personnel qualified to monitor structures and components for applicable aging effects and mechanisms.

In response to an RAI (SMUD, 2019a), the applicant stated that it performs quarterly radiation surveys of the Rancho Seco ISFSI in accordance with site procedures that require measuring the radiation dose rate inside the ISFSI perimeter fence as well as reading area monitoring badges every 3 months. The applicant also stated that these surveys would detect any significant decrease in HSM shielding effectiveness in normally non-accessible areas in the HSMs that are not periodically inspected.

The staff reviewed the scope of the program to verify that the applicant adequately described the subcomponents within the scope of the program and the aging effects and mechanisms to be managed by the program, as recommended in NUREG-1927, Revision 1. Based on confirmation that the applicant accurately and clearly specified the details of the subcomponents addressed under the program, the staff finds the scope of the program to be acceptable.

(2) Preventive Actions

The applicant stated that the program does not include any preventive actions and is a condition-monitoring program.

The staff reviewed the preventive actions program element and confirmed that the program does not rely on preventive actions but uses visual inspections to manage the aging effects. The staff finds the applicant's preventive actions program element to be acceptable because, consistent with the recommendations in NUREG-1927, Revision 1, preventive actions are not needed for condition-monitoring programs.

(3) Parameters Monitored or Inspected

The applicant stated that it will conduct a direct visual inspection of the accessible areas of the HSM, which include (1) the external concrete surfaces of the HSM roof and walls, (2) external surfaces of the HSM access door, and (3) attachment hardware. The applicant also stated that remote visual inspection of the normally non-accessible areas of the HSM for loss of material and cracking includes (1) portions of the concrete floor slab and visible areas of front, back, and side walls and (2) portions of the DSC support structure and attachment hardware. The applicant further stated that the HSM surface areas that are inaccessible for direct and remote visual inspections include (1) the internal surface of the HSM roof blocked from view due to the upper heat shield and (2) heat shields at the internal surface of the roof and side walls.

The applicant specified that parameters monitored for the HSM concrete structures include (1) loss of material (spalling and scaling) due to corrosion of embedded steel or aggressive chemical attack, (2) cracking due to expansion from reaction with aggregates or corrosion of embedded steel, and (3) change in material properties due to leaching of calcium hydroxide or aggressive chemical attack. The applicant also specified that parameters monitored for metallic surfaces of the HSM include (1) loss of material due to general, pitting, crevice, and galvanic corrosion of carbon steel and stainless steel subcomponents, (2) loss of material due to general, pitting, and crevice corrosion of stainless steel (Nitronic® 60) subcomponents, (3) cracking due to SCC of the welds attaching the Nitronic® 60 rail face, and (4) loose or missing anchors and missing or degraded grout.

The staff reviewed the parameters monitored or inspected program element to confirm that the parameters will be capable of identifying degradation before a loss of intended function and provide a clear link to the aging effects identified in the scope of the program. The staff notes that, in the AMP's acceptance criteria program element, the applicant stated that the visual inspections would use the acceptance criteria consistent with ACI 349.3R to identify the specific conditions that are acceptable without the need for further evaluation, such as the absence of evidence of leaching and quantitative measures for acceptable dimensions for voids, scaling, and cracks. The staff also notes that the visual inspections would use the acceptance criteria in accordance with ASME Code Section XI, IWF-3400 (ASME, 2013), for HSM steel subcomponents to identify indications of relevant degradation requiring further evaluation, such as corrosion and material wastage. The staff determined that the inspection requirements in ASME Code Section XI, Subsection IWF, which is intended for class 1, 2, 3, and metal containment component supports of light-water-cooled plants is appropriate for the support structures used in HSMs. The staff finds the applicant's parameters of the monitored or inspected program element to be acceptable because the visual inspections will monitor conditions in a manner consistent with the recommendations in NUREG-1927, Revision 1, and the ACI and ASME standards.

(4) Detection of Aging Effects

The applicant stated that it will detect cracking, loss of material, and indications of changes in material properties by direct or remote visual inspections of the HSM concrete in both external and sheltered environments using Section 3.5.1 of ACI 349.3R (ACI, 2010), as appropriate. The applicant also stated that VT-3 direct or remote visual inspections of HSM steel surfaces will detect discontinuities and imperfections on the surfaces, following procedures consistent with ASME Code Section XI, IWA-2200 (ASME, 2013).

The applicant indicated that it can inspect the normally non-accessible internal surfaces of the HSM concrete using a video camera, fiber-optic scope, or other remote inspection technology through existing access points of the HSM. The remote inspection system is qualified and demonstrated to have sufficient resolution capability and enhanced lighting to resolve the acceptance criteria. The applicant stated that it will develop crack maps with a photographic record and physical dimensions for the HSM concrete using the guidance in ACI 224.4R, "Guide to Design Detailing to Mitigate Cracking," issued 2013 (ACI, 2013), and will monitor and trend them as a means of identifying the progressive growth of defects as evidence of degradation due to specific aging effects, such as rebar corrosion. The applicant also stated that it will detect the aging effects for inaccessible areas indirectly by monitoring the inspection findings in accessible and normally non-accessible areas and will only conduct the inspections if the SMUD Corrective Action Program determines it is necessary to ensure the component's intended function is maintained.

The applicant stated that the baseline visual inspection will occur no later than 2 years after entering the period of extended operation and the follow-on inspections are at a 10±2-year interval. The applicant also stated that the interval between HSM inspections is decreased to 5±1 years if the preceding inspection acceptance criteria have been exceeded or the trending from previous inspections is unclear.

In response to an RAI, the applicant stated that it will inspect the accessible surfaces of all of the HSMs and the normally non-accessible surfaces of at least one HSM (SMUD, 2019a, c). The applicant stated that the 10-year inspection frequency referred to in ACI 349.3R for below-grade structures and structures in controlled interior environments is appropriate for the inspection frequency of the exterior of the HSM, based on the lack of an aggressive environment and the physical conditions of the HSM. Based on the results of the preapplication inspections, the applicant stated that the HSMs are expected to perform their intended safety functions through the period of extended operation without any significant aging-related degradation due to the benign environment. Therefore, the applicant concluded that an inspection frequency of 10±2 years is justified for the exterior of the HSMs.

The staff reviewed the detection of aging effects program element to confirm that the applicant adequately described the inspection details, including the methods used, inspection frequency, and inspection timing, in a manner consistent with the recommendations in NUREG-1927, Revision 1. The staff notes that the use of the direct or remote visual inspection techniques specified in ACI 349.3R is consistent with the recommendations in NUREG-1927, Revision 1, for the inspection of the HSM concrete. Similarly, the performance of VT-3 direct or remote visual inspections in accordance with the ASME Code is considered to be a proven and widely accepted approach to identifying the degradation of HSM steel subcomponents.

With respect to the proposed inspection frequency of the HSM concrete, the staff notes that the inspection interval of the accessible and normally non-accessible areas of the HSM concrete exceeds the interval of at least once every 5 years recommended in ACI 349.3R and the example AMP for reinforced concrete structures in Table B-2 of NUREG-1927, Revision 1. In its review of the proposed inspection frequencies, the staff reviewed operating experience associated with concrete inspections. The staff notes that the results of the applicant's preapplication inspection showed that no degradation of the HSM concrete was found after 15 years of operation. The staff notes that degradation of the DSC support structure, which was limited to isolated areas of coating degradation and minor rust staining, was found after 15 years of operation. In the evaluation of the timing of the inspections, the staff reviewed the results and details of the preapplication inspections to ensure that these inspections were sufficiently rigorous to provide a valid baseline for the condition of the concrete. The staff finds the detection of aging effects program element to be acceptable because the AMP uses techniques capable of identifying degradation, and the frequency of inspections will support the timely identification of degraded conditions and implementation of corrective actions, consistent with the guidance in Section 3.6.1.4 of NUREG-1927, Revision 1.

(5) Monitoring and Trending

The applicant stated that it conducts inspections and monitoring activities periodically to identify areas of degradation. The applicant also stated that the SMUD Corrective Action Program collects information on the conditions adverse to quality identified during the inspection and monitoring activities. Visual inspection assessments will consider cumulative operating experience from previous inspections and assessments to monitor and trend the progression of aging effects over time. The applicant will compare the data obtained during these inspections to previous inspection data from the site as well as to industry operating experience.

