UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

FINAL SAFETY EVALUATION REPORT
NAC INTERNATIONAL
NAC-UMS® STORAGE SYSTEM
DOCKET NO. 72-1015
AMENDMENT NO. 8

Summary

This safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of an amendment to Certificate of Compliance No. 1015 for the Model No. NAC-UMS® spent nuclear fuel (SNF) storage system. On December 18, 2019 (Agencywide Documents Access and Management System (ADAMS) Package Accession No. ML20006D749), as supplemented on April 24, 2020 (ADAMS Package Accession No. ML20122A201), and August 7, 2020 (ADAMS Package Accession No. ML20227A066), NAC International (NAC or the applicant) submitted a request to the NRC in accordance with Title 10 of the *Code of Federal Regulations* (10 CFR) 72.244 to amend Certificate of Compliance No. 1015.

NAC requested that storage of damaged SNF from boiling-water reactors (BWR) and a basket to hold the damaged BWR SNF be permitted by amendment of Certificate of Compliance No. 1015. NAC also requested the revision of the definitions for high burnup fuel and initial peak planar-average enrichment. The transportable storage canister (TSC), vertical concrete cask (VCC), and transfer cask remain unchanged.

In support of the amendment, NAC submitted Revisions 19A and 20C of the safety analysis report (SAR) for the NAC-UMS® storage system. The NRC staff reviewed the amendment request using guidance in NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities", dated April 2020. After completing its review of the statements and representations in the application and the conditions specified in the certificate of compliance and technical specifications, the NRC staff concludes that the requested changes meet the requirements in 10 CFR Part 72.

Chapter 1 GENERAL INFORMATION EVALUATION

This chapter evaluates the design changes made to the NAC-UMS® storage system to ensure they are described adequately and provide the NRC's reviewers and other interested parties with the pertinent features of the system, including the requested changes.

1.1 General Description and Operational Features

The NAC-UMS® system is a dry storage system whose principal components are a TSC, VCC, and transfer cask. The TSC, which contains the SNF fuel, is a stainless steel cylindrical shell stored within the central cavity of the VCC. The VCC provides radiation shielding and contains inlet and outlet vent and internal air flow paths allowing the decay heat to be removed by natural air circulation from around the outside of the TSC wall. A transfer cask is used to move the loaded TSC to and from the VCC and provides radiation shielding while the TSC is being closed

and sealed. The TSC is placed in the VCC by positioning the transfer cask on top of the VCC and lowering the TSC into place.

1.1.1 Storage Overpack – Vertical Concrete Cask

Five VCC designs accommodate five classes of TSCs of different lengths. The VCC is a reinforced concrete structure with a structural steel inner liner. The top of the VCC is closed by a shield plug and lid. The shield plug is approximately 5 inches thick and incorporates carbon steel plate as gamma radiation shielding, and either NS-4-FR or NS-3 borated polymers as neutron shielding. The VCC lid is constructed from 1.5-inch-thick carbon steel. The VCC shell is 28.3 inches thick and 136 inches in diameter. Its length varies depending on the length of the TSC.

1.1.2 Transportable Storage Canister

As with the VCC, there are five classes of TSCs of different lengths, each to accommodate a different class of pressurized-water reactor (PWR) or BWR fuel assembly. Each TSC has an outside diameter of about 67 inches and length of a TSC may vary from about 175 to 192 inches long. The maximum weight of a loaded PWR TSC is slightly less than 73,000 pounds (lbs.). The maximum weight of a loaded BWR TSC is slightly less than 76,000 lbs.

The TSC assembly consists of a right circular cylindrical shell with a welded bottom plate, a fuel basket, a shield lid, two penetration port covers, and a structural lid. The cylindrical shell, plus the bottom plate and welded lids, constitute the confinement boundary. The TSC consists of a cylindrical, 5/8-inch thick steel shell with a 1.75-inch thick stainless steel bottom plate and a Type 304 stainless steel shield lid support ring. A basket assembly is placed inside the canister. The shield lid assembly is a 7-inch thick stainless steel disk that is positioned on the shield lid support ring above the basket assembly.

The TSC assembly is designed to facilitate filling with water and subsequent draining and vacuum drying. Vent and drain ports through the shield lid allow the inner cavity to be drained, evacuated, and backfilled with helium to provide an inert atmosphere for long-term storage. After draining, drying, backfilling, and testing operations are completed, port covers are installed and welded to the shield lid to seal the penetration. The designs of the shield and structural lids provide a redundant confinement seal at the top of the canister.

The stainless steel BWR fuel basket is a right circular cylinder configuration with 56 stainless steel fuel tubes for BWR contents. The fuel tubes are laterally supported by a series of up to 41 5/8-inch-thick carbon steel support disks (depending on the length of the TSC), which are retained by spacers on 6 radially located tie rods. Aluminum heat transfer disks are spaced midway between the support disks and are the primary path for conducting heat from the SNF assemblies to the TSC wall.

1.1.3 Transfer Cask

Five transfer casks of different lengths are designed to handle the five classes of TSCs of different lengths. The transfer cask is a multi-wall (steel/lead/NS-4-FR/steel) design, each with about an 85-inch outer diameter. The five transfer casks range in height between about 177 and 194 inches, and in empty weight from about 112,000 to 121,000 lbs. The transfer cask has a bolted top retaining ring to prevent a loaded canister from being inadvertently removed

through the top of the transfer cask. Retractable (hydraulically operated) bottom shield doors on the transfer cask are used during unloading operations.

1.2 Drawings

In support of this application, NAC submitted the following 11 non-proprietary drawings and 1 proprietary drawing for NRC review:

Drawing No. 790-561, Revision No. 16 – Weldment, Structure, Vertical Concrete Cask (VCC), NAC-UMS®
Drawing No. 790-562, Revision No. 20 – Reinforcing Bar and Concrete Placement, Vertical Concrete Cask (VCC), NAC-UMS®
Drawing No. 790-570, Revision No. 6 – Fuel Basket Assembly, 56 Element BWR, NAC-UMS®
Drawing No. 790-575, Revision No. 11, – BWR Fuel Tube, NAC-UMS®
Drawing No. 790-582, Revision No. 13, – Shell Weldment, Canister, NAC-UMS®
Drawing No. 790-583, Revision No. 9, – Assembly, Drain Tube, Canister, NAC-UMS
Drawing No. 790-585, Revision No. 25, – Transportable Storage Canister (TSC), NAC-UMS®
Drawing No. 790-590, Revision No. 9, – Loaded Vertical Concrete Cask (VCC), NAC-UMS®
Drawing No. 790-601, Revision No. 2P, – Damaged Fuel Can (DFC), BWR, NAC-UMS®
Drawing No. 790-605, Revision No. 12, – BWR Fuel Tube, Over-Sized Fuel, NAC-UMS®
Drawing No. 790-671, Revision No. 1, – Bottom Weldment, Fuel Basket, 56 Element BWR DF [damaged fuel], NAC-UMS®
Drawing No. 70-672, Revision No. 1, – Top Weldment, Fuel Basket, 56 Element BWR DF, NAC-UMS®

1.3 Contents

The NAC-UMS® is designed to store up to 24 PWR or up to 56 BWR SNF assemblies. The PWR basket may contain up to four damaged fuel assemblies loaded in the corner position numbers 3, 6, 19, and 22 as shown in Figure B2-1 of the Appendix B to the Technical Specifications. Up to four damaged BWR fuel assemblies may be loaded into a Class 5 damaged BWR fuel basket (BWR5 DF). Based on the length of the fuel assemblies, PWR fuels are grouped into three classes (Classes 1 through 3), and BWR fuels are grouped into two classes (Classes 4 and 5). Class 1 and 2 PWR fuel assemblies include non-fuel-bearing inserts (components which include thimble plugs and burnable poison rods installed in the guide tubes). Classes 4 and 5 BWR assemblies include the zircaloy channels. The SNF is loaded into a TSC which contains a stainless steel gridwork, referred to as a basket. The applicant provided specifications for the higher enriched and higher burnup SNF in Technical Specifications, Appendix B for the new contents.

1.4 Evaluation Findings

Based on the NRC staff's review of information provided for the Amendment No. 8 to the NAC-UMS® system, the staff determined the following:

F1.1 A general description and discussion of Amendment No. 8 to NAC-UMS® system is presented in Chapter 1 of the SAR, with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations, and the description is sufficient and provide NRC's reviewers and stakeholders with the pertinent features of the system, including the requested changes.

F1.2 Drawings for structures, systems, and components (SSCs) that are important to safety (ITS) presented in Section 1.8 of the SAR were reviewed. Details of specific SSCs are evaluated in Chapters 3 through 17 of this SER.

F1.3 The specifications for the SNF to be stored in NAC-UMS® storage system provided in SAR Sections 1.3.1, 2.1.2, and Appendix B to the Technical Specifications are sufficient to provide NRC's reviewer and stakeholders with the details of the contents to be stored and provide fuel specifications in accordance with 10 CFR 72.236(a).

Chapter 2 SITE CHARACTERISTICS EVALUATION FOR DRY STORAGE FACILITIES

Site characteristics evaluation is not applicable to an application for a certificate of compliance.

Chapter 3 PRINCIPAL DESIGN CRITERIA EVALUATION

The applicant revised the principal design criteria by increasing the maximum weight of BWR canister by 200 pounds, providing details of the new contents to be stored in transferable transportation casks and fuel baskets, and including the reference to the BWR5 DF basket. These changes are evaluated in further detail, in Chapters 4 through 10 of this SER. In addition to these changes, NAC modified the quality category classification of several components, as show in Table 2.3-1, "Quality Category Classification of Universal Storage System Components."

3.1 Changes to Quality Category Classification

NAC proposed to change the quality category of the transfer cask coating system to non-quality (NQ). The coating system functions to protect the transfer cask during in-pool use and to provide a smooth surface to facilitate decontamination. The NAC-UMS® FSAR details the specific coating system by name and the quality category assigned.

Based on the review of the protective coating functions as described in the NAC-UMS® SAR and Section 8.5.12 of the NRC's NUREG 2215, the NRC staff finds the change to a NQ category is consistent with NUREG-2215 and is acceptable. The NAC-UMS® SAR does not credit the coating system for the transfer cask to perform an ITS function. This coating system does not need periodic coating inspections or maintenance to protect the transfer cask. Additionally, the failure of the coating system would not render the transfer cask nonfunctional or impact the general public including occupational workers. The NRC staff also noted that there are no known adverse chemical or galvanic reactions associated with the coating material.

In addition, NAC proposed to change the quality category of the canister shell and the components of the 56 element BWR DF basket to ITS category A and the canister drain tube assembly to ITS category C. The staff finds the change to the ITS categories A and C acceptable. The NRC staff reviewed the guidance in NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety" to evaluate the canister and basket assembly quality categories. The NRC staff finds the revised quality categories of these components are consistent with the guidance described in Section 6.2.1 of NUREG/CR-6407.

3.2 Findings

F3.1 The SSCs have been classified as ITS meet the requirements given in 10 CFR 72.236 for a certificate of compliance.

Chapter 4 STRUCTURAL EVALUATION

The applicant requested the NRC's approval to add damaged BWR fuel as authorized content for the NAC-UMS® cask system. Design features, supported by appropriate structural analyses, are introduced to the cask system to allow loading the DFCs at four corner basket locations. The amendment request also provides SAR evaluations to demonstrate that the cask system components, including the DFC and the high burnup BWR fuel, are structurally capable for all loading conditions associated with the cask handling and storage operations. The system's structural design bases and acceptance criteria for the initial certification of the NAC-UMS® cask system remain applicable. For this reason, the NRC staff's review focuses on verifying that the NAC-UMS® cask system, as modified for storing the damaged and high burnup BWR fuel, continues to be structurally capable for meeting the 10 CFR Part 72 requirements.

The NRC staff's structural review focused on three aspects: (1) the performance of the cask system components, including fuel basket upper and lower weldments under the applicable handling and operation loading conditions, (2) the structural capability of the DFC, and (3) the structural performance of the high burnup fuel for maintaining its analyzed configuration under the cask drop accident conditions. The NRC staff determined the previously approved design criteria and safety analyses for components of the NAC-UMS® cask system were unaffected by changes in this amendment. The staff finds the amended storage system meets the requirements of 10 CFR Part 72 and is consistent with the guidance in NUREG-2215.

