FINAL SAFETY EVALUATION REPORT
NAC INTERNATIONAL, INC.
MAGNASTOR® STORAGE SYSTEM
DOCKET NO. 72-1031
AMENDMENT NO. 11

Enclosure 37

Table of Contents

Summary

By letter dated July 14, 2020 (Agencywide Documents Access and Management System [ADAMS] Package Accession No. ML20210M079), as supplemented on August 23, 2021 (ML21242A053), January 11, 2022 (ML22018A059), and July 15, 2022 (ML22196A022), NAC International, Inc., (NAC) submitted an application for amendment No. 11 to the Model No. MAGNASTOR® storage cask. In support of the application, the NAC applicant submitted revised safety analysis reports (SARs), revision nos. 20A, 21B, and 22B. The applicant proposes to:

- add a seventh concrete overpack (CC7) and a lightweight MAGNASTOR® transfer cask (LMTC)
- increase the maximum heat load for the system when using CC7 and the LMTC
- new loading patterns
- add new 81-assembly and 89-assembly boiling-water reactor (BWR) spent fuel basket designs, and associated loading patterns
- remove existing 87-assembly and 82-assembly BWR basket designs
- add a new BWR damaged fuel basket design with a capacity of up to 81 undamaged BWR fuel assemblies
- add a new damaged fuel can for BWR fuel

By letter dated March 18, 2022 (ML22077A769), and April 18, 2022 (ML22108A197), NAC requested that the proposed technical specification changes submitted in amendment no. 12 to the Model No. MAGNASTOR® storage cask, also be incorporated in amendment no. 11. The proposed changes include addition of a definition for the concrete cask lid and concrete cask body and alternate fabrication criteria and techniques in technical specification A4.2 for the concrete cask lid. Note that NAC added a definition of the CONCRETE CASK LID, which differentiates it from the CONCRETE CASK BODY, to the technical specifications. In this safety evaluation report (SER) use of the term “concrete cask” includes the cask body and the lid, unless otherwise specified.

The U.S. Nuclear Regulatory Commission (NRC) staff reviewed the amendment request using guidance in NUREG-2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities - Final Report,” dated April 2020. Chapter 2, “Site Characteristics Evaluation for Dry Storage Facilities,” chapter 13, “Waste Management Evaluation,” and chapter 14, Decommissioning Evaluation,” are not included in this evaluation since they are only applicable to a specific license application.

For the reasons stated below and based on the statements and representations in NAC’s application, as supplemented, and the conditions specified in the certificate of compliance (CoC) and the technical specifications, the staff concludes that the requested changes meet the requirements of Title 10 of the *Code of Federal Regulations* (10 CFR) Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater Than Class C Waste.”

Chapter 1 GENERAL INFORMATION EVALUATION

The objective of the review of this chapter is to evaluate design changes made to the MAGNASTOR® storage system to ensure that NAC provided a description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system, including the requested changes.

1.1 General Description and Operational Features

The MAGNASTOR® system is a spent fuel, dry storage system consisting of a storage overpack containing a welded, stainless-steel transportable storage canister (TSC), which contains the spent fuel, and a transfer cask. In the storage configuration, the TSC is placed in the central cavity of the storage overpack. The storage overpack provides structural protection, radiation shielding, and internal airflow paths that remove the decay heat from the TSC surface by natural air circulation. The storage overpack also provides protection during storage for the TSC and the spent fuel it contains against adverse environmental conditions. The MAGNASTOR® system is designed to accommodate storage of up to 37 pressurized-water reactor (PWR) fuel assemblies or 89 BWR fuel assemblies.

The transfer cask is used to move the TSC between the workstations during TSC loading and preparation activities, and to transfer the TSC to or from the overpack. There are two approved designs for the transfer cask, the standard MAGNASTOR® transfer cask (MTC) and the passive MAGNASTOR® transfer cask (PMTC). The MTC provides shielding during TSC movements between workstations, the overpack, or the transport cask. It is a multiwall (carbon steel/lead/NS-4-FR/steel) design with retractable (hydraulically operated) bottom shield doors that are used during loading and unloading operations. There is a second version of the MTC, called the MTC2. The only difference from the MTC is that the MTC2 has stainless-steel walls. This amendment adds a third transfer cask, the LMTC.