The staff reviewed the applicant's monitoring and trending activities to ensure that they will evaluate the extent of aging and the need for timely corrective or mitigative actions. The staff notes that the baseline and periodic inspections offer sufficient opportunity to identify adverse trends so that corrective actions can be implemented before a loss of functions. The staff also notes that the preapplication visual inspections and trending future inspection results against that baseline can effectively evaluate and respond to any identified effects of aging. The staff further notes that the applicant's Corrective Action Program is implemented to ensure that conditions adverse to quality are promptly identified and corrected. As a result, the staff finds the monitoring and trending program element to be acceptable because activities will be in place to ensure that the applicant adequately evaluates the rate of degradation so that it will conduct future inspections or repair components before a loss of functions, consistent with the guidance in NUREG-1927, Revision 1.

(6) Acceptance Criteria

The applicant stated that the acceptance criteria for the visual inspections of the HSM concrete are those specified in ACI 349.3R. The applicant also stated that the findings from a visual inspection within the first-tier criteria of ACI 349.3R are considered acceptable without requiring any further evaluation. These visual inspection findings include the absence of evidence of leaching and chemical attack, signs of corrosion, and drummy areas; quantitative thresholds for acceptable dimensions of pop-outs, voids, scaling, spalling, and cracks; and no evidence of excessive deflections in the concrete. The applicant further stated that the second-tier criteria of ACI 349.3R provide acceptable conditions for observed degradation that has been determined to be inactive.

The applicant also established the acceptance criteria for the VT-3 inspections of HSM steel subcomponents in accordance with ASME Code Section XI, IWF-3400. The applicant stated that, for metallic surfaces of the HSM, further evaluation through the SMUD Corrective Action Program is required if any of the following indications of degradation are detected: (1) corrosion and material wastage, (2) crevice, pitting, and galvanic corrosion, (3) corrosion stains on adjacent components and structures, (4) surface cracks, and (5) stains caused by leaking rainwater if evidence of corrosion is exhibited.

The staff reviewed the applicant's acceptance criteria to verify that they provide specific benchmarks to prompt corrective actions before a loss of intended functions. The staff notes that the criteria are consistent with those recommended in NUREG-1927, Revision 1. Therefore, the staff finds the acceptance criteria program element is acceptable.

(7) Corrective Actions

The applicant stated that it takes corrective action in accordance with SMUD QA procedures and review and approval processes and that it applies administrative controls according to the requirements of the Rancho Seco Quality Manual. The applicant also stated that its Corrective Action Program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determinations and prevention of recurrence. Deficiencies are either corrected or are evaluated as acceptable for continued service through engineering analysis using the same methodology used in the licensing and design-basis calculations. The applicant recognized that an extent of condition investigation may trigger additional inspections using a different method, increased inspection frequency, or expanded inspection sample size. The applicant stated that repair, restoration, or corrective action of an unacceptable condition will be performed consistent with ACI 224.4R and ASME Code Section XI, IWA-4000.

The staff reviewed the corrective actions program element following the guidance in NUREG-1927, Revision 1. The staff finds the element to be acceptable because, consistent with NUREG-1927, Revision 1, inspection and monitoring results that do not meet the acceptance criteria will be entered in the SMUD Corrective Action Program, and the QA program provides reasonable assurance that corrective actions will be adequate to manage the aging effects of the HSM.

(8) Confirmation Process

The applicant stated that procedural controls are in place to ensure that the applicant reviews the responses to corrective action assignments entered into the SMUD Corrective Action Program and verifies the adequacy of the response. The applicant also stated that it reviews condition reports for trending purposes and will establish a tollgate to assess the effectiveness of corrective actions and update the AMP as necessary on a periodic basis.

The staff notes that NUREG-1927, Revision 1, states that NRC-approved QA programs are an accepted approach to ensuring that the effectiveness of corrective actions are verified and that adverse trends are monitored to address potential degradation before a loss of function. The staff also notes that performing periodic AMP effectiveness reviews is consistent with the guidance in NUREG-1927, Revision 1, which recommends that programs incorporate future reviews of operating experience to maintain effectiveness. Therefore, the staff finds the confirmation process program element to be acceptable.

(9) Administrative Controls

The applicant stated that the administrative controls are conducted in accordance with the requirements of a QA program under 10 CFR Part 72, Subpart G, and will continue for the period of extended operation. The applicant also stated that the Rancho Seco Quality Manual meets this requirement.

The staff notes that NUREG-1927, Revision 1, states that NRC-approved programs under 10 CFR Part 72, Subpart G, are an accepted approach to providing adequate review, approval, and fulfillment of activities that ensure SSCs will continue to perform satisfactorily in service. On this basis, the staff finds the administrative controls program element to be acceptable.

(10) Operating Experience

The applicant stated that Section 3.2 of the LRA evaluates operating experience for the HSMs, which supports an assessment that the effects of aging are adequately managed to the extent that the HSM design functions are maintained during the period of extended operation.

The staff reviewed the operating experience program element to ensure that past operating experience was appropriately considered and that the program provides for future reviews of operating experience to confirm the program's continued effectiveness. The staff notes that the degradation of the DSC support structure described in the applicant's preapplication inspection was minor (e.g., coating defects, rust stains) and is effectively addressed by the proposed AMP activities. The staff also notes that the applicant's proposal to share operating experience with the industry and conduct periodic AMP effectiveness reviews is consistent with the guidance in NUREG-1927, Revision 1, which recommends that programs incorporate future reviews of operating experience to ensure continued effectiveness. Therefore, the staff finds the operating experience program element to be acceptable because the applicant included sufficient prior operating experience to support the effectiveness of AMP activities and provided a framework for future operating experience reviews to ensure that AMPs will be adjusted as knowledge becomes available from new analyses, experiments, and inspection activities.

The staff concludes that (1) the applicant adequately addressed the 10 program elements of an AMP described in NUREG-1927, Revision 1, and (2) there is reasonable assurance that the AMP is adequate for managing the aging mechanisms and effects of the in-scope SSCs identified by the AMR, such that the in-scope SSCs will continue to perform their intended functions during the requested period of extended operation.

3.5.3 Transfer Cask Aging Management Program

The applicant credited the TC AMP described in Appendix B.5 to the LRA for managing the aging effects for the exposed surfaces of the only onsite TC at the Rancho Seco ISFSI, the MP187 TC. The applicant stated that the purpose of the AMP is to ensure that all the accessible TC subcomponent surfaces are intact and free from evidence of loss of material due to various aging mechanisms. The applicant identified the materials used in the TC as (1) stainless steel used to construct the shell assembly and rails, (2) carbon steel used to construct attachment hardware, (3) lead and NS-3 used as inner and outer annulus shielding materials, respectively, and (4) aluminum used to construct inner neutron shield plugs. Steels are exposed to a sheltered environment, and lead, NS-3, and aluminum are enclosed or embedded. Aging effects requiring management identified by the applicant include (1) loss of material due to general corrosion of carbon steel, (2) loss of material due to crevice corrosion and pitting corrosion for carbon steel and stainless steel components, (3) loss of material due to wear, and (4) loss of material due to galvanic corrosion for the stainless steel rail, inner shell, and bottom end closure.

In response to an RAI, the applicant stated that the MP187 TC does not serve a confinement function in the Rancho Seco ISFSI licensing basis; the MP187 TC is only allowed to be used as a transfer cask (SMUD, 2019a). The TC will remain in a static condition at the Rancho Seco site in a sheltered environment. Therefore, no additional environments or aging effects need to be considered in the AMR.

The staff reviewed the adequacy of the TC AMP to address the identified aging mechanisms and effects against the criteria in Section 3.6.1 of NUREG-1927, Revision 1. The staff's evaluation of each of the AMP program elements follows.

(1) Scope of Program

The applicant explained that the scope of the program includes performing VT-3 visual inspection of all the accessible TC subcomponent surfaces that may be subject to loss of material due to general, crevice, galvanic, or pitting corrosion and due to wear. The applicant stated that it would conduct the VT-3 visual inspections in accordance with ASME Code Section XI (ASME, 2013). The applicant stated that it will also perform a VT-1 visual inspection on any area where VT-3 inspections detect signs of degradation.

The staff reviewed the scope of the program to verify that the applicant adequately described the components covered under the program, as recommended in NUREG-1927, Revision 1. The staff also notes that both the interior and exterior surfaces of the TC are readily accessible. Based on confirmation that the applicant accurately and clearly specified the details of the components addressed under the program, the staff finds the scope of the program to be acceptable.

(2) Preventive Actions

The applicant stated that the program does not include any preventive actions and is a condition-monitoring program.