4.1 Structural Evaluation

4.1.1 Structural Design Features

Damaged Fuel Can

The applicant provided design details in Drawing No 790-601 of the NAC-UMS® BWR DFC, which accommodates BWR damaged fuel. As noted in SAR Section 3.9.2, the DFC walls consist of 0.05-inch thick Type 304 stainless steel sheet (18 gauge) plate at a total length of 177.57 inches. The DFC weldment has a bottom plate that is 0.625 inches thick SA-240 Type XM-19 stainless steel. Four holes in the bottom plate, filter-screened with Type 304 stainless steel wire screens, permit water to be drained from the can during loading operations. At the top of the DFC, the top flange extends beyond each of the four walls to allow the use of a handling tool to lift the can and its content. The NRC staff finds that the applicant provided sufficient information to characterize SSCs that are ITS, including dimensions and materials, in the above drawing, consistent with the guidance in NUREG-2215.

Class 5 BWR DF Configuration, Fuel Basket Top and Bottom Weldments

The applicant described the design features for the BWR5 DF configuration in SAR Section 3.9, which include: (1) the BWR5 DF canister and (2) the BWR5 DF basket assembly that may contain up to four BWR DFCs. The applicant indicates that the BWR5 DF canister is identical to the BWR Class 5 (BWR5) canister for undamaged fuel with a slight increase (0.3 inches) of the nominal length. For the BWR5 DF basket, which is essentially identical to the BWR5 basket,

except for the locations for the DFC. Drawing Nos. 790-671 and 790-672 in the SAR depict the increased size of the four corner slot openings of the top and bottom weldments that allow for the placement of the DFCs. Correspondingly, SAR Drawing No. 790-605 provides design details for the over-sized, replacement fuel tubes at the four corner slots of the BWR5 basket for the DFCs. The NRC staff finds that the applicant provided sufficient information to characterize ITS components in the above drawings, including dimensions and materials of construction, consistent with the guidance in NUREG-2215.

4.1.2 Structural Evaluation of Boiling-Water Reactor Damaged Fuel Basket Configuration

In SAR Section 3.9, the applicant stated the VCC and the transfer cask are the same for the BWR5 DF canister and the BWR5 canister for undamaged fuel. As the only change with the loaded cask is with the canister, the applicant identified negligible differences in the weight and center of gravity of the loaded concrete cask for the BWR5 and BWR5 DF configurations. The NRC staff finds that SAR Section 3.9.1 appropriately considers the weight difference and the gravity or inertia loading for the structural components to evaluate the lifting and storage conditions for the BWR5 DF configuration of the NAC-UMS® system. The NRC staff concluded that Section 3.9.2 provides a structural evaluation of the BWR5 DFC that is consistent with the guidance in NUREG-2215.

4.1.2.1 Lifting Evaluation

In SAR Section 3.9.1.1, the applicant performed a lifting evaluation of the VCC loaded with BWR5 DF and analyzed two lifting situations for the VCC: 1) the bottom lift by hydraulic jack, and 2) the top lift using the cask's lifting lugs. The applicant also evaluated the loaded BWR5 DF canister and transfer cask for lifting conditions.

The applicant presented the safety bases and evaluations for the BWR Class 5 configurations in the final safety analysis report (FSAR) at Sections 3.4.3.1[1] for the VCC bottom/top lift, 3.4.3.2[2] for the canister lift, and 3.4.3.3[3] for the transfer cask lift. In Section 3.9 of the SAR, the applicant evaluated the lifting conditions of the BWR5 DF configurations by comparing them to the previously approved lifting evaluations of the BWR5 configurations. The applicant compared the weight difference and, in the case of the VCC bottom lift, the center of gravity location between the lift evaluations performed for the BWR Class 5 configurations and the BWR5 DF configurations. The applicant determined the weights and center of gravity location assessed in previous BWR5 lift evaluations are bounding, and therefore, no further evaluation is required. The staff reviewed the previous cask lift analyses and verified those design features bound the current cask lift analyses. The staff concluded that the maximum weights associated with the canisters and loaded casks of the BWR5 DF configuration are less than the maximum weights evaluated in the previous analyses and the center of gravity locations for the BWR5 DF canister are lower than those in previous analyses that adequately bound the new structural design features. Based on the NRC staff's review of the previous lifting analyses and the design

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[1] Section 3.4.3.1 covers Pages 3.4.3-5 through 3.4.3-27. Pages 3.4.3-5 through 3.4.3-22 were submitted in FSAR Update Revision 3 (ADAMS Accession No. ML041040369). Pages 3.4.3-23 through 3.4.3-26 were submitted in FSAR Update Revision 6 (ADAMS Accession No. ML082970749). Page 3.4.3-27 was submitted in FSAR Update Revision 3.

[2] Section 3.4.3.2 covers Pages 3.4.3-28 through 3.4.3-34. Pages 3.4.3-28 was submitted in FSAR Update Revision 4 (ADAMS Accession No. ML051290405). Pages 3.4.3-29 through 3.4.3-34 were submitted in FSAR Update Revision 3.

[3] Section 3.4.3.3 covers Pages 3.4.3-35 through 3.4.3-65. All pages were submitted in FSAR Update Revision 3.

features of BWR5 DF configuration, the staff finds that the previous evaluations are applicable and bounding for the normal, off-normal, and accident handling conditions. Therefore, the staff finds the applicant's assessment of the lifting operation is consistent with the guidance in NUREG-2215 and therefore is acceptable.

4.1.2.2 Canister

In SAR Section 3.9.1.2, the applicant noted that canister structural evaluations documented in FSAR Sections 3.4.4.1[4] and 11.1.3[5] use an upper-bounding canister weight, which is greater than the weight of the loaded BWR5 DF canister. Additionally, the maximum internal pressures of the BWR5 DF canister are bounded by the maximum internal pressures that are used in FSAR Sections 3.4.4.1 and 11.1.3 for all normal, off-normal, and accident conditions. The applicant stated that the canister weight and internal pressures used in previous FSAR analyses are bounding. The applicant concluded that no further evaluation is required because the bounding shows the absence of new conditions. The NRC staff verified the accuracy of the previous analyses and concludes the applicant's structural evaluation of the canister is applicable and bounding for the BWR5 DF configuration. The NRC staff finds the structural design features and the applicant's assessment of the canister structural evaluation are consistent with guidance in NUREG-2215 and are therefore acceptable.

4.1.2.3 Basket Assembly

In SAR Section 3.9.1.3, the applicant noted that the BWR fuel basket support disks and weldments are evaluated for normal, off-normal, and the 24-inch drop accident conditions in FSAR Sections 3.4.4.1, 11.1.3, and 11.2.4[6], respectively. The BWR fuel basket support disks are also evaluated for the tip-over condition as shown in FSAR Section 11.2.12[7]. The following

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[4] Section 3.4.4.1 covers pages 3.4.4-1 through 3.4.4-62. Page 3.4.4-1 was submitted in FSAR Update Revision 0 (ADAMS Accession No. ML010180095). Pages 3.4.4-2 through 3.4.4-7 were submitted with FSAR Update Revision 3. Pages 3.4.4-8 through 3.4.4-9 were submitted with FSAR Update Revision 12 (ADAMS Accession No. ML18319A102). Pages 3.4.4-10 was submitted with FSAR Update Revision 0. Pages 3.4.4-1 through 3.4.4-18 were submitted with FSAR Update Revision 3. Pages 3.4.4-19 was submitted with FSAR Update Revision 8 (ADAMS Accession No. ML090370421). Pages 3.4.4-20 was submitted with FSAR Update Revision 3. Pages 3.4.4-21 through 3.4.4-38 were submitted with FSAR Update Revision 0. Pages 3.4.4-39 through 3.4.4-48 were submitted with FSAR Update Revision 3. Pages 3.4.4-49 through 3.4.4-51 were submitted with FSAR Update Revision 0. Pages 3.4.4-52 through 3.4.4-62 were submitted with FSAR Update Revision 3.

[5] Section 11.1.3 covers Pages 11.1.3-1 through 11.1.3-16. Page 11.1.3-1 was submitted in FSAR Update Revision 3. Pages 11.1.3-2 through 11.1.3-3 were submitted in FSAR Update Revision 12. Pages 11.1.3-4 through 11.1.3-16 were submitted in FSAR Update Revision 3.

[6] Section 11.2.4 covers Pages 11.2.4-1 through 11.2.4-27. Pages 11.2.4-1 through 11.2.4-11 were submitted with FSAR Update Revision 3. Pages 11.2.4-12 was submitted with FSAR Update Revision 13 (ADAMS Accession No. ML19073A165). Pages 11.2.4-13 through 11.2.4-21 were submitted in FSAR Update Revision 0. Pages 11.2.4-22 through 11.2.4-24 were submitted in FSAR Update Revision 3. Pages 11.2.4-25 was submitted in FSAR Updated Revision 7 (ADAMS Accession No. ML112710131). Pages 11.2.4-26 through 11.2.4-27 were submitted in FSAR Update Revision 3.

[7] Section 11.2.12 covers Pages 11.2.12-1 through 11.2.12-71. Pages 11.2.12-1 was submitted with FSAR Update Revision 0. Page 11.2.12-2 was submitted with FSAR Update Revision 7. Pages 11.2.12-3 through 11.2.12-10 were submitted with FSAR Update Revision 3. Pages 11.2.12-11 through 11.2.12-12 were submitted with FSAR Update Revision 0. Pages 11.2.12-13 through 11.2.12-14 were submitted with FSAR Update Revision 3. Pages 11.2.12-15 through 11.2.12-17 were submitted with FSAR Update Revision 12. Pages 11.2.12-18 was submitted with FSAR Update Revision 0. Pages

describes the NRC staff's review of the performance of the fuel basket assembly resulting from the use of the DFC and the modified design feature of the top/bottom weldments.

Support Disk Evaluation

In SAR Section 3.9.1.3 the applicant stated that the support disks have an identical design for both the BWR5 and BWR5 DF configurations. The applicant noted the governing load condition for the support disks is the side impact. The applicant documented that a maximum weight (per disk) of 1,095 lb. was used in previous analyses, which exceeds the weight for the support disk for BWR5 DF basket (1,076 lb.). The applicant calculated the 30g side impact in a tip-over event produced a controlling load of 32.6 thousand pounds-force (kips) per support disk, for the previous configuration. The earlier calculation is greater than the load of 32.4 kips per support disk calculated for the BWR5 DF basket. The applicant determined that the previous support disk analyses for normal, off-normal, and accident conditions are bounding, and therefore no further evaluation is required. The NRC staff verified the previous analyses and concludes that the applicant's support disk assessment is consistent with guidance in NUREG-2215 and is therefore acceptable. The NRC staff concludes that the applicant's support disk assessment demonstrates that the fuel basket support disk will maintain its structural integrity under normal, off-normal, and accident conditions.

Basket Top Weldment Evaluation

The applicant stated in SAR Section 3.9.1.3 that the top weldment for the BWR DF basket configuration is based on the top weldment for BWR basket configuration with increased openings of the four corner slots. The applicant determined the governing load condition for the top weldment resulted from the 24-inch cask drop accident and the associated 60g end impact. So, for the BWR DF basket configuration, the applicant evaluated the support disk stress performance of the most critically stressed locations for a 60g end impact load, particularly the thinner ligaments adjacent to the increased slots. The applicant compared the ratio of the previous ligament width to the reduced ligament width to calculate the stresses in the reduced ligaments in the new basket design. The stresses for the previous ligaments where taken from the results of the finite element analysis described in FSAR Section 11.2.4. The applicant calculated the maximum sectional stress intensity of the reduced ligaments in the new basket design as 17.0 kips per square inch (ksi), which is less than the previous maximum stress intensity of 34.1 ksi listed in Table 11.2.4-3 and less than the allowable stress of 64 ksi in the design code, American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code, Section III, Subsection NG. The applicant concluded that no further analysis of the top weldment of the new basket design is required. The NRC staff reviewed the stress analysis of the top weldment of the previous basket design in FSAR Section 11.2.4, and the staff verified that it bounds the design of the top weldment for the BWR DF basket configuration. The NRC staff concludes that the applicant's assessment of the basket top weldment is consistent with guidance in NUREG-2215 and with the ASME B&PV Code, Section III, Subsection NG and is therefore acceptable to ensure its structural integrity under normal, off-normal, and accident conditions.

11.2.12-19 through 11.2.12-20 were submitted with FSAR Update Revision 3. Page 11.2.12-21 was submitted with FSAR Revision 0. Page 11.2.12-22 was submitted with FSAR Revision 3. Page 11.2.12-23 was submitted with FSAR Revision 0. Pages 11.2.12-24 through 11.2.12-70 were submitted with FSAR Revision 3. Page 11.2.12-71 was submitted with FSAR Revision 13.