1.1.1 Storage Overpack

NAC proposed adding CC7 which is cylindrical with a variable height ranging from 210 inches to 217 inches, which includes a 3-inch thick carbon steel liner. The concrete shell is 25.3 inches thick with rebar of various lengths. The diameter of the lid is 136 inches, and its thickness is 12 inches. CC7 is equipped with additional shielding at the air inlets.

MAGNASTOR® storage system included three variations of the concrete overpack lid, two variations are shown in drawing no. 71160-L261, Sheet 5 and the third variation is shown on drawing no. 71160-664. Note that the lid shown on drawing no. 77160-561, Sheet 3 is the same design as the lid shown on drawing no. .71160-L261, Sheet 5, section K-K. Note that the lid shown on drawing no. 71160-664, Sheet 3 is the same design as the lid shown on drawing no. 71160-L364, Sheet 1.

1.1.2 Transportable Storage Canister

In this amendment, NAC added a fifth TSC (TSC5), which is closed by an 8-inch thick solid stainless-steel closure lid, which is welded to the shell. The shell and bottom plate are the same as the other four TSCs. The shell is constructed of ½-inch thick stainless-steel and with a 72 inch diameter. The bottom plate is constructed of 2.75-inch-thick stainless-steel plate welded onto the shell.

1.1.3 Transfer Cask

NAC added an LMTC whose structural components are all fabricated from stainless-steel. Radially, the LMTC is constructed of an inner shell, variable lead thickness, intermediate shell, and a liquid neutron shield that is formed by eight neutron shield sectors, which form the LMTC's outer shell. The inner shell is constructed of ½-inch thick stainless-steel and has a 73 inch inner diameter. It is surrounded by a lead that ranges from 2.5 to 4 inches thick. The intermediate shell is constructed of ¼-inch thick stainless-steel. The neutron shield sectors, which hold the liquid neutron shield, are constructed from ½-inch thick stainless-steel plate. The liquid neutron shield has a variable thickness between 3 and 4.5 inches and is filled with demineralized water.

1.2 Drawings

In support of this application, NAC submitted the following 28 drawings for NRC review:

Drawing No. 71160-561, Revision 11, – "Structure, Weldment, Concrete Cask, MAGNASTOR"
Drawing No. 71160-562, Revision 11 – "Reinforcing Bar and Concrete Placement, Concrete Cask"
Drawing No. 71160-581, Revision 6, – "Shell Weldment, PWR TSC, MAGNASTOR"
Drawing No. 71160-584, Revision 10, – "Details, PWR TSC, MAGNASTOR"
Drawing No. 71160-585, Revision 14, – "TSC Assembly, PWR, MAGNASTOR"
Drawing No. 71160-590, Revision 10, – "Loaded Concrete Cask, MAGNASTOR"
Drawing No. 71160-L104, Revision 0P, – "Damaged Fuel Can (DFC) BWR, MAGNASTOR"
Drawing No. 71160-L178, Revision 0P, – "Corner Weldment, BWR DF Basket, MAGNASTOR"
Drawing No. 71160-L180, Revision 0P, – "Basket Assembly, BWR DF, MAGNASTOR"
Drawing No. 71160-L186, Revision 0P, – "TSC Assembly, BWR DF, MAGNASTOR"
Drawing No. 71160-L257, Revision 0P, – "Cask Assembly, Lightweight MAGNASTOR Transfer Cask (LMTC)"
Drawing No. 71160-L258, Revision 0P, – "Cask Body Weldment, Lightweight MAGNASTOR Transfer Cask (LMTC)"
Drawing No. 71160-L261, Revision 0P, – "Structure, Weldment, Concrete Cask, MAGNASTOR"
Drawing No. 71160-L262, Revision 0P, – "Reinforcing Bar and Concrete Placement, Concrete Cask, MAGNASTOR"
Drawing No. 71160-L272, Revision 0P, – "Details, Neutron Absorber, Retainer, BWR, MAGNASTOR"
Drawing No. 71160-L290, Revision 0P, – "Loaded Concrete Cask, MAGNASTOR"
Drawing No. 71160-L291, Revision 0P, – "Fuel Tube Assembly, BWR, MAGNASTOR"
Drawing No. 71160-L297, Revision 0P, – "Side Support Weldment, BWR, MAGNASTOR"
Drawing No. 71160-L298, Revision 0P, – "Corner Support Weldment, BWR, MAGNASTOR"
Drawing No. 71160-L361, Revision 0P, – "Structure, Weldment, Concrete Cask, MAGNASTOR"
Drawing No. 71160-L362, Revision 0P, – "Reinforcing Bar and Concrete Placement, Concrete Cask MAGNASTOR"
Drawing No. 71160-L363, Revision 0P, – "Lift Lug and Details, Concrete Cask, MAGNASTOR"
Drawing No. 71160-L364, Revision 0P, – "Upper Segment Assembly, Concrete Cask, MAGNASTOR"
Drawing No. 71160-L381, Revision 0P, – "Shell Weldment, BWR TSC, MAGNASTOR,"
Drawing No. 71160-L384, Revision 0P, – "Details, Closure Lid, BWR TSC, MAGNASTOR"
Drawing No. 71160-L385, Revision 0P, – "TSC Assembly, BWR, MAGNASTOR"
Drawing No. 71160-L390, Revision 0P, – "Loaded Concrete Cask, MAGNASTOR"
Drawing No. 71160-L399, Revision 0P, – "Basket Assembly, BWR TSC, MAGNASTOR

1.3 Contents

NAC added two new BWR fuel basket designs, one which holds up to 89 undamaged BWR fuel assemblies and the second which holds up to 81 undamaged BWR fuel assemblies, of which 12 may be damaged fuel in DFC locations. NAC added damaged BWR fuel, including a new DFC for storing damaged BWR fuel in the CoC.

NAC added three new heat load zoning patterns (Patterns I, J, and K) for use with existing new fuel qualification tables and increased the heat load capacity for the new heat load patterns. For previously existing loading patterns A through D, NAC made loading patterns B, C, and D for use with the new 89-assembly BWR basket configuration, and loading patterns A, B, and C for use with the new 81-assembly BWR basket configuration. NAC also removed the 87--assembly and 82-assembly BWR fuel basket configurations since the spent fuel assembly characteristics are bounded by the new 89--assembly and 81-assembly basket configurations (e.g., higher burnup and decay heat). However, the final safety analysis report (FSAR) analysis and licensing drawings remain as they partially support the evaluations justifying the approval of the new 89-assembly and 81-assembly configurations.

1.4 Evaluation Findings

Based on the NRC staff's review of information provided for amendment no. 11 to the MAGNASTOR® system, the staff determined the following:

F1.1 A general description and discussion of amendment no. 11 to MAGNASTOR® 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 to familiarize a reviewer or stakeholder with the design.

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

Chapter 3 PRINCIPAL DESIGN CRITERIA EVALUATION

The changes associated with principal design criteria for the addition of the new concrete overpack, LMTC, and associated basket and contents are discussed and evaluated in subsequent chapters of this SER.

Chapter 4 STRUCTURAL EVALUATION

The staff reviewed the proposed changes to the MAGNASTOR® system for spent fuel storage to verify that the applicant performed acceptable structural evaluations demonstrating that the system, as proposed, meets the requirements of 10 CFR Part 72. The staff’s review focused on the addition of a new transfer cask known as the LMTC and the addition of a new design of a concrete cask known as CC7. These additions provide additional options for the previously approved transportable storage container and the MAGNASTOR® concrete cask system for spent nuclear fuel.