The staff reviewed the preventive actions program element and confirmed that the program does not rely on preventive actions to manage the effects of aging. The staff notes that the program uses visual inspections to manage loss of material. The staff finds the applicant's preventive actions program element to be acceptable because, consistent with the recommendations in NUREG-1927, Revision 1, preventive actions are not needed for condition-monitoring programs.

(3) Parameters Monitored or Inspected

The applicant stated that the parameters inspected by the program include visual evidence of degradation of accessible surfaces of the TC, including the trunnions. Specifically. the surfaces of the cask cavity inner liner are examined for surface conditions and for indications of corrosion or excessive wear. Fasteners are examined for surface conditions and for indications of damage. In addition, visual inspections of the external surfaces of the TC, bearing surfaces of the upper and lower trunnion assemblies, fasteners, and cask lid surfaces are performed before use for signs of degradation, including corrosion and wear. The TC rails are visually inspected for indications of corrosion or excessive wear.

The staff reviewed the parameters monitored or inspected element and determined that the specific parameters that will be monitored and inspected are identified and are capable of detecting loss of material of the TC subcomponents. The staff also notes that the parameters provide a clear link to the aging effects identified in the AMR and managed by the program. Therefore, the staff finds that the parameters monitored or inspected program element provides reasonable assurance that the effects of aging will be adequately managed.

(4) Detection of Aging Effects

The applicant stated that the detection of the loss of material aging effects relies on VT-3 and VT-1 visual inspection procedures consistent with ASME Code Section XI, IWA-2213 and IWA-2211.

The applicant stated that the program calls for visual inspection of the accessible surfaces of the TC, including the trunnions and the cask lid, for loss of material due to crevice, galvanic, pitting, and general corrosion and due to wear. The applicant stated that it will conduct inspections within 1 year before the fuel loading or unloading campaign or DSC retrieval from the HSMs for inspection or offsite shipment. The applicant will conduct the visual inspections for corrosion and wear in accordance with VT-3 examinations set forth in ASME Code Section XI, IWA-2213. If the external surfaces of the TC exhibit evidence of degradation, the applicant will conduct examinations in accordance with ASME Code VT-1, IWA-2211. It also will inspect fasteners for thread condition, corrosion, wear, or other degradation.

In response to an RAI (SMUD, 2019a), the applicant stated that an inspection within 1 year before the campaign is appropriate and provides flexibility in scheduling the inspection while still providing assurance that subcomponent aging will not prevent the fulfillment of an intended safety function.

The staff reviewed the activities associated with the detection of aging effects to ensure that they are sufficient to identify degradation before a loss of intended function. The staff notes that the use of the visual inspection criteria in ASME Code Section XI is consistent with the recommendations in NUREG-1927, Revision 1, for the inspection of metallic canister components for corrosion. Also, given that the TC is currently in storage, visual inspections before use within 1 year of the campaign are considered capable of identifying degradation before a loss of function. Therefore, the staff finds that the detection of aging effects program element provides reasonable assurance that the effects of aging will be adequately managed.

(5) Monitoring and Trending

The applicant stated that it conducts the inspections and monitoring activities to identify areas of degradation. The applicant further stated that the SMUD Corrective Action Program collects information on conditions adverse to quality noted during the inspection and monitoring activities. Visual inspections consider cumulative operating experience from previous inspections and assessments to monitor and trend the progression of aging effects over time.

The staff reviewed the applicant’s monitoring and trending activities to ensure that they provide an evaluation of the extent of aging and the need for timely corrective or mitigative actions. The staff notes that the inspection within 1 year of use provides sufficient opportunity to identify adverse trends so that corrective actions can be implemented before a loss of functions. On these bases, the staff finds the monitoring and trending program element to be acceptable because activities will be in place to ensure that the applicant will adequately evaluate degradation and conduct future inspections or repair components before a loss of functions, consistent with the guidance in NUREG-1927, Revision 1.

(6) Acceptance Criteria

The applicant stated that the acceptance criteria for the TC AMP include any indications of corrosion or wear from visual examinations. These indications are entered into the SMUD

Corrective Action Program to determine the extent and impact of the degradation on the ability of the TC to perform its intended function.

The staff reviewed the applicant's acceptance criteria to verify that they include specific benchmarks to prompt corrective actions before a loss of intended functions. The staff notes that the criteria are consistent with the parameters monitored and can be detected using the inspection methods detailed in the detection of aging effects program element. Therefore, the staff finds that the acceptance criteria specified in the TC AMP provide reasonable assurance that the effects of aging will be adequately managed.

(7) Corrective Actions

The applicant stated that corrective actions are in accordance with SMUD QA procedures and review and approval processes, and that administrative controls are implemented according to the requirements of the Rancho Seco Quality Manual. The applicant further stated that SMUD's Corrective Action Program ensures that conditions adverse to quality are either corrected or are evaluated as acceptable for continued service through engineering analysis using the same methodology used in the licensing and design-basis calculations.

The staff reviewed the corrective actions program element following the guidance in NUREG-1927, Revision 1. The staff finds the element to be acceptable because, consistent with NUREG-1927, Revision 1, inspection and monitoring results that do not meet the acceptance criteria will be entered in the SMUD Corrective Action Program, and the QA program provides reasonable assurance that corrective actions will be adequate to manage the aging of the TC.

(8) Confirmation Process

The applicant stated that procedural controls are in place to ensure the review of responses to corrective action assignments entered into SMUD's Corrective Action Program and the verification of the adequacy of the response. The applicant stated that it reviews condition reports for trending purposes and will establish a tollgate to assess the effectiveness of corrective actions and update the AMP as necessary on a periodic basis.

The staff notes that NUREG-1927, Revision 1, states that NRC-approved QA programs are an accepted approach to ensuring that the effectiveness of corrective actions are verified and that adverse trends are monitored to address potential degradation before a loss of function. The staff also notes that performing periodic AMP effectiveness reviews is consistent with the guidance in NUREG-1927, Revision 1, which recommends that programs incorporate future reviews of operating experience to maintain effectiveness. Therefore, the staff finds the confirmation process program element to be acceptable.

(9) Administrative Controls

The applicant stated that the administrative controls are conducted in accordance with the requirements of a QA program under 10 CFR Part 72, Subpart G, and will continue for the period of extended operation. The applicant stated further that the Rancho Seco Quality Manual meets this requirement.

The staff notes that NUREG-1927, Revision 1, states that NRC-approved programs under 10 CFR Part 72, Subpart G, are an accepted approach to providing adequate review, approval, and fulfillment of activities that ensure SSCs will continue to perform satisfactorily in service. On this basis, the staff finds the administrative controls program element to be acceptable.

(10) Operating Experience

The applicant stated that Section 3.2 of the LRA evaluates operating experience for the TC, which supports a conclusion that no other actions will be necessary to adequately detect aging effects and mechanisms other than those prescribed above.

The staff reviewed the operating experience program element to ensure that the applicant considered past operating experience appropriately and that the program provides for future reviews of operating experience to confirm the program's continued effectiveness. The staff notes that the degradation described in the applicant's operating experience review was minor (e.g., coating defects, rust stains) and is effectively addressed by the proposed AMP activities. The staff also notes that the applicant's proposal to share operating experience with the industry and conduct periodic AMP effectiveness reviews is consistent with the guidance in NUREG-1927, Revision 1, which recommends that programs incorporate future reviews of operating experience to ensure continued effectiveness. Based on the applicant's commitments, the staff finds the operating experience program element to be acceptable because the applicant provided sufficient prior operating experience to support the effectiveness of AMP activities and included a framework for future operating experience reviews to ensure that AMPs will be adjusted as knowledge becomes available from new analyses, experiments, and inspection activities.

The staff concludes that (1) the applicant adequately addressed the 10 program elements of an AMP described in NUREG-1927, Revision 1, and (2) there is reasonable assurance that the AMP is adequate for managing the aging mechanisms and effects of the in-scope SSCs identified by the AMR, such that the in-scope SSCs will continue to perform their intended functions during the requested period of extended operation.

3.5.4 Basemat Aging Management Program

The applicant credited the Basemat AMP, described in Appendix B.6 to the LRA, for managing the aging effects on the basemat concrete. The applicant stated that the purpose of the AMP is to manage the aging effects on the above-grade portions of the basemat within the HSM array and the basemat extending around the perimeter of the HSM. In response to an RAI, the applicant explained that the basemat is exposed to sheltered, external, embedded, and underground environments (SMUD, 2019a). Aging effects requiring management identified by the applicant include (1) cracking due to corrosion of embedded steel, ASR, and settlement for reinforced concrete structures, (2) loss of material due to corrosion of embedded steel, DEF, and aggressive chemical attack, and (3) an increase in porosity and permeability due to aggressive chemical attack and leaching of calcium hydroxide.