Basket Bottom Weldment Evaluation

The NAC evaluation for the bottom weldment is similar to the top weldment described above. The bottom weldment for the BWR DF configuration is based on the bottom weldment for BWR configuration with increased openings of the four corner slots. The governing load condition for the bottom weldment resulted from the 24-inch cask drop accident and the applicant evaluated the most critically stressed locations of the associated 60g end impact load. As was done for the top weldment, the applicant compared the ratio of the previous ligament width to the reduced ligament width to calculate the stresses in the reduced ligaments in the new basket design. The stresses determined for the ligaments in the previous basket design were taken from the results of the finite element analysis results described in FSAR Section 11.2.4. The applicant calculated the maximum sectional stress intensity of the reduced ligaments as 39.8 ksi, which is less than the maximum stress intensity of 51.9 ksi listed in Table 11.2.4-3 and less than the allowable stress according to the design code, ASME B&PV Code, Section III, Subsection NG. The applicant concluded that no further analysis of the bottom weldment of the new basket design is required. The NRC staff reviewed the stress evaluation of the bottom weldment of the BWR DF configuration and verified that these weldment calculations are bounded by the previous analysis in FSAR Section 11.2.4. The NRC staff concludes that the applicant's assessment of the basket bottom weldment is consistent with guidance in NUREG-2215 and the ASME B&PV Code, Section III, Subsection NG and is therefore acceptable to ensure its structural integrity under the normal, off-normal, and accident conditions.

4.1.2.4 Vertical Concrete Cask

The applicant evaluated the concrete cask loaded with the BWR5 DF in SAR Section 3.9.1.4. The applicant compared the previously analyzed cask design features with those of the BWR5 DF configuration to assess the gravity and/or inertia load effects on the cask structural performance. The loads or conditions considered include thermal load, dead load, transfer cask load, snow/ice load, tornado and missile impact load, flood accident, and earthquake event. The applicant stated the previously analyzed loading conditions bound the new loading conditions, because the weight of the previously analyzed BWR configuration is greater than the weight of the BWR5 DF configuration and that the centers of gravity are at the same height in both configurations. For these reasons, the applicant concluded that no further analyses were needed to demonstrate the VCC structural capability. The NRC staff reviewed the previous analyses and the parameters of the new BWR5 DF configuration and finds the previous analyses are applicable and bounding, consistent with guidance in NUREG-2215. The NRC staff concludes the applicant's assessment of the VCC structural performance meets the requirements in 10 CFR 72.236 and is therefore acceptable.

4.1.2.5 Damaged Fuel Can

DFC Lifting/Handling Evaluation

The applicant presented the structural analysis of the DFC for normal off-normal, and accident conditions associated with lifting and handling operation in FSAR Section 3.9.2. In assessing the lifting tool engagement for the four sides of the DFC top flange, the applicant calculated large stress margins in key welds and structural components. The applicant calculated von Mises stress in the DFC top flange and the stresses in the full penetration welds joining the tube body to the flange during lifting operation. The applicant calculated that all factors of safety are greater than 6 and 10 against the material yield and ultimate strengths, respectively, consistent

with the guidance in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," and American National Standard (ANSI) N14.6, “Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 kg) or More.” The NRC staff reviewed this analysis and concluded that it is acceptable because the factors of safety are consistent with the guidance in NUREG-0612 and ANSI N14.6.

The applicant calculated the DFC tube body stress had a critical factor of safety greater than 20 based on the ASME Code, Section III, Subsection NG stress allowable at 600 degrees Fahrenheit (°F) for Type 304 stainless steel. Additionally, the applicant calculated the DFC tube bottom plate stress had a critical factor of safety greater than 3 based on the ASME B&PV Code, Section III, Subsection NG stress allowable for SA-240 Type XM-19 stainless steel at the underwater operating temperature of 200 °F. The NRC staff finds the large stress factors of safety for all key structural components demonstrate that the structural performance of the DFC is consistent with guidance in NUREG-2215 and the ASME B&PV Code, Section III, Subsection NG and is therefore acceptable for the normal, off-normal, and accident conditions associated with lifting and handling operation.

<u>BWR DFC Tube Body Side Impact</u>

In SAR Section 3.9.2.3, the applicant discussed its structural evaluation of the side impact condition for the BWR DFC configuration, which governs the cask tip-over accident. The applicant states that the BWR DFC tube body is primarily contained in the BWR DFC basket assembly, and therefore, the tube body will be in full longitudinal contact with the closely spaced support disks, during a potential side impact. The applicant determined the potential for significant bending stress in the DFC tube body would only occur in the 19.63-inch long unsupported portion of the DFC, which extends beyond the top-most support disk.

The applicant evaluated the DFC tube body as a cantilevered beam loaded with the combined weight of the overhanging tube body, top flange, and lid assembly, multiplied by a large dynamic amplification factor to account for the dynamic effect of the side impact. The applicant used the classical beam bending theory in a hand calculation to calculate the maximum bending stress and corresponding shear stress. The applicant reported a maximum stress intensity of 5,434 psi, a stress margin of 10.6 when compared to the allowable stress from ASME B&PV Code, Section III, Subsection NG. The NRC staff concludes the stress margin of 10.6 demonstrates adequate structural performance exists under the postulated cask tip-over accident. For these reasons, the NRC staff finds the structural performance meets the requirements in 10 CFR 72.236 and is therefore acceptable.

<u>BWR DFC Tube Body End Impact</u>

The applicant evaluated a bottom end impact for the BWR DFC configuration corresponding to the cask 24-inch drop accident in SAR Section 3.9.2.4. The applicant considered the combined mass of the lid assembly, top flange, and tube body and a deceleration of 60g, in calculating the compressive stress on the DFC tube body. The compressive load of 6,780 lbs. will result in a maximum compressive stress of 6,180 psi. The applicant evaluated this maximum compressive stress against the 600 °F at-temperature stress allowable of 44,300 psi and calculated the stress factor of safety of 6.2. The applicant calculated the DFC tube’s critical buckling load as 13,396 lbs. using the Euler buckling formula for columns. Because the maximum compressive load of 6,780 lbs. is much less than the critical buckling load of 13,396 lbs., the NRC staff finds that the applicant’s assessment that the DFC tube has sufficient resistance to buckling, is consistent with guidance in NUREG-2215 and the ASME B&PV Code, Section III,

Subsection NG, and is therefore acceptable. The staff finds the positive factors of safety for stress are adequate. Therefore, the predicted DFC structural performance of the BWR DFC tube body is acceptable for the 24-inch drop accident condition and meets the requirements in 10 CFR 72.236.

4.1.3 Boiling-Water Reactor Fuel Rod Drop Evaluation

The applicant presented the structural evaluation of the BWR high burnup fuel assemblies for the 24-inch end drop condition of the storage cask in SAR Section 11.2.16.2.

The applicant recognized that, in the end drop orientation, the fuel rods are laterally restrained by the grids and come into contact with the fuel assembly base. Similar to that used on previously approved methods of evaluation in other NAC FSARs, the fuel assembly is considered to have an initial bow of 0.157 inch as a boundary condition to initiate an LS-DYNA modeling transient response analysis. As depicted in FSAR Figure 11.2.16-5, other modeling parameters for the three, 8×8, 9×9, and 10×10 representative fuel assemblies include: (1) the maximum fuel rod lateral displacement of 0.937 inch, (2) shell and brick element use for the grid/fuel tube, respectively, (3) an initial downward, terminal velocity of 136 in/sec corresponding to the 24-inch drop, and (4) the acceleration time history, as obtained from the 24-inch VCC end drop analysis in FSAR Section 11.2.4.

The applicant reduced the thickness of the clad by 125 microns to account for the oxidation effect on the high burnup fuel. The applicant determined the maximum stress intensities with the margin of safety against yield strength of 2.53, 2.67, and 2.38 for the 8×8, 9×9, and 10×10 BWR fuel assemblies, respectively. These positive margins demonstrate that all clad stresses are well below the yield strength. Based on the positive margins, the NRC staff finds that high burnup BWR fuel will remain structurally adequate for the storage design basis cask end drop load condition and meets the requirements in 10 CFR 72.236.

4.1.4 Boiling-Water Reactor Fuel Rod Side Drop

In SAR Section 11.2.16.2, the applicant presented the structural evaluation of the BWR high burnup fuel assemblies for a 60g side drop condition, which bounds the cask tip-over accident condition.

The applicant used the ANSYS finite element analysis software to perform a quasi-static analysis evaluating the response in the side drop for the BWR fuel rod. Considering that the maximum deflection of a fuel rod is based on the fuel rod spacing of the fuel assembly, the applicant used the bounding 10×10 array of fuel assembly to obtain the maximum fuel rod deflection of 1.12 inch. Figure 11.2.16-6 shows the fuel rod beam element may have a maximum lateral displacement of 1.12 inches to account for the clad thickness reduction of 125 microns. The applicant relied on the ANSYS finite element analysis to demonstrate the maximum stresses on the 8×8, 9×9, and 10×10 BWR fuel assemblies have a margin of safety against yield strength of 1.68, 1.32, and 1.05, respectively. These positive margins demonstrate that all clad stresses fall below the yield strength for cladding from ASME B&PV Code, Section II. The positive margins, the staff finds that high burnup BWR fuel will remain structurally adequate for a 60g side drop condition, which bounds the tip-over accident condition. The NRC staff finds the structural performance of the BWR high burnup fuel assemblies is consistent with NUREG-2215 and meets the requirements in 10 CFR 72.236(l) and is therefore acceptable.

4.2 Evaluation Findings

F4.1 Spent nuclear fuel (SNF) handling, packaging, transfer, and storage systems are designed to ensure subcriticality. The structural margins of safety of these systems are adequate for the nature of the immediate environment under accident conditions, and therefore meet the requirements in 10 CFR 72.124(a).

F4.2 The SSCs that are ITS are designed to provide and maintain favorable geometry or permanently fixed neutron-absorbing materials, and therefore meet the requirements in 10 CFR 72.124(b).

F4.3 The design bases and design criteria are provided for SSCs important to safety that meet the requirements in 10 CFR 72.236(b).

F4.4 The applicant has met the requirements of 10 CFR Part 72.236(c), for the structural evaluation of items relied on for maintaining subcritical conditions. The structural design and fabrication of the NAC-UMS® storage system includes structural margins of safety for those SSCs important to nuclear criticality safety. The applicant has demonstrated adequate structural safety for the handling, packaging, transfer, and storage under normal, off-normal, and accident conditions.

F4.5 The structural evaluation of items relied on for meeting the radiation shielding and confinement requirements is sufficient to meet the requirements of 10 CFR 72.236.

F4.6 The SNF storage cask is designed to provide redundant sealing of confinement systems, and therefore meets the requirements in 10 CFR 72.236(e).

F4.7 The SNF storage cask is designed to store the SNF safely for the term proposed in the application, and therefore meets the requirements in 10 CFR 72.236(g).

F4.8 The SNF storage cask is compatible with wet or dry SNF loading and unloading facilities, and therefore meets the requirements in 10 CFR 72.236(h).

F4.9 The SNF storage cask and its systems important to safety have been evaluated by appropriate test or other acceptable means and have demonstrated that they will reasonably maintain confinement or radioactive material under normal, off-normal, and credible accident conditions, and therefore meet the requirements in 10 CFR 72.236(l).

F4.10 To the extent practicable, the SAR has considered the design of the SNF storage cask for compatibility with the removal of the stored SNF from a reactor site, transportation, and ultimate disposition by the Department of Energy, and therefore meets the requirements in 10 CFR 72.236(m).

The NRC staff concludes the structural properties of the NAC-UMS® cask system comply with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The NRC staff finds reasonable assurance the structural design of the NAC-UMS® cask system and bounding of the storage of BWR high burnup fuel and the damaged BWR fuel meet the structural performance requirements of 10 CFR Part 72 and provide adequate protection of the public health and safely.

Chapter 5 THERMAL EVALUATION

The NRC staff reviewed the proposed Amendment No. 8 changes to the NAC UMS® storage system to ensure that the applicant had performed an adequate thermal evaluation to demonstrate the system is acceptable. The changes evaluated are the use of a BWR5 DF basket loaded with four BWR DFCs to store damaged fuel and/or fuel debris in a Class 5 TSC.

The TSC can be loaded up to 56 BWR fuel assemblies with a total heat load up to 23 kilowatts (kW) and a maximum decay heat of 0.4107 kW per assembly. The fuel cladding temperature limits for the BWR fuel rods are 400 degrees Celsius (°C) (752 °F) for normal conditions of storage and transfer operations, and 570 °C (1058 °F) for off-normal and accident conditions of storage, consistent with NUREG-2215.

5.1 Material Properties and Component Specifications

The applicant specified the design characteristics and component specifications of the NAC-UMS® Universal Storage System in SAR Table 1.2-1[8], which the NRC staff reviewed and approved in previous applications. The proposed four BWR DFCs in this amendment are made of Type 304 stainless steel enclosing neutron absorber material.