4.1 Description of the Structures

Lightweight MAGNASTOR® Transfer Cask

The new transfer cask known as the LMTC is primarily a shielded lifting device used to handle the TSC. It provides biological shielding for a loaded TSC with high heat loads. The LMTC includes a demineralized water-filled neutron shield tank that can be drained for pool loading operations to reduce the hook wet weight, then refilled to restore neutron shielding prior to performing canister draining, drying, and closure operations. The LTMC structural components are all fabricated from stainless-steel.

Concrete Cask Number 7

CC7, is a reinforced concrete cylinder cask, which has a similar design to the concrete overpacks that the staff previously reviewed and approved (CC1) in the FSAR, amendment no. 0 (ML090350509). Dimensionally, the CC7 has an outside diameter of 136 inches and an overall height of 210 to 221 inches depending on the configuration. The internal cavity of the concrete cask is lined with a carbon steel shell with an inside diameter of 79.5 inches and a thickness of 3.0 inches. The concrete shell, constructed using Type II Portland cement, has a nominal density of 145 pounds per cubic foot (pcf) and a compressive strength of 4,000 pounds per square inch (psi) at ambient temperature.

A ventilation airflow path is formed by inlets at the bottom of the concrete cask, the annular space between the concrete cask inner shell and the TSC, and outlets in the concrete cask lid assembly. The passive ventilation system operates by natural convection as cool air enters the bottom inlets, is heated by the contents of the TSC, and exits from the outlets. Both the air inlets and air outlets are formed with carbon steel in the concrete cask body. For CC7 configuration, similar to the CC3, CC4 and CC5, a labyrinth of steel bars is included in each inlet vent for enhanced radiation protection.

4.2 Design Criteria

The structural design criteria and classification used for the design of the LMTC and CC7 are described in chapter 2 “Principal Design Criteria” of the FSAR, Rev. 8 (ML17038A506), which was previously reviewed and accepted by the staff. The applicant stated that the CC7 body is a reinforced concrete structure that is designed in accordance with the requirements of American Concrete Institute (ACI) 349, “Code Requirements for Nuclear Safety Related Concrete Structures,” and evaluated per American National Standard Institute an American National Standard (ANSI/ANS) 57.9, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type)”. The CC7 body with heat shield is evaluated for critical-lift along with the CC7 upper segment using ANSI N14.6, “Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 Kg) or More.” Non-critical-load lifts are evaluated using American Society of Mechanical engineers (ASME) Boiler and Pressure Vessel Code (BPV), Section III, Division 1 - Subsection NF. The LMTC lifting devices are designed, load-tested and fabricated in accordance with the requirements of ANSI N14.6 and NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants.”

The staff reviewed the design criteria and found the designs of the LMTC and CC7 are in accordance with acceptable design codes and standards and meet the requirements of 10 CFR 72.236(b).

4.3 Structural Evaluation

The applicant evaluated the MAGNASTOR® system with the LMTC, TSC, and CC7 using both hand calculations and finite element analysis (FEA). The applicant provided four calculation packages to support the structural analysis: (i) calculation package no. 71160-2021, Rev. 0, “Structural Evaluation of CC3, CC5 and CC7 Concrete Casks with Heat Shield,” (ii) calculation package no. 71160-2027, Rev 0, “LMTC Structural Evaluation,” (iii) calculation package no. 71160-2031, Rev 0, “LS-DYNA Tip-Over Analysis for CC7 Concrete Cask,” and (iv) calculation package no. 71160-2033, Rev. 0, “BWR DF Basket Structural Evaluation for Storage Condition.” The applicant described the structural evaluations in section 3.11 of the SAR.

4.3.1 Evaluation for Lightweight MAGNASTOR® Transfer Cask

The applicant describes the evaluation of lifting LMTC in section 3.4.3.3.4 of the SAR, which is supported by calculation package 71160-2027. This LMTC evaluation includes vertical lift and inadvertent lift. The vertical lift analysis included the vertical lift with a fully loaded weight and an applied dynamic factor of 1.1, as well as the vertical lift with a loaded LMTC after vacuum drying weight including an applied dynamic factor of 1.1. The evaluation consisted of an FEA using the ANSYS program to calculate the induced stresses in the LMTC forgings, shells, and the trunnion region for the vertical lift condition. The structural evaluations of the door rails, door assemblies, rail welds, and trunnions were performed by the applicant using hand calculations.