The staff reviewed the AMP according to the guidance in Section 3.6.1 of NUREG-1927, Revision 1. The staff's evaluation of each of the program elements follows.

(1) Scope of Program

The applicant stated that the scope of the program includes visual inspection of the accessible above-grade portions of the basemat within the HSM array and the basemat extending around the perimeter of the HSM.

The staff reviewed the scope of the program to verify that the applicant adequately described the component within the scope of the program and the aging effects and mechanisms to be managed by the program, as recommended in NUREG-1927, Revision 1. Based on the staff's confirmation that the applicant accurately and clearly specified the details of the component addressed under the program, the staff finds the scope of the program to be acceptable.

(2) Preventive Actions

The applicant stated that the program does not include any preventive actions and is a condition-monitoring program.

The staff reviewed the preventive actions program element and confirmed that the program does not rely on preventive actions but uses visual inspections to manage the aging effects. The staff finds the applicant's preventive actions program element to be acceptable because, consistent with the recommendations in NUREG-1927, Revision 1, preventive actions are not needed for condition-monitoring programs.

(3) Parameters Monitored or Inspected

The applicant stated that it will conduct periodic visual monitoring to determine the surface condition of the ISFSI basemat, which is a leading indicator for the overall integrity of the basemat. The applicant stated that parameters monitored for the ISFSI basemat surface include (1) cracking and loss of material (spalling and scaling) due to corrosion of embedded steel, (2) cracking due to expansion from reaction with aggregates or settlement, (3) change in material properties due to leaching of calcium hydroxide or aggressive chemical attack, and (4) loss of material due to general, pitting, and crevice corrosion for carbon steel reinforcing bars, DEF, and aggressive chemical attack. The applicant further stated that changes in site conditions (e.g., water table, soil consolidation, construction) may cause settlement of the ISFSI basemat; therefore, cracking and distortion due to settlement is also a monitored parameter.

The staff reviewed the parameters monitored or inspected program element to confirm that the parameters will be capable of identifying degradation before a loss of intended function and provide a clear link to the aging effects identified in the scope of the program. The staff notes that, in the AMP's acceptance criteria program element, the applicant stated that the visual inspections would use the acceptance criteria consistent with ACI 349.3R to identify the specific conditions that are acceptable without the need for further evaluation, such as the absence of evidence of leaching and quantitative measures for acceptable dimensions for voids, scaling, and cracks. The staff finds the applicant's parameters monitored or inspected program element to be acceptable because the visual inspections will monitor conditions in a manner consistent with the recommendations in NUREG-1927, Revision 1, and ACI 349.3R.

(4) Detection of Aging Effects

The applicant stated that it will detect cracking, loss of material, and indications of changes in material properties of the concrete basemat by visual inspections of the above-grade portion of the concrete basemat using Section 3.5.1 of ACI 349.3R, as appropriate. The applicant also stated that the baseline visual inspection will occur no later than 2 years after entering the period of extended operation, and the follow-on inspections are at a 10±2-year interval. The applicant further stated that the inspection interval is decreased to 5±1 years if the preceding inspection acceptance criteria have been exceeded or the trending from previous inspections is indeterminate.

In response to an RAI, the applicant stated that it will address the detection of aging affects for inaccessible areas (e.g., beneath the HSM and below grade) using the inspection findings in accessible areas (SMUD, 2019a). If the applicant identifies aging effects in accessible locations, it will conduct further evaluation of the aging effects in inaccessible locations using its corrective action program to ensure the aging effect is adequately managed and that the component's intended function is maintained during the period of extended operation.

In response to an RAI, the applicant determined that the 10-year inspection frequency referred to in ACI 349.3R for below-grade structures and structures in controlled interior environments is appropriate for the inspection frequency of the exterior of the concrete basemat based on the lack of an aggressive environment and the physical conditions of the concrete basemat (SMUD, 2019a, c).

The staff reviewed the detection of aging effects program element and confirmed that the applicant adequately described the inspection details, including the methods used, inspection frequency, and inspection timing in a manner consistent with the recommendations in NUREG-1927, Revision 1. The staff notes that the use of visual inspection techniques specified in ACI 349.3R is consistent with the recommendations in NUREG-1927, Revision 1, for the inspection of the concrete basemat.

With respect to the proposed inspection frequency, the staff notes that the inspection interval of the above-grade portion of the concrete basemat exceeds the interval of at least once every 5 years recommended in ACI 349.3R and NUREG-1927, Revision 1. In its review of the proposed inspection frequencies, the staff reviewed operating experience associated with concrete inspections. The staff notes that the results of the applicant’s preapplication inspection showed that no degradation was found after 15 years. In the evaluation of the timing of the inspections, the staff reviewed the results and details of the preapplication inspections to ensure that these inspections were sufficiently rigorous to provide a valid baseline for the condition of the basemat. The staff finds the detection of aging effects program element to be acceptable because the AMP uses techniques capable of identifying degradation, and the applicant will conduct inspections at a frequency that supports timely identification of degraded conditions and implementation of corrective actions, consistent with the guidance in NUREG-1927, Revision 1.

(5) Monitoring and Trending

The applicant stated that it conducts inspections and monitoring activities periodically to identify areas of degradation. The applicant also stated that it will enter conditions adverse to quality identified during the inspection and monitoring activities into the SMUD Corrective Action Program. The applicant further stated that visual inspection assessments will consider cumulative operating experience from previous inspections and assessments to monitor and trend the progression of aging effects over time. Data obtained during these inspections will be compared to previous inspection data from the site as well as to industry operating experience.

The staff reviewed the applicant’s monitoring and trending activities to ensure that they provide for an evaluation of the extent of aging and the need for timely corrective or mitigative actions. The staff notes that the baseline and periodic inspections offer sufficient opportunity to identify adverse trends so that corrective actions can be implemented before a loss of functions. The staff also notes that the preapplication visual inspections and trending future inspection results against that baseline can effectively evaluate and respond to any identified effects of aging. The staff further notes that the SMUD Corrective Action Program is implemented to ensure that conditions adverse to quality are promptly identified and corrected. As a result, the staff finds the monitoring and trending program element to be acceptable because activities will be in place to ensure that the applicant will adequately evaluate the rate of degradation and conduct future inspections or repair components before a loss of functions, consistent with the guidance in NUREG-1927, Revision 1.

(6) Acceptance Criteria

The applicant stated that the acceptance criteria for the visual inspections of the concrete basemat are those specified in ACI 349.3R. The applicant also stated that the findings from a visual inspection within the first-tier criteria of ACI 349.3R are considered acceptable without requiring any further evaluation. These visual inspection findings include the absence of evidence of leaching and chemical attack, signs of corrosion, and drummy areas; quantitative thresholds for acceptable dimensions of pop-outs, voids, scaling, spalling, and cracks; and no evidence of excessive settlements or deflections in the basemat. The applicant further stated that the second-tier criteria of ACI 349.3R provide acceptable conditions for observed degradation that has been determined to be inactive.

The staff reviewed the applicant’s acceptance criteria to verify that they provide specific benchmarks to prompt corrective actions before a loss of intended functions. The staff notes that the criteria are consistent with those recommended in NUREG-1927, Revision 1. Therefore, the staff finds the acceptance criteria program element to be acceptable.

(7) Corrective Actions

The applicant stated that corrective actions are in accordance with SMUD QA procedures and review and approval processes, and that administrative controls are implemented according to the requirements of the Rancho Seco Quality Manual. The applicant also stated that the SMUD Corrective Action Program ensures that conditions adverse to quality are promptly identified and corrected, including root cause determinations and prevention of recurrence. Deficiencies are either corrected or are evaluated as acceptable for continued service through engineering analysis using the same methodology as in the licensing and design-basis calculations. The applicant recognized that an extent of condition investigation may trigger additional inspections using a different method, increased inspection frequency, or an expanded inspection sample size. The applicant further stated that it will perform the repair, restoration, or corrective action of an unacceptable condition consistent with ACI 224.4R and ASME Code Section XI, IWA-4000.