The NRC staff reviewed the cask components and their corresponding materials provided in SAR Table 1.2-1 and finds this information is consistent with the details in the previous applications approved by the NRC. The NRC staff confirmed that the design characteristics, and component specifications provided in FSAR Table 1.2-1 are applicable to the TSC containing four BWR DFCs loaded with damaged fuel or fuel debris, and therefore, meets the requirements in 10 CFR 72.236(b).

5.2 Thermal Model

The applicant noted in SAR Sections 1.2.1 and 4.4, that a BWR5 DF fuel basket will be loaded with four BWR DFCs at the corner slots. The BWR5 DF fuel basket, similar in design to the BWR5 fuel basket that was evaluated in Amendment No. 3 (ADAMS Accession No. ML040830048), has slightly larger openings at the four corner slots in the fuel basket top and bottom weldments, which allow for the proposed storage of the four BWR DFCs. The NRC staff also previously approved the thermal evaluation for the design of the BWR fuel basket in Amendment Nos. 0 and 3 (ADAMS Accession Nos. ML003762577 and ML040830048, respectively).

The NRC staff reviewed the BWR5 DF fuel basket, canister, and the BWR DFC, as described in SAR Sections 1.2.1 and 4.4 and as shown in Drawing Nos. 790-582, 790-585, and 790-601. The NRC staff determined the three-dimensional BWR canister model, previously approved by the NRC, is acceptable with the minor changes to the evaluation of the proposed BWR DF configuration. The NRC staff finds, the slightly longer openings at the four corners slots of the BWR DF fuel basket and canister will not cause significant change in heat removal when compared to those of the BWR fuel basket and canister, consistent with the guidance in NUREG-2215.

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[8] Table 1.2-1 was submitted in FSAR Update Revision 6 (ADAMS Accession No. ML082970749).

5.3 Thermal Evaluations

Temperatures

In SAR Section 4.4, the applicant evaluated a 100% failure of the fuel rods and cladding in the DFCs under two debris compaction levels: debris compaction levels 1, 100% compaction of the fuel debris (SAR Figure 4.4.1.2-5) and debris compaction level 2, 50% compaction of the fuel debris. The applicant stated in SAR Section 4.4 that the decay heat for a single fuel assembly (0.411 kW) is concentrated in the debris region located at the lower part of the active fuel region and pointed out the remainder of the active fuel region would not result in an additional heat generation rate.

The applicant stated in SAR Section 4.4.1.5 that effective thermal conductivities for the design basis BWR fuel assembly were used in evaluating the fuel debris region, as described in NUREG-2215 and accepted by the NRC in license applications. The applicant pointed out that the effective thermal conductivities used for fuel debris region are conservative because the 100% compaction of failed rods creates fuel debris is expected to have higher density for better conduction. The increased conduction exposes a greater surface area to radiation than would occur in assembly storing an undamaged fuel. This analysis of the extreme debris compaction level is bounding. The applicant presented the maximum temperatures of the fuel cladding, damaged fuel, support disks, and heat transfer disks for the two debris compaction levels analyzed in SAR Table 4.4.3-2.

The NRC staff reviewed description of the three-dimensional BWR canister model in SAR Section 4.4.1.2 and finds it is consistent with the evaluation for proposed BWR5 DF configuration. The staff confirmed that modeling the decay heat in the fuel debris region located at the lower part of the active fuel region, rather than the bottom of the DFC, provides an acceptable thermal analysis. The active fuel region is closer to the center of the basket where the maximum fuel cladding temperature occurs and therefore represents real conditions with fuel debris. The NRC staff also confirmed that using the effective thermal conductivities derived from the design basis BWR for a 100% compaction of the fuel debris, is acceptable because the fuel debris has higher density and more surface area, and the BWR canister has more helium-filled space for heat removal. In addition, this approach for modeling fuel debris is consistent with the methodology approved by the NRC in the previous applications and is in compliance with 10 CFR 72.236(f).

The NRC staff reviewed SAR Table 4.4.3-2 and confirmed that for the analyzed 100% and 50% compaction debris compaction levels, both the peak cladding temperatures (PCTs) of 100% compaction and 50% compaction are below the acceptance criteria of 752 °F for normal conditions of storage. The NRC staff confirmed that the maximum temperatures of the fuel cladding, damaged fuel, support disks, and heat transfer disks are bounded by the corresponding temperatures for the design basis BWR fuel and are below the allowable temperature limits. Therefore, the NRC staff confirmed that use of the BWR DFC to contain damaged fuel or fuel debris is acceptable under normal conditions of storage and transfer operations and will meet the requirements in 10 CFR 72.236(f).

The NRC staff also confirmed the temperature bounding correlations calculated under normal storage conditions, for a TSC loaded with either undamaged design-basis BWR fuel or with BWR DFCs that contain damaged fuel or fuel debris, are safely extended to off-normal and accident conditions of storage. The NRC staff finds this acceptable because the heat load and

TSC/DFCs configuration remain unchanged under normal storage conditions, as well as under off-normal and accident conditions of storage, in compliance with 10 CFR 72.236(f).

Maximum Operating Pressure under Normal Conditions

The applicant assessed the BWR system's internal pressures during normal, off-normal and accident conditions calculated with the addition of the four BWR DF cans and BWR 10x10 fuel assembly design, as shown in NAC Calculation Package No. EA790-3312, "UMS® BWR System Pressure Evaluation for Damaged Fuel and Undamaged 10x10 Assemblies."

The applicant noted in SAR Table 4.4.5-4 that the proposed 10×10 fuel mass and fission gas inventory at a fixed burnup is less than that of 9×9 fuel assembly. The applicant noted in SAR Table 4.4.5-5 that the BWR5 DF canister has a slightly larger free volume. The applicant stated that using the BWR 9×9 fuel assembly and the BWR5 DF canister volumes provides bounding measures for the maximum operating pressure under normal conditions.

The NRC staff reviewed SAR Tables 4.4.5-4 and 4.4.5-5 for gas inventory and canister free volume and agrees it is acceptable to the use of the BWR 9×9 fuel assembly, and the BWR5 DF canister free volume as bounding for comparison of the maximum normal operating pressures (MNOPs) between the design-basis BWR canister and the BWR5 DF canister, consistent with NUREG-2215.

The applicant stated in SAR Section 4.4.5.2 that: (a) damaged fuel materials are bounded by the design-basis evaluations prepared for fuel assemblies containing clad failure, as these assemblies would release both fission and fill gas prior to placement into the storage system, and (b) vacuum drying would remove any gases in the failed clad. The applicant presented the maximum normal operating pressure of 3.96 pounds per square inch gauge (psig) for the BWR5 DF canister in SAR Table 4.4.5-6.

The NRC staff finds that the released fission and fill gas would be removed from damaged fuel or fuel debris prior to placement into the storage system when operational procedures are followed. The staff also independently calculated the maximum normal operating pressure for the BWR5 DF canister and confirmed it is below the pressure limit of 15 psig. The NRC previously approved these calculations in the initial license application.

The NRC staff's confirmatory analysis of the maximum operating pressure under the normal conditions of storage and its comparison of fission gas inventory, canister free volume, and released fission gas for the design-basis BWR canister and the BWR Class-5DF canister found reasonable assurance that the maximum normal operating pressure (3.96 psig) of the BWR5 DF canister, when loaded with four DFCs storing damaged fuel or fuel debris, will remain below the design limit of 15.0 psig for normal storage. The NRC staff reached the same conclusion in the initial license application.

The NRC staff also confirmed that the pressure bounding correlations of the design-basis BWR canister can be extended to the BWR5 DF canister, when loaded with four DFCs storing damaged fuel or fuel debris, under off-normal and accident conditions of storage and transfer operations. For these reasons, the NRC staff finds the maximum pressures for off-normal and accident conditions will remain below the design limit of 65 psig, as required by 10 CFR 72.236(e) and 72.236(f).

5.4 Evaluation Findings

F5.1 The NRC staff found reasonable assurance that use of the TSC containing four BWR DFCs to store damaged fuel or fuel debris is described in sufficient details in the application to enable an evaluation of the heat removal effectiveness. The SSCs that are ITS remain within their operating temperature ranges and will meet the requirements in 10 CFR 72.236(f).

F5.2 The NRC staff found reasonable assurance that use of the TSC containing four BWR DFCs to store damaged fuel or fuel debris with a heat load up to 23 kW continues to perform its heat-removal capability is consistent with its importance to safety and will meet the requirements in 10 CFR 72.236(f).

F5.3 The NRC staff found reasonable assurance that the BWR DF configuration is bounded by the design basis BWR configuration, and therefore the maximum fuel cladding temperatures are maintained below the limits specified in NUREG-2215, and the maximum cask component temperatures will continue to be maintained below the allowable limits for normal, off-normal, and accident conditions of storage and transfer operations and meets the requirements of 10 CFR 72.236(f).

F5.4 The NRC's staff found reasonable assurance that the TSC containing a BWR5 DF basket, loaded with four BWR DFCs for storage of damaged fuel or fuel debris under a design heat load of 23 kW is able to sustain the maximum canister pressures under normal, off-normal, and accident-level conditions and transfer operations and will meet the requirements in 10 CFR 72.236(e) and (f).

F5.5 The NRC staff concludes that the thermal design of the TSC, loaded with four BWR DFCs for storage of damaged fuel or fuel debris under a design heat load of 23 kW, complies with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied, as required by 10 CFR 72.236(f). The NRC staff finds the thermal design provides reasonable assurance that the TSC will allow safe storage of SNF during the license period, as required by 10 CFR 72.236(f). The NRC staff's review considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices and determined NAC-UMS® storage system provides adequate assurance that the applicant had performed an adequate thermal evaluation to demonstrate the system is acceptable.

Chapter 6 SHIELDING EVALUATION

The purpose of the shielding evaluation is to determine whether the changes proposed in Amendment No. 8 for the NAC-UMS® storage cask provide adequate protection against direct radiation and meets the requirements in 10 CFR 72.236(d). The regulatory requirements that ensure that the design has adequate shielding to protect licensee personnel and members of the public include 10 CFR 72.236(a) and (d). An overall assessment of the design's compliance with regulatory dose limits is evaluated in the Chapter 11, "Radiation Protection."

The Amendment No. 8 for NAC-UMS® includes expansion of the BWR Class 5 fuel inventory to include the following:

- Addition of 10×10 BWR fuel type at low burnup (undamaged fuel),
- High burnup 8×8, 9×9, and 10×10 undamaged fuel, and
- Low/High burnup 8×8, 9×9, and 10×10 damaged fuel.

6.1 Shielding Design Description

6.1.1 Design criteria

The major components of the TSC are the shell and bottom, basket assembly, shield lid, and structural lid. The canister, shield, and structural lids provide a confinement boundary during storage, shielding, and lifting capability for the basket. The applicant specified that a BWR5 DF basket is provided to load Class 5 BWR undamaged fuel assemblies and up to four BWR DFCs at corner slots. Class 5 BWR undamaged fuel are modified BWR assemblies. BWR Class 5 fuel is presented in Table 6.7.2-1 of the SAR. Table 6.7.2-1 of the SAR shows the modified parameters that are applied to the Class 5 Fuel that impact the shielding evaluation. The modifications of the fuel assemblies that are important to the shielding and criticality evaluations were evaluated, specifically the maximum number of fuel pellets, minimum cladding thickness, and maximum loading mass.

6.1.2 Design features

The Class 5 DF canister was used to develop the shielding modeling for this amendment. The material densities and compositions are described in Section 6.3.4 of the NAC-UMS® FSAR[9]. The TSC consists of a cylindrical, 5/8-inch thick Type 304L stainless steel shell with a 1.75-inch thick Type 304L stainless steel bottom plate and a Type 304 stainless steel shield lid support ring. A basket assembly is placed inside the canister. The shield lid assembly is a 7-inch thick Type 304 stainless steel disk that is positioned on the shield lid support ring above the basket assembly.

6.1.3 Maximum Dose Rates

The maximum dose rates at various locations on the surface and 1 meter from the transfer cask and storage cask are summarized in Table 5.8.2-4 through Table 5.8.2-7 of the SAR. The applicant analyzed the dose rates for 56 undamaged Class 5 BWR fuel assemblies in the 56-assembly BWR basket, which is designed to contain up to four DFCs in the basket corners. The analysis used the Monte Carlo N-Particle Transport Code (MCNP) response function method and source terms in Calculation Package No. EA790-4601, Revision No. 0, "TRITON-ARP BWR High Burnup Source Term and Minimum Cool Time Generation." Calculating the cool times for low burnup 10×10 fuel assemblies (Section 5.8.1 of the SAR) and for High Burnup 8×8, 9×9, and 10×10 fuel assemblies (Section 5.8.2 and 5.8.3 of the SAR), the applicant concluded the 8×8 fuel assembly produces bounding system dose rates, since it provides the highest source term.