For the inadvertent lift condition, the applicant used an FEA using the ANSYS program to determine the stresses in the components. Hand calculations were performed using standard engineering equations to ensure that the bolt tensile stress, shear stress at the bolt threads, and shear stress at the bolt hole threads were below the code required limits. The applicant evaluated the hydrostatic load for the neutron shield’s internal pressure load, using ANSYS program, to determine the maximum stress in the LMTC shells. Additionally, hand calculations were performed using standard engineering equations to ensure that the manifold and port weld stresses were below the required the ASME BPV Code stress limits.

The staff reviewed the analyses and the results of the LMTC evaluation documented in the SAR section 3.4.3.3.4, which provide the following results:

<u>Vertical lift</u>:

1. Minimum factor of safety (FS) for yield = 6.3, which is greater than the required FS of 6.0 for yield strength (S_y).
2. Minimum FS for ultimate = 14.7, which is greater than the required FS of 10.0 for ultimate strength (S_u).
3. Allowable load for handling shield door = 12,628 lbs., which is greater than the actual door weight of 4,140 lbs. Therefore, FS = 3.05, which is greater than the required FS of 1.0.

In accordance with ANSI N14.6 and NUREG-0612, to qualify the cask to be lifted as part of a special lifting device, these calculated FSs are greater than the required FSs by the ASME BPV Code and, therefore, are acceptable.

Inadvertent lift:

1. Retaining Ring (off-normal condition) FS for primary membrane plus primary bending is 1.21, which is greater than the required FS of 1.0.
2. Retaining Ring Bolts (off-normal condition) FS for tensile stress = 1.33, which is greater than the required FS of 1.0.

The applicant performed evaluation of the door rails which are attached to the bottom forging of the transfer cask by $1^1/_8$-inch partial penetration bevel groove welds with ¼-inch cover fillet welds along the rail inner surface and at the outer diameter of the cask. In accordance with the requirements of NUREG-0612 and ANSI N14.6, the stresses in the door rail attachment welds are limited to the lesser of $S_y/6$ and $S_u/10$. The calculated FSs for S_y and S_u equal to 8.67 and 22.1, which are greater than the required FSs of 6.0 and 10.0, respectively.

Based on NRC review of the lifting evaluations described above, since the results show that sufficient design margin is provided, the staff finds that the LMTC has sufficient structural capacity to withstand lifting and that the evaluations of the LMTC are acceptable.

4.3.2 Evaluation for Concrete Cask Number 7

4.3.2.1 Design Load Combinations of Concrete Cask Number 7

The load combinations used for the evaluation of the CC7 are identical to the load combinations previously used for the evaluation of the CC1 through CC5 described in the SAR, amendment no. 8. Table 4.1 of this SER below provides a summary of the eight load combinations used for the structural evaluations of the CC7, where:

DL = Dead Load
LL = Live Load
To = Normal Temperature
W = Wind
Wt = Tornado/Tornado Missile
E = Design-Basis Earthquake
FL = Flood
Ta = Off-Normal or Accident Temperature
A = Drop/Impact

The staff reviewed the load combinations in table 4-1 and found that they comply with the code requirements in ANSI 57.9 and ACI 349 and, therefore, are acceptable.

Table 4.1 – Load Combination for CC7 Concrete Cask Evaluation

Load Combination	Event	Loads
1	Normal	1.4 DL + 1.7 LL
2	Normal	1.05 DL + 1.275 (LL + To)
3	Normal	1.05 DL + 1.275 (LL + To + W)
4	Off-Normal and Accident	DL + LL + Ta
5	Accident	DL + LL + To + E
6	Accident	DL + LL + To + A
7	Accident	DL + LL + To + FL
8	Accident	DL + LL + To + Wt

4.3.2.2 Lift Analysis of Concrete Cask Number 7

The applicant used a combination of FEA using the ANSYS computer program and hand calculations to evaluate the concrete cask lift. This analysis method is the same method used for the evaluations of the concrete casks CC1 through CC5.