The staff reviewed the corrective actions program element following the guidance in NUREG-1927, Revision 1. The staff finds the element to be acceptable because, consistent with NUREG-1927, Revision 1, inspection and monitoring results that do not meet the acceptance criteria will be entered into the SMUD Corrective Action Program, and the QA program provides reasonable assurance that corrective actions will be adequate to manage the aging effects on the basemat.

(8) Confirmation Process

The applicant stated that procedural controls are in place to ensure that it reviews the responses to corrective action assignments entered into the SMUD Corrective Action Program and verifies the adequacy of the response. The applicant also stated that it reviews condition reports for trending purposes and will establish a tollgate to assess the effectiveness of corrective actions and update the AMP as necessary on a periodic basis.

The staff notes that NUREG-1927, Revision 1, states that NRC-approved QA programs are an accepted approach to ensuring that the effectiveness of corrective actions are verified and that adverse trends are monitored to address potential degradation before a loss of function. The staff also notes that performing periodic AMP effectiveness reviews is consistent with the guidance in NUREG-1927, Revision 1, which recommends that programs incorporate future reviews of operating experience to maintain effectiveness. Therefore, the staff finds the confirmation process program element to be acceptable.

(9) Administrative Controls

The applicant stated that the administrative controls are conducted in accordance with the requirements of a QA program under 10 CFR Part 72, Subpart G, and will continue for the period of extended operation. The applicant also stated that the Rancho Seco Quality Manual meets this requirement.

The staff notes that NUREG-1927, Revision 1, states that NRC-approved programs under 10 CFR Part 72, Subpart G, are an accepted approach to providing adequate review, approval, and fulfillment of activities that ensure SSCs will continue to perform satisfactorily in service. On this basis, the staff finds the administrative controls program element to be acceptable.

(10) Operating Experience

The applicant stated that Section 3.2 of the LRA evaluates operating experience for the basemat, which supports a conclusion that the Basemat AMP will be adequate to detect and manage aging effects on the basemat during the period of extended operation. The applicant referenced the groundwater evaluation documented in Section 2.4.6 of the Rancho Seco ISFSI FSAR, which indicates the presence of groundwater underlying the site approximately 150 feet below the original ground surface and indicates that the water table has receded over recent years and is expected to recede further due to the grape vineyards adjacent to the site. The applicant referenced the soil liquefaction evaluation documented in Section 3.4.2.2 of the Rancho Seco ISFSI FSAR, which states that soil liquefaction generally occurs in areas where groundwater is not deeper than 50 feet below grade. As a result, the applicant concluded that soil liquefaction at the SMUD ISFSI site is highly unlikely.

The staff reviewed the operating experience program element to ensure that the applicant appropriately considered past operating experience and that the program provides for future reviews of operating experience to confirm the program's continued effectiveness. The staff notes that the no degradation of the basemat was identified in the applicant's preapplication inspection. The staff also notes that the applicant's proposal to share operating experience with the industry and conduct periodic AMP effectiveness reviews is consistent with the guidance in NUREG-1927, Revision 1, which recommends that programs incorporate future reviews of operating experience to ensure continued effectiveness. Therefore, the staff finds the operating experience program element to be acceptable because the applicant provided sufficient prior operating experience to support the effectiveness of AMP activities and included a framework for future operating experience reviews to ensure that AMPs will be adjusted as knowledge becomes available from new analyses, experiments, and inspection activities.

The staff concludes that (1) the applicant adequately addressed the 10 program elements of an AMP described in NUREG-1927, Revision 1 and (2) there is reasonable assurance that the AMP is adequate for managing the aging mechanisms and effects of the in-scope SSCs identified by the AMR, such that the in-scope SSCs will continue to perform their intended functions during the requested period of extended operation.

3.5.5 Evaluation Findings

The staff reviewed the AMPs in the LRA following the guidance in NUREG-1927, Revision 1, and NUREG-2214. Based on its review, the staff finds the following:

F3.4 The applicant has identified programs that provide reasonable assurance that aging effects will be managed effectively during the period of extended operation, in accordance with 10 CFR 72.42(a)(2).

4 LICENSE CONDITIONS TO ADDRESS RENEWAL

This section includes a consolidated list of the changes to the license conditions resulting from the review of the LRA, some which have been described throughout the previous sections of this SER. The discussion presents the basis for the changes that are not described elsewhere in this SER.

4.1 Final Safety Analysis Report Update

The NRC added the following condition to the license:

> Within 90 days after issuance of the renewed license, SMUD shall submit an updated final safety analysis report (FSAR) to the U.S. Nuclear Regulatory Commission (NRC), in accordance with 10 CFR 72.4 and continue to update the FSAR pursuant to the requirements in 10 CFR 72.70(b) and (c). The updated FSAR shall reflect the information provided in Appendix C of the Rancho Seco ISFSI License Renewal Application, Revision 3, dated July 12, 2019 (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19204A248). The licensee may make changes to the updated FSAR, consistent with 10 CFR 72.48(c).

The applicant submitted the proposed changes to the FSAR in Appendix C to the Rancho Seco ISFSI LRA, Revision 3 (SMUD, 2019c), which reflects the final proposed FSAR changes to address the aging management activities described in the LRA. This license condition requires the applicant to submit an updated FSAR that includes the information from Appendix C to the LRA within 90 days after issuance of the license renewal. This condition ensures that the changes to the FSAR are made in a timely fashion to enable the licensee to develop and implement necessary procedures related to renewal and aging management activities during the period of extended operation.

4.2 Aging Management Program Implementation

The NRC added the following condition to the license:

> A program document(s) shall be revised, or a new one created, for implementing the activities in the aging management programs (AMPs) described in the updated FSAR within one year after the issuance of the renewed license. The program document(s) shall contain a reference to the specific AMP provision(s) that the program document(s) is intended to implement, and the reference shall be maintained even if the program document(s) is modified. The licensee shall maintain the program document(s) throughout the term of this license.

This license condition requires the applicant to revise or create a program for implementing the AMPs described in the FSAR supplement. This condition ensures that the program addresses AMP activities required for extended storage operations. The timeframe (1 year) in the condition is to ensure that the applicant develops the program document in a timely manner. This timeframe is consistent with the guidance in NUREG-1927, Revision 1.

4.3 Sacramento Municipal Utility District MP187 Cask

The NRC revised the TS 1.1 definition of "MP187 CASK" to read as follows:

> The MP187 cask is used for onsite transfer of a loaded DSC.

The NRC revised TS 4.2.1, footnote 1, to read as follows:

> The MP187 cask is used for onsite transfer of loaded DSCs.

The NRC revised the first sentence of the fourth paragraph of TS 4.2.1, to read as follows:

> The MP187 cask is used for onsite transfer of a loaded DSC.

The applicant proposed a license condition to clarify that an NRC approval under 10 CFR Part 71, "Packaging and Transportation of Radioactive Material," is needed for transporting the MP187. The staff reviewed the applicant's proposed condition and determined that it could inadvertently be interpreted to extend the SNM-2510 licensing basis (under 10 CFR Part 72) to transportation of the MP187, which is not addressed in 10 CFR Part 72, but could only be addressed through the requirements in 10 CFR Part 71. The staff determined that a revision to the existing TS to remove the reference to transportation of the MP187 would accomplish the same goal, and make it apparent that the SNM-2510 license does not address or authorize transportation of the MP187.

4.4 References to Rancho Seco Nuclear Generating Station

The NRC revised condition 2 and deleted condition 20 of the license.

The applicant proposed the revision of the name of the facility in license condition 2. The NRC changed the facility name from the "Rancho Seco Nuclear Generating Station" (RSNGS) to the "Rancho Seco Independent Spent Fuel Storage Installation." The NRC terminated DPR-54, the 10 CFR Part 50 license for RSNGS, on August 31, 2018. Therefore, an administrative change to this license condition to refer to the name of the ISFSI is appropriate.

The applicant proposed deleting license condition 20 that requires that fuel and cask movement and handling activities that are to be performed in the RSNGS fuel storage building be governed by the requirements of RSNGS Operating License DPR-54 and the associated TS. Because the NRC terminated the DPR-54 license, license condition 20 is no longer needed. Therefore, an administrative change to delete this license condition is appropriate.

The applicant also requested conforming revisions to references and titles in the TS (SMUD, 2020). The NRC staff determined that the following administrative changes to the TS are appropriate to reflect the termination of the DPR-54 license for RSNGS:

- TS 4.1—Added "former" before the reference to the site location "Rancho Seco Nuclear Generating Station (RSNGS) site."

- TS 5.1—Changed the title "The Manager, Plant Closure and Decommissioning (MPC&D)" to "The Manager, Rancho Seco Assets (MRSA)" in two places.