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[9] Section 6.3.4 covers Pages 6.3-7 through 6.3-9. Page 6.3-7 was submitted in FSAR Update Revision 7 (ADAMS Accession No. ML112710131). Page 6.3-8 was submitted in FSAR Update Revision 4 (ADAMS Accession No. ML051290405). Page 6.3-9 was submitted in FSAR Update Revision 3 (ADAMS Accession No. ML051290403).

The applicant calculated the maximum dose rate of 805.25 millirem per hour (mrem/hr) on the radial surface of the transfer cask using a BWR 8×8 fuel assembly as the source with a burnup of 59 gigawatt days per metric ton (GWd/MTU), with an enrichment of 3.1 weight percent (wt%) U-235, and with a cooling time of 11.3 years (yrs). Calculating dose rates for the top of the transfer cask, the applicant reported the maximum dose rate of 454.99 mrem/hr using a source term consisting of the BWR 8×8 fuel assembly with a burnup of 59 GWd/MTU, with enrichment of 3.1 wt% U-235, and with a cooling time of 11.3 yrs. At the bottom level of the transfer cask, the maximum dose rate of 595.31 mrem/hr was calculated using BWR 8×8 fuel as a source with a burnup of 49 GWd/MTU, with enrichment of 2.7 wt% U-235, and with a cooling time of 7.0 yrs.

For the concrete storage cask at the radial surface, the maximum dose rate of 20.96 mrem/hr was calculated using a source term consisting of the BWR 8×8 fuel assembly with a burnup of 46 GWd/MTU, with enrichment of 2.7 wt% U-235, and with a cooling time of 6.1 yrs. At the top of the concrete storage cask, the maximum dose rate of 94.65 mrem/hr was calculated using a source term consisting of BWR 8×8 fuel assembly with a burnup of 59 GWd/MTU, with enrichment of 3.1 wt% U-235, and with a cooling time of 11.3 yrs. At the inlet of the concrete cask, the maximum dose rate of 234.01 mrem/hr was calculated using a source term consisting of BWR 8×8 fuel assembly with a burnup of 46 GWd/MTU, with enrichment of 2.7 wt% U-235, and with a cooling time of 6.1 yrs. At the outlet of the concrete cask, the maximum dose rate of 44.54 mrem/hr was calculated using a source term consisting of BWR 10×10 fuel assembly with a burnup of 38 GWd/MTU, with enrichment of 2.3 wt% U-235, and with a cooling time of 5.0 yrs. The fuel midplane height dose rates for 8x8, 9x9 and 10x10 BWR fuel are 16.2 mrem/hr, 15.3 mrem/hr, and 17.8 mrem/hr, respectively.

The NRC staff reviewed the MCNP input models used in the applicant's dose rate analyses of the transfer cask and the concrete storage cask. Based upon the information provided by the applicant, staff found reasonable assurance that the dose rates determined by NAC are representative of dose rates for a transfer cask containing a canister loaded with additional Class 5 BWR fuel and up to four DFCs. While the dose rates for a transfer cask and filled canister are higher than for the concrete cask, the loading of the transfer cask will be performed under appropriate radiation protections and with the as low as (is) reasonably achievable (ALARA) program in place, to adequately protect against higher dose rates, in accordance with 10 CFR 20.1003.

6.2 Radiation Source Definition

6.2.1 Low Burnup 10x10 Undamaged Fuel Source Terms

In Section 5.8.1 of the SAR, the applicant described the Transport Rigor Implemented with Time-dependent Operation for Neutronic (TRITON) depletion computer code inputs used to generate Automatic Rapid Processing (ARP) cross section libraries and ARP inputs, which were in turn executed with Oak Ridge Isotope Generation (ORIGEN) computer code. The applicant used the codes to establish bounds for the ranges of burnup, initial enrichment, and cooling times required to populate minimum cool timetables. The 238-group Evaluated Nuclear Data File (ENDF)/B-VII neutron library was used. Source terms were generated using the neutron and gamma group structures in Tables 5.2-28 and 5.2-29 of the SAR, respectively.

Dose rates were based on a uniformly loaded NAC-UMS® system with a maximum cask heat load of 23.0 kW, using the minimum cool times from Section 5.8.2 of the SAR in the calculations. The applicant used a cobalt impurity level of steel/inconel of 0.8 g steel/kg inconel

(800 ppm). Technical Specifications Table B2-10 shows the loading table for low burnup BWR 10×10 fuel assemblies.

The applicant reviewed the BWR fuel assemblies in the NAC-UMS® fuel database to determine a set of design-basis fuel assemblies that effectively bound the prevalent vendor types for each array size. Based on the prevalent designs, the applicant created assemblies that maximize uranium loading and activated fuel hardware without adding undue conservatism.

Based on the design-basis fuel assemblies, the applicant created TRITON inputs to develop ORIGEN-ARP inputs and cross section libraries. The ORIGEN-ARP inputs establish bounds for a wide range of burnup, initial enrichment, and cool times utilized in subsequent analyses. The applicant computed source terms using the ORIGEN-ARP module of the Standardized Computer Analyses for Licensing Evaluation (SCALE) 6.1 package.

The loading tables in Section B.2 of the Technical Specifications set forth the minimum cooling times allowed for each burnup and enrichment combination and for each fuel type. The bounding source terms for a BWR low burnup 10×10 fuel assembly has a burnup of 38 GWd/MTU, with enrichment of 2.3 wt% U-235, and with a cooling time of 5.0 yrs.

6.2.2 High Burnup Source Terms

Source term generation model descriptions of the fuel assemblies analyzed for this amendment are mainly based on parameters taken from the NAC-UMS® System FSAR, 72-1015 Rev. 11. Source terms were comprised of fuel neutrons, fuel gammas, and activated hardware gammas. The applicant modeled the activation of the light elements within the cladding using the composition in Table 4-2 of the SAR. The neutron and gamma energy group structures are shown in Table 4-4 and Table 4-5 of the of the Calculation Package No. EA790-4602, Revision No. 0, “UMS 56-Assembly BWR Shielding Analysis for High Burnup Undamaged Fuels,” respectively.

The design-basis fuel assemblies were based on the maximum uranium masses. The design of the BWR 8×8 fuel assembly model is based on the General Electric (GE) 8×8-2 assembly, which has a uranium loading of 0.1864 MTU. The design of the BWR 9×9 fuel assembly model is based on the GE 9×9-2 assembly, which has a uranium loading of 0.1979 MTU. NAC provides the design-basis BWR 10×10 fuel assembly parameters in Table 5.8.1-1 of the SAR, including the uranium loading of 0.1975 MTU.

The loading tables in proposed Technical Specifications, Appendix B to Certificate of Compliance Number 1015 identify the minimum allowed cooling times for each burnup and enrichment combination, for each fuel type. The bounding source terms for BWR 10×10 fuel contains a burnup of 59 GWd/MTU, enrichment of 3.1 wt% U-235, and cooling time of 11.3 years.

6.2.3 Burnup Profile

The applicant developed the burnup profile for BWR fuel, which is detailed in calculation package EA790-4006 Rev. 1, “BWR Fuel Assembly Burnup Profile.” The axial burnup profile was converted into a source profile based on the following relation between burnup and source rate:

$$S = B^b$$

where parameter b is determined based on fits to SAS2H computed radiation source rate plots for various fuel burnups. For neutron sources, parameter b is 4.22. For gamma radiation sources, the relation between burnup and source rate is linear and b is 1.0.

The staff found the reported burnup profiles are acceptable because the applicant followed the guidance in NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages" which describes the well-accepted relationship between source term and burnup. The burnup profile calculated for BWR fuel is presented in Table 6-5 of the Calculation Package EA790-4602. The applicant selected the peaking factor of 1.18 since it is the highest value in Table 6.5 to account for the higher burnup towards the axial midplane of the fuel assembly. The 1.18 peaking factor value in the burnup profile was increased slightly to 1.181 to increase the conservatism of the peak factor. The NRC staff finds using a peaking factor of 1.181 for the burnup profile to be acceptable based on the data shown in Table 6.5.

The axial source profiles are asymmetric with respect to the fuel axial midplane. To ensure that the correct total source is modeled in each ORIGEN-ARP model, a scale factor is computed which relates the actual source rate in each model to the total assembly source rate. This scaling is necessary in order to ensure that the dose rates are calculated correctly near the top or bottom of the cask. The NRC reviewed the methodology that NAC used to determine the scaling factors.

6.2.4 Activated Fuel Hardware

For the source from fuel hardware (e.g., top and bottom end fittings in the upper end fittings or the springs in the upper plenum region), the applicant applied an activation ratio to account for the distance offset of the axial region of the fuel assembly(upper and lower end fittings and upper plenum region) from the fuel assembly. In each axial region, the applicant determined the actual source strength by scaling the spectrum by an activation ratio that accounts for the neutron flux spectrum and intensity in that region. The activation ratios employed are given in Table 4-3 of the of the SAR. The activation ratios are as follows:

Region: Upper End Fitting - Flux Factor = 0.10;
Region: Upper Plenum - Flux Factor = 0.20;
Region: Fuel-Flux Factor = 1.00;
Region: Lower End Fitting – Flux Factor = 0.15.

The source terms in the fuel hardware regions include radiation from activated fuel assembly end fittings and plenum regions. The applicant specified that, to compute the total source for each hardware region, an effective mass was defined which represents the combination of the modeled mass and the activation flux factor. This effective mass can then be multiplied by the computed hardware spectra directly to determine a total source rate.

The applicant used the scaling factor from PNL-6906, "Spent Fuel Hardware Characterization and 10 CFR 61 Classification for Waste Disposal," which are values recommended in NUREG/CR-6802. The NRC staff found this approach to calculating source terms for fuel hardware acceptable because it is consistent with scaling factors cited in NUREG/CR-6802.

6.2.5 Confirmatory Analysis

The NRC staff performed confirmatory source term analyses using the TRITON input provided by the applicant and ORIGEN-ARP from SCALE6.1 code system. The NRC staff ran

confirmatory calculations for the burnup, enrichment, and cooling times and finds these parameters are bounding. The NRC staff's confirmatory analysis showed that the source term calculations provided in Calculation Package No. EA790-4601 are nearly identical to the staff analysis. The NRC staff's review of the information in the application and its confirmatory calculations provide reasonable assurance that the design basis gamma and neutron source terms are adequate for the shielding analysis.

6.3 Shielding Model Specification

6.3.1 Low Burnup Fuel

The BWR5 fuel assemblies were evaluated to low enrichment/low burnup combinations (down to 0.7 wt% U-235 with a maximum burnup of 5 GWd/MTU). The minimum enrichment limits exclude the loading of fuel assemblies enriched to less than 1.9 wt% U-235, for burnups greater than 25 GWd/MTU. NAC demonstrated in Section 5.8.2 of the SAR that the high burnup fuel bounds the low burnup fuel. The applicant computed dose rates using MCNP6.1 using a three-dimensional response function methodology. Bounding dose rates for undamaged and damaged fuel are reported in Sections 5.8.2 and 5.8.3 of the SAR, respectively. These bounding dose rates were calculated consistent with NUREG-2215 and meet the criteria in 10 CFR 72.104 and 72.106.

6.3.2 High Burnup Undamaged Fuels

Dose rates for high burnup undamaged BWR fuel intended for storage in the NAC-UMS storage system are calculated by the applicant based on a full payload of 56 assemblies. The applicant used the MCNP6.1 Monte Carlo radiation transport code to develop dose rate response functions. These response functions give the dose rate response as a function of energy for each source region. The applicant's analysis of the transfer cask was limited to the dry cask condition with both the shield lid and structural lid in place because this analysis provides a conservative dose rate since water in the TSC would add shielding for neutrons. This analysis applies during operations to load and prepare the canister for storage, transfer to the pad, and during storage. The dose rate analysis at the top of the transfer cask with shield lid showed significant gamma dose rates due to the top hardware components (e.g., plenum and end-fitting). At a fixed heat load, fuel hardware sources are maximized at low burnup/low cool time combinations and are therefore not impacted by the increase in burnup requested by the applicant. The NRC staff finds the applicant's model acceptable since it is consistent with the guidance in NUREG/CR-6802.