The applicant provided the results of the lift analysis of the CC7 in the calculation package, 71160-2021, Rev. 0. Table 4.2 of this SER provides the results of the analysis. The staff reviewed the stress calculations and found that the calculated stresses induced by the lifting operations are less than the allowable stresses in the ASME BPV Code and the calculated FSs are larger than the required FSs specified in NUREG-0612 and ANSI N14.6.

Based on NRC staff's review of the lift analysis and results, and since the results show that sufficient design margin is provided, the staff finds the design of the CC7 for lift is acceptable.

4.3.2.3 Structural Analysis of Concrete Cask Number 7 for Combined Load Conditions

<u>Normal Conditions</u>

The applicant evaluated the normal condition events for the CC7 design in the calculation package, 71160-2021, Rev. 0. The applicant considered load combinations 1, 2 and 3 of table 4.1, above, and calculated a maximum stress of 2,170 psi on the heat shield inner surface of the CC7 using the ANSYS computer program when the Load Combination 3 [1.05 DL + 1.275 (LL + To + W)] was applied to the cask for the normal conditions. Since the allowable compressive stress of the heat shield inner surface of CC7 concrete cask is 2,660 psi at 300°F, a minimum FS for the normal and off-normal conditions is 1.22.

Based on NRC staff's review of the normal conditions analyses on CC7 and the associated results, and because the calculated FS of 1.22 is greater than the minimum required FS of 1.0, the staff finds the design of the CC7 against the normal conditions acceptable since none of the components would fail when subjected to loads for normal conditions.

Accident Conditions

The applicant also evaluated the accident condition events for the CC7 design in the calculation package no. 71160-2021, Rev. 0. The applicant considered load combinations 4, 5, 7 and 8 of table 4.1, and calculated a maximum stress of 1,814 psi on the heat shield inner surface of the CC7 for the accident conditions when the Load Combination 5 of (DL + LL + To + E) was applied. Based on the allowable compressive stress for CC7 of 2,660 psi at 300°F, the applicant calculated a minimum FS for the accident conditions of 1.47.

Based on NRC staff's review of the accident conditions analyses on CC7 and the associated results, and because the calculated FS of 1.47 for compressive stress is greater than the required minimum FS of 1.0, the staff finds the design of the CC7 against the accident conditions is acceptable since none of the components would fail when subjected to loads for accident conditions.

Table 4.2 – Evaluation of CC7 Concrete Cask Components for Lift

Component	Strength	Calculated Factor of Safety (FS)	Required Factor of Safety (FS)
Lift Lug (Bearing)	Yield Ultimate	4.0 5.2	3.0 5.0
Embedded Lift Ring Weldment	Yield Ultimate	4.1 7.2	3.0 5.0
Lift Lug Bolt at - thread	Yield Ultimate	4.5 5.7	3.0 5.0
Lift Lug Bolt at – shank	Yield Ultimate	4.5 5.7	3.0 5.0
Lift Lug Bolt- thread in shear	Yield Ultimate	9.3 11.8	3.0 5.0
Lift Lug Weldment	Yield Ultimate	4.6 7.5	3.0 5.0
Upper Segment	Yield Ultimate	8.6 11.1	6.0 10.0
Concrete	Bearing	2.9	1.0
Concrete	Shear	2.7	1.0

4.3.2.4 Stability Analysis of Concrete Cask Number 7 for Tornado and Tornado-Generated Missiles

In SAR section 3.11.3.4.2, the applicant performed an overturning analysis of the CC7 under tornado wind loading. The applicant used the same analytical approach that was previously used for the overturning analyses of the CC1 through CC5. The applicant considered the maximum wind pressure, gust factor, and cask dimensions, and calculated an overturning moment of 4.4 x 10^5 ft-lb. The applicant also calculated the stability moment of 1.13 x 10^6 ft-lb for the cask. A minimum FS of 1.72 against overturning was calculated using the method in the American Society of Civil Engineers, Standard No. 7-93.