- TS 5.2—Changed the reference, “Defueled Safety Analysis Report (DSAR),” to “ISFSI Final Safety Analysis Report (IFSAR).” Deleted “SMUD will operate the Rancho Seco ISFSI under the same organization responsible for the Rancho Seco Nuclear Generating Station.”

- TS 5.5—Changed the text to reflect the termination of the 10 CFR Part 50 license, and to read as follows:

  > The managerial and administrative controls for the conduct of operations at the Rancho Seco ISFSI, built upon the RSNGS organization under the former 10 CFR 50 license, include the requirements of the 10 CFR 72 license.
  >
  > With the termination of the 10 CFR Part 50 license, appropriate 10 CFR 72.48 reviews will ensure continued compliance with the Rancho Seco ISFSI license requirements. SMUD will maintain the appropriate administrative and managerial controls at the Rancho Seco ISFSI until DOE takes title to the fuel.

- TS 5.5.1—Changed “SAR” to “Rancho Seco IFSAR.”

4.5 <u>Other Changes to the Technical Specifications</u>

The NRC revised the following technical specifications:

- TS 4.2.1—Changed the number of DSCs from 21 to 22. There are 22 DSCs stored at the Rancho Seco ISFSI. Twenty-one of the DSCs contain spent fuel; one DSC contains GTCC waste.

- TS 4.2.2—Added text to refer to the GTCC waste and the GTCC DSC stored at the Rancho Seco ISFSI, and to read as follows:

  > The Rancho Seco ISFSI can accommodate all of Rancho Seco's 493 spent fuel assemblies and greater-than-Class C radioactive waste. The ISFSI storage capacity consists of 18 FC-DSCs, 2 FO-DSCs, 1 FF-DSC, and 1 GTCC DSC.

- TS 5.2—Changed the title “General Manager (GM)” to “Chief Executive Officer & General Manager (CEO & GM).” Changed the title “senior site manager” to “Manager, Rancho Seco Assets.”

The applicant proposed these editorial changes to the TS to update organization and management titles and for clarity (SMUD, 2020). The NRC staff determined that the changes are administrative in nature and do not affect or change licensed operations.

5 CONCLUSION

Under 10 CFR 72.42(a), the Commission may issue a renewed license if it finds that actions have been identified and have been or will be taken such that there is reasonable assurance that the activities authorized by the renewed license will continue to be conducted in accordance with the design bases. In 10 CFR 72.42(a), the NRC requires the application for license renewal to include TLAAs and AMPs demonstrating that the SSCs important to safety will continue to perform their intended functions for the requested period of extended operation.

The NRC staff reviewed the LRA for the Rancho Seco ISFSI, in accordance with NRC regulations in 10 CFR Part 72. The staff followed the guidance in NUREG-1927, Revision 1; NUREG-2214; and NUREG-1757, Volume 3, Revision 1. Based on its review of the LRA and the license conditions, the staff determines that the requirements of 10 CFR 72.42(a) have been met.

6 REFERENCES

10 CFR Part 20, U.S. Code of Federal Regulations, “Standards for Protection Against Radiation,” Part 20, Chapter I, Title 10, “Energy.”

10 CFR Part 50, U.S. Code of Federal Regulations, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Chapter I, Title 10, “Energy.”

10 CFR Part 51, U.S. Code of Federal Regulations, “Environmental Protection Regulations for Domestic Licensing and Related Regulatory Functions,” Part 51, Chapter I, Title 10, “Energy.”

10 CFR Part 71, U.S. Code of Federal Regulations, “Packaging and Transportation of Radioactive Material,” Part 71, Chapter I, Title 10, “Energy.”

10 CFR Part 72, U.S. Code of Federal Regulations, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste,” Part 72, Chapter I, Title 10, “Energy.”

American Concrete Institute (ACI). ACI 224.4R-13, “Guide to Design Detailing to Mitigate Cracking.” Farmington Hills, Michigan: American Concrete Institute. 2013.

_____. ACI 349.3R-02, “Evaluation of Existing Nuclear Safety-Related Concrete Structures.” Farmington Hills, Michigan: American Concrete Institute. 2010.

_____. ACI 201.1R-08, “Guide for Conducting a Visual Inspection of Concrete in Service.” Farmington Hills, Michigan: American Concrete Institute. 2008a.

_____. ACI 209R-92, “Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures (Reapproved 2008).” Farmington Hills, Michigan: American Concrete Institute. 2008b.

_____. ACI 221.1R-98, “State-of-the-Art Report on Alkali-Aggregate Reactivity.” Farmington Hills, Michigan: American Concrete Institute. 2008c.

_____. ACI 318-05, “Building Code Requirements for Structural Concrete and Commentary.” Farmington Hills, Michigan: American Concrete Institute. 2005.

_____. ACI 215R-74, “Considerations for Design of Concrete Structures Subjected to Fatigue Loading.” Farmington Hills, Michigan: American Concrete Institute. 1997.

Alhasan, S.J. “Corrosion of Lead and Lead Alloys.” ASM Handbook, Vol. 13B. pp. 195–204. *Corrosion: Materials*. Materials Park, Ohio: American Society of Metals (ASM) International. 2005.

Alexander, D.J. and R.K. Nanstand. “The Effects of Aging 1 for 50,000 Hours at 343°C on the Mechanical Properties of Type 308 Stainless Steel Weldments.” Proceedings of the Seventh International Symposium on Environmental Degradation of Materials in Nuclear Power Systems–Water Reactors. pp. 747–758. Breckenridge, Colorado. Houston, Texas: NACE. 1995.

Anderson, B.L., M.K. Sheaffer, and L.E. Fischer. NUREG/CR-6673, "Hydrogen Generation in TRU Waste Transportation Packages." Washington, DC: U.S. Nuclear Regulatory Commission (NRC). Livermore, California: Lawrence Livermore National Laboratory. Agencywide Documents Access and Management System (ADAMS) Accession No. ML003723404. May 2000.

American Society of Mechanical Engineers (ASME). Boiler and Pressure Vessel Code, Section XI, "Rules for Inservice Inspection of Nuclear Power Plants and Fuel Processing Plants." New York, New York: American Society of Mechanical Engineers. 2013 Edition.

_____. Boiler and Pressure Vessel Code, Section III, Division 1, "Rules for Construction of Nuclear Facility Components." New York, New York: American Society of Mechanical Engineers. 1992 Edition, with Addenda through 1993.

American Society for Testing and Materials International (ASTM). ASTM C216, "Standard Specification for Facing Brick (Solid Masonry Units Made from Clay or Shale)." West Conshohocken, Pennsylvania: American Society for Testing and Materials. 2016.

Baggerly, R. "Environmental Failures of High Strength Bolts, in Case Histories on Integrity and Failures in Industry." V., ed. Proceedings of an International Symposium on Case Histories on Integrity and Failures in Industry, September 28–October 2, 1999. Milan, Italy. 1999.

Bare, W.C. and L.D. Torgerson. NUREG/CR-6745, "Dry Cask Storage Characterization Project-Phase 1: CASTOR V/21 Cask Opening and Examination." Washington, DC: U.S. Nuclear Regulatory Commission. Idaho Falls, Idaho: Idaho National Engineering and Environmental Laboratory. ADAMS Accession No. ML013020363. September 2001.

Bisco Products, Inc (BISCO). "NS-3 Specification Sheet." Park Ridge, Illinois: Bisco Products, Inc. ADAMS Accession No. ML110730731. June 23, 1986.

Bryan, C.R. and E.J. Schindelholz, "Analysis of Samples Collected from the Surface of Interim Storage Canisters at Calvert Cliffs in June 2017: Revision 01." SAND2017-12429xx. Albuquerque, New Mexico, and Livermore, California: Sandia National Laboratories. Printed November 2017.

Bryan, C.R. and D.G. Enos, "Analysis of Dust Samples Collected from Spent Nuclear Fuel Interim Storage Containers at Hope Creek, Delaware, and Diablo Canyon, California." SAND2014-16383. Albuquerque, New Mexico, and Livermore, California: Sandia National Laboratories. Printed July 2014.

Cadek, J. *Creep of Metallic Materials*. Materials Science Monographs, 48. New York, New York, and Amsterdam, Netherlands: Elsevier Science Publishing Company, Inc. 1988.