6.3.3 NAC-UMS® 56 Assembly Boiling-Water Reactor Damaged Fuel

The applicant used the MCNP6.1 Monte Carlo radiation transport code to calculate the dose rates for damaged fuel loaded in the cask. The source terms which resulted in the maximum surface dose rates for the undamaged storage and the transfer casks were adopted by the applicant to calculate the dose rates in this calculation. The cask shielding was the same as the model developed for the high burnup undamaged fuel calculations. The fuel rod end plugs were homogenized in the upper plenum and lower end fitting regions. The end plug volume is computed using the clad diameter, end cap length, and number of fuel pins. The applicant's MCNP models used in the shielding analysis were based on the drawings of the NAC-UMS® storage system. According to the applicant, the mass and volume of damaged fuel in the lower end fitting and active fuel regions was calculated based on a matrix of packing fractions from 30% to 75%. The NRC staff finds this approach acceptable because it is consistent with the

analysis presented in NUREG/CR-7203, “A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages.”

The applicant divided the active fuel region into two axial zones, with the lower region containing only damaged fuel and the upper region containing homogenized grid material. The applicant based the cut plane for the two regions on the assumed packing fraction and is the regional volume divided by the DFC inner width. The computed cut plane heights are summarized in Table 6-2 of the SAR. Based on the damaged fuel and hardware density values, the damaged fuel homogenizations are computed and summarized in Table 6-3 of the SAR. MCNP material definitions are summarized in Figure 6-1 of the SAR. Damaged BWR fuel assemblies may be loaded in DFCs in the four corner assembly locations of the BWR damaged fuel basket. The x and y coordinates for the DFC slots locations 6, 12, 45, and 51 are summarized in Table 6-4 of the Calculation Package EA790-4603, Revision No. 0 “UMS 56 Assembly BWR Damaged Fuel Shielding Analysis.”

The applicant made the following assumptions: damaged fuel is modeled with a packing fraction up to 75%, the active fuel region for the damaged fuel locations was modeled with only uranium dioxide (UO_2) for the lower region and only grid material for the upper region, the source terms which produce maximum dose rates are retained from the BWR undamaged fuel evaluation, the flange of the DFC lid is omitted in the MCNP model, and the thickness of DFC lid is modeled as 0.07 inches instead of 0.25 inches. The NRC staff finds this approach acceptable since reducing the thickness of the lid in the analysis provides conservatism because the thicker lid provides better shielding against radiation than the modeled lid.

The maximum dose rates for BWR fuel in the NAC-UMS® system are summarized in Table 1-2 and Table 1-3 of the SAR for transfer and storage casks, respectively. The NRC staff noted the tables indicate that the 52-assembly basket with 4 damaged fuel slots increases maximum dose rates for the transfer cask bottom and side. The applicant explained that this increase is due to fuel debris at the bottom of the DFC near the bottom forging and door where neutron shielding is minimal. There is also an increase in maximum dose rates on the concrete cask side. The applicant explained that this increase is due to azimuthal peaking from the 4 DFC locations.

6.3.3 Material Properties

Section 4.4 of the Calculation Package EA790-4602 describes the materials used in the shielding analysis. Definitions for zircalloy (zirc2), stainless steel (ss304), carbon steel (carbonsteel), concrete (reg-concrete), lead, aluminum, and water are based on the SCALE6.1 Standard Composition Library. The material compositions are shown in Table 4-6 of the SAR. In Section 5.8.2 of the SAR, the applicant described that the concrete in the VCC was evaluated at 145 lb/ft^3. The applicant modeled the fuel material as fresh UO_2 fuel. Applying a 95% factor to the UO_2 theoretical density of 10.96 g/cm^3 yields an analyzed value of 10.412 g/cm^3. Section 9.1.6.4.9 of the SAR discusses that the specified areal density of the neutron absorber material is 0.011 g ^{10}B/cm^2. The neutron absorber contains aluminum layers. The neutron absorber was modeled as a borated aluminum neutron absorber. The core thickness and the total plate thickness are 0.0885 inches and 0.135 inches, respectively. The density of the neutron absorber core is 2.6849 g/cm^3. The homogeneous material composition of the neutron absorber is listed in Table 4-8 of Calculation Package EA790-5601, Revision No. 0, “UMS BWR Undamaged Fuel Criticality Analysis for Selected Fuel Types and Increased Absorber Credit.”

The applicant described that all source regions were represented as homogenized. The homogenization employed in the analysis was performed over discrete axial sections of a single

fuel assembly. Homogenization of a source region involves accounting for the (modeled) material present and determining the volume fraction that each material occupies in the source region. An initial enrichment of 5.0 wt% U-235 was assumed in the MCNP material definition for fuel. A BWR 9×9 was used to present the fuel homogenization, hardware source description, and material composition. The NRC staff reviewed the composition and density of the materials used in the shielding evaluation and found that they are consistent with standard values from open literature or consistent with the design as specified in the drawings, and are, therefore acceptable.

6.4 Evaluation Findings

The NRC staff reviewed the applicant's shielding analysis for the addition of the proposed BWR fuel contents and finds it acceptable.

F6.1 The SAR provides specifications of the SNF contents to be stored in the NAC-UMS® in sufficient detail to allow evaluation of the dry storage system shielding design for the proposed contents. The SAR analyses are adequately bounding for the radiation source terms. The NRC staff finds the specifications of the SNF contents to be stored in the NAC-UMS® meet the requirements in 10 CFR 72.236(b) and (d).

F6.2 The SAR describes the structures, systems, and components (SSCs) important to safety that are relied on for shielding in sufficient detail to allow evaluation of their effectiveness for the proposed term of storage and meets the requirements in 10 CFR 72.236(b) and 10 CFR 72.236(g).

F6.3 The amendment request sufficiently describes the shielding methodology and design criteria for the added fuel contents. The radiation shielding analysis provides the NRC staff with reasonable assurance that the NAC-UMS® system will safely store SNF. The NRC staff reviewed the radiation shielding evaluations, including the calculations of the sources and the dose rates for the transfer cask and the concrete overpack. The NRC staff finds the applicant has correctly determined the bounding dose rates for all proposed BWR 8×8, 9×9, and 10×10 undamaged and damaged fuel, and found that it meets the design requirements in 10 CFR 72.236(d).

F6.4 The SAR provides reasonable and appropriate information and analyses, including dose rates, to allow evaluation of consideration of ALARA in the NAC-UMS® design and evaluation of occupational doses, consistent with the guidance in NUREG-2215. This evaluation is described in Chapter 10, "Radiation protection for Dry Storage Systems."

The NRC staff concludes that the NAC-UMS® system with the changes proposed in Amendment No. 8 has been designed and can be operated in compliance with the shielding requirements in 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied.

Chapter 7 CRITICALITY EVALUATION

The NRC staff reviewed the applicant's criticality analysis to confirm that all credible normal, off-normal, and accident conditions were identified and the potential consequences on criticality safety were considered and that the NAC UMS® system, as revised in Amendment No 8, continues to meet the following regulatory requirements in 10 CFR 72.124, 10 CFR 72.236(c), and(g). The applicant proposed adding BWR fuels consisting of low burnup 10×10 undamaged fuel, high burnup 8×8, 9×9, and 10×10 undamaged fuel, and high burnup 8×8, 9×9, and 10×10

damaged fuel as authorized contents in the NAC UMS® cask. The NRC staff evaluated the proposed new contents and the impact of the contents on the criticality safety evaluation provided by the applicant in the revised SAR pages submitted on December 18, 2019. This review considers the criticality safety requirements in 10 CFR Part 72 as well as the review guidance presented in NUREG-2215.

7.1 Criticality Design Criteria and Features

The NRC staff previously reviewed and accepted the design criteria and features of the NAC-UMS® cask in Amendment No. 7 (ADAMS Accession No. ML19183A268). Amendment 8 does not propose changes to the design criteria and features for the cask system currently approved by the NRC in Amendment No. 7. The NRC staff notes that the methods for evaluating criticality safety used in the current amendment request are consistent with those approved by Amendment 7.

7.2 Fuel Specification

The applicant reviewed the proposed modified and new fuel types proposed for the NAC UMS® system in this amendment request. These fuels include: ge08n, ge08k, ge08i, ex08b, ex09c, B9_91A, B10_91A, and B10_92A. The specific fuel parameters are identified in Table 6.7.2-1 of the amended SAR. Fuel conditions include both undamaged and damaged fuel contained in DFCs. The NRC staff reviewed the fuel parameters for the fuels finds all parameters important to criticality safety were adequately addressed and are consistent with the guidance in NUREG-2215 and therefore, meets the requirements in 10 CFR 72.236(a).

7.3 Model Specification

The current amendment request used the previously approved NAC-UMS® cask design components that are unchanged in this amendment. The undamaged BWR basket assembly, fuel canister, and transfer and storage cask dimensions are identical to those found acceptable for the storage of other allowable fuel contents. The NAC-UMS® BWR basket design (BWR5 and BWR5 DF) is composed of carbon steel and aluminum heat transfer discs that contain 56 fuel tubes. Each tube can contain one BWR fuel assembly. The applicant explained each NAC-UMS® cask will continue to be neutronically isolated from the cask exterior conditions and from other casks in an array because the extensive radiation shielding of casks during storage. The NRC staff finds the applicant's fuel composition acceptable because the isotopes present in the fuel are considered as unburned fuel, and the actual fissile content of the SNF is lower than what is modeled. The applicant performed a borated neutron absorber analysis applying a credit of 75% and 90% for boron-10 (B-10). The NRC staff finds the applicant's neutron absorber composition acceptable for this amendment because no changes were made to these materials in current amendment. The applicant used the standard SCALE 6 composition library for all other materials. The NRC staff concludes fuel types and material composition used in the criticality analysis for this amendment is consistent with guidance in NUREG-2215.

7.4 Criticality Analysis

The applicant performed a parametric criticality study for the proposed undamaged and damaged fuel contents. The applicant determined that the most reactive assembly among the proposed new fuel configurations was the B10_92A. The NRC staff performed confirmatory calculations using the bounding parameters identified by the applicant and finds that the B10_92A fuel assembly (see Table B2-3, "BWR Fuel Assembly Characteristics", in the

Technical Specifications, Appendix B) with vanished partial length fuel rods was bounding for the other fuel types. The applicant used this bounding configuration in its criticality safety analysis. The NRC staff finds this bounding analysis acceptable because the fuel utilized provide the most reactive configuration for the calculations and is therefore bounding for all of the proposed new fuels. Using a bounding fuel for the criticality analysis is consistent with the guidance in NUREG-2215.

7.4.1 Undamaged Fuel

The applicant evaluated the optimum moderator density configurations and demonstrated that the most reactive condition for the moderator remains a fully-flooded, loaded TSC. The applicant calculated the k_{eff} for the transfer cask under normal conditions (i.e., wet or dry canister with a wet or dry cask exterior) and accident conditions (i.e., full interior, exterior, and fuel clad gap water intrusion) as shown in Table 6.7.4-2 of the SAR. The applicant also evaluated the VCC by performing calculations under normal conditions (i.e., dry basket, dry transfer annulus, and dry exterior), off-normal conditions (i.e., dry basket, with the transfer annulus and cask exterior flooded), and accident conditions (i.e., full interior, exterior, and fuel clad gap water intrusion) as shown in Table 6.7.4-3 of the SAR. The applicant's calculations demonstrate the most reactive condition was the NAC-UMS®, under accident conditions. The applicant also evaluated partial flooding and found no increase in reactivity under that condition. The applicant evaluated the maximum enrichments of the new BWR fuel assemblies between 4.0 wt.% U-235 and 5.0 wt.% U-235 for each of the most reactive configurations. The applicant evaluated accident conditions using both 75% and 90% boron credit, and in all cases the applicant demonstrated that the calculated $k_{eff}+2\sigma$ was less than the upper safety limit (USL) of 0.9427 as shown in Table 6.7.4-5 of the SAR. The NRC staff found that the applicant used conservative assumptions in all potential flooding and enrichment configurations and are consistent with NUREG-2215.

The applicant also performed an evaluation of preferential loading of higher enrichments in specific locations as shown in Figure 6.7.5-1 of the SAR. The applicant evaluated several enrichment variations of higher enriched assemblies in the three outer location configurations and developed two maximum initial enrichment options as specified in Table 6.7.5-2 of the SAR. Enrichment up to 5.0 wt% U-235 is allowed in these locations provided the remaining locations in the NAC-UMS® are at a reduced maximum enrichment (i.e., 4.4 wt% U-235). The NRC staff reviewed the applicant's calculations for the preferential loading of higher enrichment fuels in specific locations and found that the k_{eff} for these loading patterns continue to be lower than the USL for all configurations and are consistent with NUREG-2215.