Caskey, G.R., R.S. Ondrejcin, P. Aldred, R.B. Davis, and S.A. Wilson. "Effects of Irradiation on Intergranular Stress Corrosion Cracking of Type 304 Stainless Steel." Proceedings of 45th NACE Annual Conference, April 23–27, 1990, Las Vegas, Nevada. 1990.

Chandra, K., K. Vivekanand, V.S. Raja, R. Tewari, and G.K. Dey. "Low Temperature Thermal Ageing Embrittlement of Austenitic Stainless Steel Welds and its Electrochemical Assessment." *Corrosion Science*. Vol. 54, pp. 278–290. 2012a.

Chandra, K., V. Kain, V. Bhutani, V.S. Raja, R. Tewari, G.K. Dey, and G.K. Chakravartty. "Low Temperature Thermal Aging Embrittlement of Austenitic Stainless Steel Welds: Kinetics and Effects on Mechanical Properties." *Materials Science and Engineering A.* Vol. 534, pp. 163–175. 2012b.

Cook, A.J.M.C., C. Padovani, and A.J. Davenport, "Effect of Nitrate and Sulfate on Atmospheric Corrosion of 304L and 316L Stainless Steels," *Journal of The Electrochemical Society*, Vol. 164 No. 4, pp. C148–C163. 2017.

Cragnolino G. and N. Sridhar, "A Review of Localized Corrosion of High-Level Nuclear Waste Container Materials-I," CNWRA 91-004, San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. ADAMS Accession No. ML003747940. April 1991.

Das, B.J. *Principles of Foundation Engineering*. 4th Edition. Pacific Grove, California: Brooks/Cole Publishing Company. 1999.

Davison, R.M., T. DeBold, and M.J. Johnson. "Corrosion of Stainless Steels." In ASM Handbook Vol. 13, pp. 547–565. *Corrosion*. Materials Park, Ohio: ASM International. 1987.

Department of Energy, U.S. (DOE). FCRD NFST 2015 000648, "Cost Implications of an Interim Storage Facility in Waste Management System." Revision 1. U.S. Department of Energy. 2016.

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

_____. "Susceptibility Assessment Criteria for Chloride-Induced Stress Corrosion Cracking (CISCC) of Welded Stainless Steel Canisters for Dry Cask Storage Systems," EPRI Report 3002005371. Palo Alto, California: Electric Power Research Institute. 2015.

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

_____. "Failure Modes and Effects Analysis (FMEA) of Welded Stainless Steel Canisters for Dry Cask Storage Systems." Palo Alto, California: Electric Power Research Institute. 2013.

_____. "Plant Support Engineering: Aging Effects for Structures and Structural Components (Structural Tools)." EPRI Report 1015078. Palo Alto, California: Electric Power Research Institute. 2007.

_____. "Climatic Corrosion Considerations for Independent Spent Fuel Storage Installations in Marine Environments." EPRI Report 1013524. Palo Alto, California: Electric Power Research Institute. 2006.

_____. "Dry Cask Storage Characterization Project, Final Report." TR-1002882, Palo Alto, California: Electric Power Research Institute. 2002.

Fontana, M.G. *Corrosion Engineering*. New York, New York: McGraw-Hill Book Company. 1986.

Furuya, T., H. Tomari, K. Fujiwara, S. Muraoka, and K. Araki. "Influence of Gamma Irradiation on Stress Corrosion Cracking of Austenitic Stainless Steels." Proceedings of the American Nuclear Society, International Topical Meeting on Fuel Reprocessing and Waste Management. August 26–29, 1984, Jackson Hole, Wyoming. pp. 493–502. La Grange Park, Illinois: American Nuclear Society. 1984.

Gamble, R. "BWRVIP-100-A: BWR Vessel and Internal Project, Updated Assessment of the Fracture Toughness of Irradiated Stainless Steel for BWR Core Shrouds." EPRI Report 1013396. Palo Alto, California: Electric Power Research Institute. 2006.

Gavendra, D.J., W.F. Michaud, T.M. Galvin, W.F. Burke, and O.K. Chopra. NUREG/CR-6428, "Effects of Thermal Aging on Fracture Toughness and Charpy-Impact Strength of Stainless Steel Pipe Welds." Washington, DC: U.S. Nuclear Regulatory Commission. May 1996.

Glass, R.S., G.E. Overturf, R.A. van Konynenburg, and R.D. McCright. "Gamma Radiation Effects on Corrosion—I. Electrochemical Mechanisms for the Aqueous Corrosion Processes of Austenitic Stainless Steels Relevant to Nuclear Waste Disposal in Tuff." *Corrosion Science*. Vol. 26, pp. 577–590. 1986.

Government Accountability Office, U.S. (GAO). GAO-10-48, "Nuclear Waste Management; Key Attributes, Challenges, and Costs for the Yucca Mountain Repository and Two Potential Alternatives." U.S. Government Accountability Office. 2009. www.gao.gov/products/GAO-10-48.

Government Accountability Office, U.S. (GAO). GAO-15-141, "Spent Nuclear Fuel Management; Outreach Needed to Help Gain Public Acceptance for Federal Activities That Address Liability." U.S. Government Accountability Office. 2014. www.gao.gov/products/GAO-15-141.

Grubb, J.F., T. DeBold, and J.D. Fritz. "Corrosion of Wrought 1 Stainless Steels." In ASM Handbook. Vol. 13B, pp. 54–77. *Corrosion: Materials*. Materials Park, Ohio: ASM International. 2005.

Hanson, B., H. Alsaed, C. Stockman, D. Enos, R. Meyer, and K. Sorenson. "Gap Analysis to Support Extended Storage of Used Nuclear Fuel, Rev. 0." FCRD-USED-2011-000136. PNNL-20509. Richland, Washington: Pacific Northwest National Laboratory. 2012.

He, X., T. Mintz, R. Pabalan, L. Miller, and G. Oberson. NUREG/CR-7170, "Assessment of Stress Corrosion Cracking Susceptibility for Austenitic Stainless Steels Exposed to Chloride and Non-chloride Atmospheric Salts." San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2014.

Jung, H., P. Shukla, T. Ahn, L. Tipton, K. Das, X. He, and D. Basu. "Extended Storage and Transportation: Evaluation of Drying Adequacy." San Antonio, Texas: Center for Nuclear Waste Regulatory Analyses. 2013.

Kain, R. "Marine Atmospheric Stress Corrosion Cracking of Austenitic Stainless Steel." *Materials Performance*. Vol. 29, No. 12, pp. 60–62. 1990.

Kim, S. and Y. Kim. "Estimation of Thermal Aging Embrittlement of LWR Primary Pressure Boundary Components." *Journal of the Korean Nuclear Society*. Vol. 30, No. 6, pp. 609–616. 1998.

King, F., P. Robinson, C. Watson, J. Burrow, R. Metcalfe, and S. Watson, "The Atmospheric Corrosion of Stainless Steel in Stores (ACSIS) Model," RWMD/03/003, Amec Foster Wheeler Project 17391, Quintessa Project 1558A. Amec Foster Wheeler Nuclear UK Limited. 2016.

Kodama, T. "Corrosion of Wrought Carbon Steels." In ASM Handbook, Vol. 13B, pp. 5–10. *Corrosion: Materials*. Materials Park, Ohio: ASM International. 2005.

Krauss, G., "Steels: Processing, Structure, and Performance." Materials Park, Ohio. ASM International. pp. 396–402. 2005.

Macfarlane, A., "Interim Storage of Spent Fuel in the United States," Massachusetts Institute of Technology, Cambridge, MA, Annual Reviews, Annual Review of Energy and the Environment, 26:201–35, 2001. http://web.mit.edu/stgs/pdfs/annurev.energy.pdf

Marsh, G.P., K.J. Taylor, G. Bryan, and S.E. Worthington. "The Influence of Radiation on the Corrosion of Stainless Steel." *Corrosion Science*. Vol. 26, pp. 971–982. 1986.

McCuen, R.H. and P. Albrecht. "Composite Modeling of Atmospheric Corrosion Penetration Data." STP 1194, Application of Accelerated Corrosion Testing to Service Life Prediction of Materials. West Conshohocken, Pennsylvania: ASTM International. 1994.

Meyer, R.M., A.F. Pardini, J.M. Cuta, H.E. Adkins, A.M. Casella, A. Qiao, A.A. Diaz, and S.R. Doctor. "NDE to Manage Atmospheric SCC in Canisters for Dry Storage of Spent Fuel: An Assessment." PNNL–22495. Richland, Washington: Pacific Northwest National Laboratory. 2013.

Morgan, J.D. "Report on Relative Corrosivity of Atmospheres at Various Distances from the Seacoast." NASA Report MTB 099-74. Cape Canaveral, Florida: National Aeronautics and Space Administration, Kennedy Space Center. 1980.

Morrison, J.D. "Corrosion Study of Bare and Coated Stainless Steel." NASA TN D-6519. Washington, DC: National Aeronautics and Space Administration. 1972.

National Association of Corrosion Engineers (NACE) International. *Corrosion Engineer's Reference Book*. Third Edition. R. Baboian, ed. Houston, Texas: NACE International. 2002.

Naval Facilities Engineering Command (NAVFAC). "Soil Mechanics." Design Manual NAVFAC DM-7.01. Alexandria, Virginia: Naval Facilities Engineering Command. 1986.

Neville, A.M. and W. Dilger. "Creep of Concrete: Plain, Reinforced and Prestressed." Amsterdam, Holland: North-Holland Publishing Co. 1970.

Nikolaev, Yu., A.V. Nikolaeva, and Ya. I. Shtrombakh. "Radiation Embrittlement of Low-Alloy Steels." *International Journal of Pressure Vessels and Piping*. Vol. 79. pp. 619–636. 2002.

Nuclear Regulatory Commission, U.S. (NRC). "Final Environmental Assessment for the Proposed Renewal of the U.S. Nuclear Regulatory Commission License Number SNM-2510 for the Rancho Seco Independent Spent Fuel Storage Installation in Sacramento County, California." Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML19241A378. 2020.

_____. NUREG-2214, "Managing Aging Processes In Storage (MAPS) Report." Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML19214A111. 2019.

_____. "Amendment No. 4 to Materials License No. SNM-2510 for the Rancho Seco Independent Spent Fuel Storage Installation." Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML17290A009. 2017a.

_____. "Safety Evaluation Report, Rancho Seco Independent Spent Fuel Storage Installation, Materials License No. SNM-2510, Amendment No. 4." Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML17290A010. 2017b.

_____. NUREG-1927, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel." Revision 1. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML16179A148. 2016.

_____. "Identification and Prioritization of the Technical Information Needs Affecting Potential Regulation of Extended Storage and Transportation of Spent Nuclear Fuel." Washington, DC: U.S. Nuclear Regulatory Commission. May 2014a.

_____. "Safety Evaluation Report for License Renewal: Calvert Cliffs Nuclear Power Plant Independent Spent Fuel Storage Installation." Washington, DC: U.S. Nuclear Regulatory Commission. 2014b.

_____. "Finite Element Analysis of Weld Residual Stresses in Austenitic Stainless Steel Dry Cask Storage System Canisters." NRC Technical Letter Report. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML13330A512. 2013.

_____. "Potential Chloride-Induced Stress Corrosion Cracking of Austenitic Stainless Steel and Maintenance of Dry Cask Storage System Canisters." NRC Information Notice 2012-20. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML12319A440. 2012b.

_____. NUREG-1757, "Consolidated Decommissioning Guidance: Financial Assurance, Recordkeeping, and Timeliness." Vol. 3, Revision 1. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML12048A683. 2012a.

_____. NUREG-1801, "Generic Aging Lessons Learned (GALL) Report." Revision 2. Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML103490041. 2010a.

_____. NUREG-1536, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility—Final Report." Revision 1. Washington, DC: U.S. Nuclear Regulatory Commission. 2010b.

_____. "Safety Evaluation Report for License Renewal: Surry Independent Spent Fuel Storage Installation." Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML050590266. 2005.

_____. Interim Staff Guidance-11, Rev. 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel." Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML033230335. 2003.

_____. NUREG-1522, "Assessment of Inservice Conditions of Safety-Related Nuclear Plant Structures." Washington, DC: U.S. Nuclear Regulatory Commission. ADAMS Accession No. ML062510407. 1995.

Odette, G.R. and G.E. Lucas. "Embrittlement of Nuclear Reactor Pressure Vessels." *Journal of Metals*. Vol. 53, Issue 7. pp. 18–22. 2001.

Plante. P. "NUTUG Pre-application Inspection of the NAC UMS and MPC Systems at Maine Yankee." On behalf of the NAC Utility Technology User's Group. Presented at the U.S. NRC Division of Spent Fuel Management Regulatory Conference. December 12, 2018.

Revie, R.W. *Uhlig's Corrosion Handbook*. Second Edition. Hoboken, New Jersey: John Wiley and Sons. 2000.

Sacramento Municipal Utility District (SMUD). "Request for Minor Editorial and Clarification Revisions for the Technical Review of the Application for Renewal of the Rancho Seco Independent Spent Fuel Storage Installation License No. SNM-2510." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML20038A131. January 23, 2020.

_____. "Response to Request for Clarification of Response to Additional Information for the Technical Review of the Application for Renewal of the Rancho Seco Independent Spent Fuel Storage Installation License No. SNM-2510." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML19204A239. July 12, 2019c.

_____. "Replacement Files for the Response to Request for Additional Information for Review of the Application for the Renewal of 10 CFR Part 72 Special Nuclear Materials License No. SNM-2510 for the Rancho Seco Independent Spent Fuel Storage Installation." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML19184A179. June 26, 2019b.

_____. "Response to Request for Additional Information for Review of the Application for the Renewal of 10 CFR Part 72 Special Nuclear Materials License No. SNM-2510 for the Rancho Seco Independent Spent Fuel Storage Installation." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML19121A269. April 22, 2019a.

_____. "Response to Request for Information For Review of the Application for the Renewal of 10 CFR Part 72 Special Nuclear Materials License No. SNM-2510 For The Rancho Seco Independent Spent Fuel Storage Installation." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML18285A414. September 26, 2018f.

_____. "Application for the Renewal of 10 CFR Part 72 Special Nuclear Materials License No. SNM-2510 for the Rancho Seco Independent Spent Fuel Storage Installation—Replacement Revision 1 Files [for June 25, 2018 submittal]." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML18221A293. August 6, 2018e.

_____. "Application for the Renewal of 10 CFR Part 72 Special Nuclear Materials License No. SNM-2510 for the Rancho Seco Independent Spent Fuel Storage Installation—Replacement Files [for March 19, 2018, submittal]." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML18221A283. August 6, 2018d.

_____. "Response to Request for Supplemental Information for Acceptance Review of the Application for the Renewal of 10 CFR Part 72 Special Nuclear Materials License No. SNM-2510 for the Rancho Seco Independent Spent Fuel Storage Installation." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML18179A255. June 25, 2018c.

_____. "Submittal of Application for the Renewal of 10 CFR Part 72 Special Nuclear Materials License No. SNM-2510 for the Rancho Seco Independent Spent Fuel Storage Installation." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML18101A026. March 19, 2018b.

_____. "Rancho Seco Report on Decommissioning Funding Status." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML18081A579. March 5, 2018a.

_____. "Rancho Seco Independent Spent Fuel Storage Installation Final Safety Analysis Report Revision 6." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession Nos. ML16236A115, ML16253A004, and ML16300A264. 2016b.

_____. "Notification of Mailing Date of the Rancho Seco Report on Decommissioning Funding Status." Sacramento, California: Sacramento Municipal Utility District. ADAMS Accession No. ML16106A109. April 7, 2016a.

Samuels, I.E. "Metals Engineering: A Technical Guide." Metals Park, Ohio: ASM International. 1988.

Sindelar, R.L., A.J. Duncan, M.E. Dupont, P.-S. Lam, M.R. Louthan, Jr., and T.E. Skidmore. NUREG/CR-7116, "Materials Aging Issues and Aging Management for Extended Storage and Transportation of Spent Nuclear Fuel." SRNL-STI-2011-00005. Aiken, South Carolina: Savannah River National Laboratory. Washington, DC: U.S. Nuclear Regulatory Commission. 2011.

Was, G.S., J. Busby, and P.L. Andresen. "Effect of Irradiation on Stress Corrosion Cracking and Corrosion in Light Water Reactors." In ASM Handbook, Vol. 13C, pp. 386–414. *Corrosion: Environments and Industries*. Materials Park, Ohio: ASM International. 2006.

Worthington, S.E., G.P. Marsh, and K.J. Taylor. "The Influence of Radiation on Crevice Corrosion." In *Improvements in Materials Reliability in the Back End of the Nuclear Fuel Cycle*. IAEA-TECDOC-686, pp. 55–64. Vienna, Austria: International Atomic Energy Agency. 1989.