7.4.2 Damaged Fuel

The applicant evaluated the subcriticality of damaged fuel contained in DFCs. The applicant considered various configurations, including clad and unclad rods, pitch variations, rod arrays, missing rods, and homogenized fuel mixtures. The applicant also evaluated preferential flooding of DFCs and examined the effects of moderator density variations using the B10_92A fuel assembly at 4.4 wt% U-235. The NRC staff finds the evaluation of the subcriticality of damaged fuel contained in DFCs to be acceptable because the B10_92A has the maximum mass of uranium for all fuels evaluated and therefore provides the maximum reactivity. The applicant assumed a conservative boron credit contained in the neutron absorber sheets. The NRC staff finds this assumption acceptable because the actual amount of boron is reduced in the calculations, providing a more conservative measure than the minimum neutron absorber required in the Technical Specifications, Appendix B, Section B 3.2.1. The applicant determined

maximum reactivity to be for the configuration containing full density moderator within the DFC and full density moderator in the TSC cavity. The NRC staff reviewed the applicant's analysis and found that the parametric studies performed were conservative and identified the maximum k_{eff} for the bounding fuel was below the USL, consistent with NUREG-2215. Loading limits for all of the fuel types specified in the technical specifications and are identified in Table 6.7.6-6 of the SAR based on the boron credit taken which meets the requirements in 10 CFR 72.236(a).

7.5 Criticality Evaluation Findings

F7.1 The applicant has described the SSCs important to criticality safety in sufficient detail in Chapters 1 of the SAR to enable an evaluation of their effectiveness in accordance with 10 CFR 72.236(b).

F7.2 The proposed changes to the NAC-UMS®, including SSCs involved in the handling, packaging, transfer, and storage of the SNF to be stored, are adequate to ensure that the system will remain subcritical. Additionally, the applicant has demonstrated that nuclear criticality accident would not be possible, unless at least two unlikely, independent, and concurrent or sequential changes in the conditions essential to nuclear criticality safety occurred, as required by 10 CFR 72.124(a). The applicant's analyses in the SAR and confirmatory analysis performed by the NRC staff show that acceptable margins of safety will be maintained in the nuclear criticality parameters. In addition, the methods used in calculations demonstrate safety for the handling, packaging, transfer, and storage conditions and in the immediate environment under accident conditions. Therefore, the NRC staff concludes the proposed changes to the NAC-UMS® SSCs, which concern criticality safety, are in compliance with 10 CFR 72.124(a) and 10 CFR 72.236(c).

F7.3 The proposed conditions in the certificate of compliance and the technical specifications provide protections necessary to ensure nuclear criticality safety in the design, fabrication, construction, and operation of the NAC-UMS® SSCs, consistent with what is considered necessary to ensure compliance with 10 CFR 72.236(c) and (g).

F7.4 The NRC staff finds the SAR provides detailed specifications of the SNF contents to be stored in the NAC-UMS® which adequately define the authorized contents of the cask system. The NRC staff also finds the NAC-UMS® nuclear criticality safety design for the proposed SNF contents included. The analyses of the specifications of the proposed contents of the casks are adequately bounded. For these reasons, the NRC staff concludes that the SAR provided the information required by 10 CFR 72.236(a).

The NRC staff concludes that the criticality design features for the NAC-UMS® are in compliance with 10 CFR Part 72 and provide reasonable assurance the new fuel types will be safely stored. The staff finds reasonable assurance that the criticality design demonstrates that the NAC-UMS® will continue to allow for the safe storage of SNF. The staff also finds the applicable design and acceptance criteria have been satisfied. The NRC staff considered the applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices in making these findings.

Chapter 8 MATERIALS EVALUATION

8.1 General

The proposed amendment No. 8 presents information supporting changes for the storage of damaged BWR fuel in a DFC, a change to the allowable fuel burnup range, and revised definitions for the fuel contents.

8.2 Areas of Review

Codes and Standards

The applicant did not revise the materials codes and standards associated with the design of the NAC-UMS® storage system in the proposed Amendment No. 8. The new designs in Drawing Nos. 790-601, 790-671 and 790-672 are consistent with ASME B&PV Code, Sections II, III, V and IX. Amendment No. 8 adopts the same code criteria used in the previously approved amendments. For these reasons, the NRC staff finds the codes and standards used to be acceptable.

Drawings

The applicant identified changes to relevant parts of the drawings in the amendment request. The NRC staff reviewed the proposed changes to drawings and confirmed that these changes are adequately described and meet the regulatory requirements in 10 CFR 72.236(b) to provide design criteria, which include the specifications for code material and welding symbols that provide the weld details with code criteria for weld inspections. Therefore, the NRC staff finds the drawings to be acceptable.

Fuel, Cladding

The applicant defined the maximum temperature limits of the fuel in SAR Table 1.2-2. The NRC staff finds these thermal limits are consistent with the guidance in NUREG-2215 and they are unchanged from the previously approved amendments. Therefore, the NRC staff finds that the materials used in the spent fuel assemblies remain acceptable for performing their intended functions.

The applicant for the NAC-UMS® storage system states that BWR fuel assemblies can be unchanneled or channeled with zirconium alloy channels. The SAR states that BWR fuel assemblies containing stainless steel channels shall not be loaded with SNF. The NRC staff reviewed the new BWR channeled fuel designs and finds that the materials for these fuel assemblies to be consistent with guidance in NUREG-2215.

The applicant revised the definitions in the Technical Specifications associated with the fuel contents, including those associated with allowing BWR damaged fuel and fuel debris in a damaged fuel can. The NRC staff reviewed the fuel-related definitions and verified that they clearly describe the allowable fuel contents and t condition of the fuel. The staff finds the characteristics of loaded fuel are adequately addressed as required by regulation. For these reasons, the NRC staff finds that the definitions of the fuel-specific or other system-related functions of the NAC-UMS® storage system provide reasonable assurance that allowed SNF may be stored safely.

Concrete

Drawing No. 790-562, Sheet 1 of 8, stated that the concrete shall develop a compressive strength of 4000 psi. The NRC staff previously concluded in earlier amendments that this compressive strength, based on American Concrete Institute 349, “Code Requirements for Nuclear Safety-Related Concrete Structures and Commentary,” is acceptable.

8.3 Evaluations and Findings

F8.1 The NRC staff finds applicant’s materials design criteria for SSCs that are important to safety are consistent with the guidance in NUREG-2215 and are sufficiently detailed in its materials evaluation to meet the requirements in 10 CFR 72.236(b).

F8.2 The NRC staff finds applicant meets the requirements in 10 CFR 72.236(g) by demonstrating that the materials in and the design of the NAC-UMS® storage system supports the safe storage of SNF for the duration of the term in the application and permit maintenance as required. For these reasons, the materials and design meet the requirements in 10 CFR 72.236(b).

F8.3 The NRC staff finds applicant followed the guidance in NUREG-2215 and demonstrated the SNF storage container materials are compatible with their operating environment and fuel loading conditions without adverse degradation or significant chemical or other reactions, as required by 10 CFR 72.236(h).

Based on a review of the statements and representations in the application for Amendment No. 8, the NRC staff concludes that the applicant adequately described and evaluated the materials performance for this application, and it is acceptable. The staff concludes that the NAC-UMS, Amendment No. 8, materials performance is in compliance with 10 CFR Part 72.

Chapter 9 CONFINEMENT EVALUATION

In NAC-UMS® Amendment No. 8, the changes proposed in the SAR did not change the confinement design/function of the TSC, which is tested to leaktight criteria, in accordance with ANSI N14.5, “American National Standard for Leakage Tests on Packages for Shipment of Radioactive Materials.”

Since the design/function of the confinement system of the TSC did not change, the NRC staff finds that its conclusions on the safety and regulatory compliance NAC-UMS® storage system, which were approved by the staff in prior amendments, remain unchanged. Therefore, the NRC staff concludes the TSC remains in compliance with 10 CFR 72.236(d) and that the applicable design and acceptance criteria specified by NAC continue to be satisfied.

Chapter 10 RADIATION PROTECTION EVALUATION FOR DRY STORAGE SYSTEMS

The NRC staff’s review evaluates effects of additional BWR contents on the NAC-UMS® storage system to determine whether operations will be ALARA and whether the dose requirements in 10 CFR 72.236(d) can be achieved. The staff determined that the performance of the radiation protection capabilities of the NAC-UMS® system containing the additional high and low burnup fuel provides radiation shielding and confinement that adequately controls direct radiation, as required by 10 CFR 72.236(d), regardless of whether the fuel is damaged or undamaged. The staff also determined that the proposed changes in this amendment will maintain workers’

exposure ALARA and that radiation doses to workers and to the general public meet regulatory standards during both normal operation and accident situations, as required by 10 CFR Part 20. The regulatory requirements for providing adequate radiation protection to site licensee personnel and members of the public include 10 CFR Part 20 and 10 CFR 72.236(d).

10.1 Radiation Protection Design Criteria and Design Features

10.1.1 Design Criteria

The NRC staff finds that SAR Section 10.2 adequately describes the radiological protection design criteria of the NAC-UMS® and that the criteria meet the limits and requirements in 10 CFR Part 20, 10 CFR 72.104, 10 CFR 72.106. Additionally, the staff finds the SAR Section 10.2 is consistent with the guidance in Regulatory Guide 8.8, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be as Low as Is Reasonably Achievable."

The radiation protection design basis for the NAC-UMS® VCC is meets 10 CFR Part 72 and follows the applicable ALARA guidelines. The design-basis surface dose rates reviewed and approved by NRC on Page 10.2-1 in Revision 3 of the NAC-UMS® FSAR[10] and the calculated surface and 1-foot dose rates are:

Vertical Concrete Cask	Design Basis Surface Dose Rate (mrem/hr)	Surface Dose Rate (mrem/hr)	1-Foot Maximum Dose Rate (mrem/hr)
		BWR	BWR
Side wall	50.0 (avg.)	22.7	24.5
Air inlet$^{(1)}$	100.0$^{(2)}$	129	44.9
Air outlet	100.0$^{(2)}$	55	12.8
Top lid	50.0 (avg.)	19.7	15.7

Table Footnotes:
(1) The air inlet dose rates are based on the use of the air inlet shields and the design basis source terms require the use of the inlet shields to remain below the technical specification limits outlined in Appendix A of the SAR, supplemental shielding may be employed to reduce radiation exposure for certain tasks specified by the operating procedures. Use of supplemental shielding is at the discretion of the User;
(2) The air inlet and outlet average dose rate of 100 mrem/hr; and
(3) BWR dose rates from the vertical concrete cask for the additional BWR contents (10×10 assembly array, high burnup fuel and damaged fuel) are shown in Sections 5.8.1 through 5.8.2 of the SAR.

10.1.2 Design Features

The radiation shielding design is described in Section 5.3.1 of the SAR. The radiation exposure rates in the design criteria of the NAC-UMS® system are presented in Table 2-1 of the FSAR[11]. The principal radiation protection design features of the storage system are: shielding that meets the design criteria, the placement of penetrations near the edge of the canister shield lid to reduce operator exposure and handling time of casks, and the placement of shaped supplemental shielding for work on and around the shield lid, as necessary to comply with Part 50 and the licensee's radiation protection program. Section 8.1.1[12] of the FSAR, "Loading and

[10] Page 10.2-1 was submitted in FSAR Update Revision 3 (ADAMS Accession No. ML051290403).
[11] See page 2-3 submitted in FSAR Update Revision 3.
[12] Section 8.1.1 covers Pages 8.1.1-1 through 8.1.1-6. Page 8.1.1-1 was submitted in FSAR Update Revision 8. Page 8.1.1-2 was submitted in FSAR Update Revision 5 (ADAMS Accession No.

Closing the Transportable Storage Canister," describes the use of supplemental shielding to reduce operator dose rates during the welding, inspection, draining, drying and backfilling operations that seal the canister. The application also discussed optional supplemental shielding, shown in Drawing No. 790-613, Revision No. 2[13], which may be installed in the air inlets to reduce the radiation dose rate at the base of the vertical concrete cask.

Dose/exposures to workers during loading and maintenance and to members of the public were calculated for the additional BWR contents using MCNP. The evaluation of dose/exposures is discussed in Section 10.6 of the SAR.

Array calculations are based on mesh tallies. The tally multiplier is the source strength and is based on the design source region and the number of casks in the array. The source strength for each region is shown in Table 6-2 of the Calculation Package EA790-4604, Rev.0, "UMS High Burnup BWR Site Dose Calculation."

The applicant stated that in order to ensure that the results of mesh tallies are converged sufficiently, 12-point detectors were placed at different distances from the array, with locations consistent with the mesh tally locations for the independent spent fuel storage installation (ISFSI) dose rates. The detector locations (in meters) are summarized in Table 6-3 of the Calculation Package EA790-4604, Rev.0. All detectors were located at an elevation of 1.5 meters (m) above the ground. All detector distances were measured from the pad center (x=0, y=0). The point detector locations for occupational dose rates at the ISFSI was located in the x-direction at 8.1656 meter. The NRC staff concludes that the design features described in the SAR are consistent with guidance in NUREG-2215.

10.1.3 Design Basis for Accident Conditions

Damage to the VCC after a design basis accident does not result in a radiation exposure at the controlled area boundary in excess of 5 rem to the whole body or any organ for the member of the public. The high energy (tornado-driven) missile impact is estimated to reduce the concrete shielding thickness, locally at the point of impact, by approximately 6 inches. Localized cask surface dose rates for the removal of 6 inches of concrete are estimated to be less than 250 mrem/hr for the BWR configurations. A hypothetical accident event, tip-over of the vertical concrete cask, is considered in Section 3.9.1.4, "Concrete Cask" of the SAR. There is no design basis event that would result in the tip-over of the vertical concrete cask. The NRC staff concludes that the offsite doses from design basis accident conditions will meet the regulations in 10 CFR 72.236(d).

10.2 ALARA

Section 10.1 of the SAR presents the ALARA considerations for the NAC-UMS® storage system. Radiation protection design features and the design criteria address ALARA requirements consistent with the requirements in 10 CFR Part 20 and guidance provided in Regulatory Guides 8.8 and 8.10, "Operating Philosophy for Maintaining Occupational and Public Radiation Exposures As Low As Is Reasonably Achievable." SAR Section 10.3 includes the estimated on-site collective dose assessment, estimated collective dose for loading a single

ML053060314). Page 8.1.1-3 was submitted in FSAR Update Revision 4. Pages 8.1.1-4 through 8.1.1-6 were submitted in FSAR Update Revision 8.

[13] Drawing No. 790-613, Revision No. 2 was submitted with FSAR Update Revision 3.

NAC-UMS® storage cask, the estimated annual dose due to routine operations for ALARA and describes optional auxiliary shielding devices to minimize occupational and public doses.

The NRC staff evaluated the ALARA elements incorporated into the NAC-UMS® storage system design and found them to be acceptable. Based upon the information presented in the SAR, there is reasonable assurance that the ALARA objectives in 10 CFR 72.104(b) is met, as required by 10 CFR 72.236(d).

10.3 Occupational Exposures

SAR Section 10.4 discusses the estimated number of personnel, the estimated dose rates, and the estimated time for each task. The estimated occupational person-rem is based upon the minimum number of personnel needed to accomplish the activities in the general operating procedures and the dose rates determined from the shielding evaluation in SAR Chapter 5.

10.3.1 Estimated On-Site Collective Dose Assessment

The evaluation for the additional BWR payloads were discussed in Section 10.3 of the SAR with input from the MCNP evaluations discussed in Section 5.8 of the SAR (cask near field dose rates). The person-mrem exposure for operation of the NAC-UMS® system is presented in SAR Table 10.6-1 of the SAR. The applicant states that the tasks during storage include operation, maintenance and surveillance, and after identifying the tasks an estimate is made as to the duration and number of personnel performing each task. Dose rates are taken from the three-dimensional radial and axial shielding analyses of the NAC-UMS® transfer and storage casks. Next, exposures for each task are calculated for a single cask and an array of casks. Using the following equation,

$$\text{Exposure (person - mrem)} = \text{\#Casks} \times \text{Frequency of Task (per year)}$$

$$\times \text{ Dose Field (mrem / hr)} \times \text{Task Duration (hr)} \times \text{\# Personnel.}$$

The assigned dose rates are multiplied by task time and by the number of personnel to calculate the sub-task dose in person-mrem. Then, the exposures from all the tasks per cask and all the casks (for the array), are summed and presented in tabular form.

The dose estimates indicate that the total annual exposure for the operation and surveillance of a single BWR fuel is approximately 25.7 person-rem. The estimated yearly exposure for surveillance and cask maintenance for a 20-cask array with BWR Casks is approximately 14.0 person-rem.

The NRC staff reviewed the estimated occupational exposures for the additional BWR fuel and found them to be acceptable. The occupational exposure dose estimates provide reasonable assurance that occupational limits in 10 CFR Part 20 Subpart C can be achieved. Actual occupational doses will depend on site-specific parameters taken to maintain exposures ALARA.

10.4 Public Exposures

Section 10.6.2 of the SAR summarizes the calculated dose rates to members of the public located beyond the controlled area. The MCNP computer code was used to evaluate the dose rates at the controlled area boundary for a single storage cask containing design basis fuel, and

for a 20-cask array. For the 20-cask array, the storage casks were assumed to be loaded with design basis fuel per Section 5.8.2 of the SAR and evaluated at full time (8760 hours) exposure for a year. The storage cask array was explicitly modeled in the code, with the source term from each cask generated using the MCNP surface source write option. The surface source file is then imposed on each of the casks in the array. Exposures were determined at distances out to 600 meters surrounding a single BWR storage cask and an array of casks, each containing design basis fuel.

Table 10.6-4 of the SAR shows the dose versus distance for a single cask containing design basis high burnup fuel. Table 10.6-5 of the SAR shows the annual exposures from a 2×10 cask array containing design basis BWR high burnup fuel. For both tables, the point detectors were placed along the long side (y=1 m) and short side (x=1 m) of the array instead of along the central axis, with its locations consistent with the mesh tally locations for the ISFSI dose rates.

Linear interpolation of results shown in Table 10.6.4 of the SAR shows that minimum distances from a single cask to the site boundary of 173.89 meters for the design basis BWR high burnup fuel is required for compliance with the dose rate limits in 10 CFR 72.104(a), i.e., a dose rate of 25 mrem/year, as required by 10 CFR 72.236(d). Table 10.6-5 of the SAR show that a minimum site boundary of 340 meters is required for a 2×10 BWR cask array to meet dose rate limit in 10 CFR 72.104(a).

The NRC staff concludes that the direct radiation (including the contribution from skyshine) is the primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions. Public exposure from normal and off-normal conditions will be from direct radiation from the storage casks.

Each licensee who intends to use the NAC-UMS® storage system must perform a site-specific dose analysis to demonstrate compliance with all the requirements in 10 CFR Part 72. Site-specific boundary distances may vary based on fuel type, fuel cooling time, natural site barriers, and number of casks in service.

The NRC staff evaluated the public dose estimates from direct radiation for normal and off-normal (anticipated occurrences) conditions and found them to be acceptable. The NRC staff finds reasonable assurance that compliance with 10 CFR 72.236(d) can be achieved by each general licensee. The actual doses to individuals beyond the controlled area boundary depend on site-specific conditions such as cask array configuration, topography, demographics, and use of engineered features (e.g., berms). In addition, as required by 10 CFR 72.236(d), the dose limits in 10 CFR 72.104(a) must include doses from all other fuel cycle activities located onsite such as reactor operations. The general licensee will also have an established radiation protection program, as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public, as required by 10 CFR Part 20, Subpart D, by evaluations and measurements.

10.5 Conclusions

The SAR sufficiently describes the radiation protection design bases and design criteria for the SSCs important to safety for the NAC-UMS® storage system. Radiation shielding features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 and therefore meets the design requirement of 10 CFR 72.236(d). The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

The NRC staff concludes that the design of the radiation protection system for the NAC-UMS® storage system with the additional low and high burnup BWR fuels is in compliance with 10 CFR Part 72. The staff finds that the applicable design and acceptance criteria provided by NAC have been satisfied. The staff evaluated the radiation protection system design and finds it provides reasonable assurance that the NAC-UMS® storage system will safely storage of SNF for the duration of the license term. The staff considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices in making this safety finding.

Chapter 11 OPERATING PROCEDURES AND SYSTEMS EVALUATION

The applicant did not request changes to the operating procedures of the NAC-UMS® storage system and the changes in this amendment will not affect the operating procedures section of the storage system.

Chapter 12 CONDUCT OF OPERATIONS

The NRC staff review determined the application includes the appropriate acceptance tests and maintenance programs for the NAC-UMS® storage system. The revisions to SAR Chapter 9, "Acceptance Criteria and Maintenance Program" assessed the inclusion of a metal matrix composite neutron absorber for criticality safety. In addition, NAC made editorial changes to include the new neutron absorber material.

12.1 Neutron Absorber/Poison Tests for Boiling-Water Reactor Baskets

The SAR for Amendment No. 8, NAC details how B-10 uniformity and areal density specifications in TS Appendix B, 3.2.1 are determined for the borated metal matrix neutron absorber. NAC primarily uses the neutron attenuation method, along with wet chemistry to verify uniformity and minimum areal density for B-10. The applicant states that the minimum areal density shall be verified in each set of 50 borated neutron absorber sheets at the 95% probability, and at the 95% confidence level. The applicant described the neutron attenuation and wet chemistry procedures to be used in the NAC-UMS® storage system to ensure B-10 uniformity and density in SAR Chapter 9. The NRC staff finds these procedures are consistent with ASTM International Guide C1071-20, "Qualification and Acceptance of Boron Based Metallic Neutron Absorbers for Nuclear Criticality Control for Dry Cask Storage and Transportation Packaging." The staff finds these procedures to be acceptable.

12.2 Findings

F12.1 The NRC staff finds that SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards and that the function(s) the SSCs are intended to perform are sustained. The NRC concludes that Chapter 3 of the SAR describes the safety importance of SSCs and Chapters 1 and 3 present the standards for the design, fabrication, and testing of SSCs in NAC-UMS® storage system meet the requirements in 10 CFR 72.124(b), 10 CFR 72.162, 10 CFR 72.234(b), and 10 CFR 72.236(b), (g), (j) and (l).

Chapter 13 WASTE MANAGEMENT

This chapter is not applicable to an application for a certificate of compliance.

Chapter 14 DECOMMISSIONING EVALUATION

This chapter is not applicable to an application for a certificate of compliance.

Chapter 15 QUALITY ASSURANCE EVALUATION

NAC did not request changes to the quality assurance program for the NAC-UMS® storage system and the changes requested in the application do not affect the quality assurance program.

Chapter 16 Accident Evaluation

The NRC staff's evaluations of the applicant's accident analysis for the proposed changes are documented in Chapters 3 through 10 of this SER. Therefore, this section does not further evaluate accidents.

Chapter 17 TECHNICAL SPECIFICATIONS AND OPERATING CONTROLS AND LIMITS EVALUATION

The NRC staff reviewed the amendment request to determine whether proposed changes to conditions in the certificate of compliance and to technical specifications are in accordance with the requirements of 10 CFR Part 72. The NRC staff reviewed the proposed changes to the technical specifications to confirm the changes were supported in the applicant's revised safety analysis report.

17.1 Revision of Definitions

NAC revised the definitions for high burnup fuel, initial peak planar-average enrichment, and BWR damaged fuel can, and added BWR fuel to the definitions of damaged fuel and fuel debris in the Technical Specifications, Appendix NAC also revised the definition of high burnup fuel, by reducing the maximum burnup from 62,000 MWd/MTU to 60,000 MWd/MTU. This change is consistent with the maximum burnup in Technical Specifications, Appendix B. The revised definition of initial peak planar-average enrichment deleted the statement that the 4.7 wt% U-235 enrichment limit for BWR fuel applies along the full axial extent of the assembly. This deletion brings consistency with the enrichment limits in Technical Specifications, Appendix B. The NRC staff's evaluation of the new contents is described in Chapters 3 through 10 of this SER.

17.2 Addition of Contents

NAC revised Technical Specifications, Appendix B with the addition of high burnup evaluations for 8×8, 9×9, and 10×10 fuel assemblies. The new evaluations assess damaged and undamaged fuel, and low burnup, undamaged 10×10 BWR fuel assemblies. The NRC staff's evaluation of the new contents is described in Chapters 3 through 10 of this SER.

17.3 Findings

F17.1 The NRC staff concludes the conditions proposed for the certificate of compliance for the NAC-UMS® storage system identify technical specifications satisfy 10 CFR Part 72 and that the applicable acceptance criteria in the SAR have been satisfied and meets the requirements in 10 CFR 72.236(b).

CONCLUSION

The NRC staff performed a detailed safety evaluation of the application for Amendment No. 8 to Certificate of Compliance No. 1015 for the NAC-UMS® storage system. The NRC staff performed the review in accordance with the guidance in NUREG-2215. Based on the statements and representations contained in the application and the conditions established in the certificate of compliance and its Technical Specifications, the NRC staff concludes that these changes do not affect the ability of the NAC-UMS® storage system to meet the requirements of 10 CFR Part 72 and that the NAC-UMS storage system provides reasonable assurance of adequate protection of public health and safety, and protects the environment..

Issued with Certificate of Compliance No. 1015, Amendment No. 8,
on September 21, 2021.