Based on NRC staff's review of the stability analyses on CC7 for tornados and tornado missiles, and the associated results, and because the calculated FS of 1.72 is greater than required minimum FS of 1.0, the staff finds that the evaluation of the CC7 against the tornado wind loading is acceptable since CC7 will not overturn during a design-basis tornado wind loading or tornado missile impact.

4.3.2.5 Stability Analysis of Concrete Cask Number 7 for Flooding

The applicant performed an overturning analysis of the CC7 under a design-basis flood accident. The applicant used the same analytical approach that was previously reviewed and approved by the staff for the overturning analyses in amendment no. 0 (ML090350509). The applicant considered the factors (i.e., drag force of the flood, cask dimensions, etc.) and calculated that a floodwater velocity of 20.4 ft/sec is required to overturn the cask, which is greater than the concrete cask design-basis floodwater velocity of 15.0 ft/sec per the SAR. The applicant calculated a FS = 20.4/15.0 = 1.36, which is greater than the required minimum FS of 1.0 against the flood loading.

Based on NRC staff's review of the stability analyses on CC7 for floods and the associated results, and because the calculated FS is greater than the minimum required by the American Society of Civil Engineers Standard 7-93 "Minimum Design Loads for Buildings and Other Structures," the staff finds that the evaluation of the CC7 against the flood loading is acceptable since CC7 would not overturn during a design-basis flood.

4.3.2.6 Stability Analysis of Concrete Cask Number 7 for Earthquake

The applicant performed an overturning analysis of the CC7 under a design-basis earthquake accident. The applicant used the same analytical approach that was previously used for the CC1 through CC5, which is presented in section 3.7.3.4 of the FSAR, Rev, 8. The applicant calculated a minimum horizontal acceleration of 0.407g for the CC7 to resist an overturning. The applicant used a standardized design earthquake ground motion (DE) described by an appropriate response spectrum anchored at 0.25 g. As a result, the FS against overturning of the CC7 under earthquake loading is FS = 0.407g/0.25g = 1.63, which is larger than the required FS of 1.1.

Based on the staff's review of the applicant's evaluation of the CC7 against the earthquake loading and the associated results, and because the calculated FS is greater than the minimum required by the American Society of Civil Engineers Standard 7-93, the staff finds the evaluation acceptable since CC7 would not overturn during a design-basis earthquake.

4.3.2.7 24-inch Drop Analysis for Concrete Cask Number 7

The applicant calculated a crush depth of the CC7 concrete cask for the 24-inch drop using an energy balance equation, which was previously accepted by the staff when the staff reviewed the SAR amendment no. 0 with the calculation package 71160-2009, Rev. 0, "Evaluation of NewGen VCC [vertical concrete cask] for a 24-inch drop." The drop height, cross sectional area, weight, and compressive strength of the concrete cask were considered in the equation, and a crush depth of 0.126 inch was calculated. The applicant did not further evaluate the 24-

inch drop analysis for the CC7 because this crush depth of 0.126 inch of the CC7 is less than the crush depths of 0.13 inch of the first two concrete casks (CC1 and CC2), which were previously reviewed and approved by the staff.

The staff reviewed NAC's analysis and results for the 24-inch drop. Based on the staff's review, it finds that the evaluation of the CC7 under the 24-inch drop is acceptable because the analysis followed a similar method previously used, which is applicable to CC7, and the results are bounded by those from CC1 and CC2.

4.3.2.8 Structural Analysis of Concrete Cask Number 7 for Tip-Over

The applicant performed an evaluation of the CC7 under a non-mechanistic tip-over using an explicit nonlinear dynamic FEA in the LS-DYNA FE program to determine the acceleration time histories of the fuel basket and TSC during a tip-over event with impact on the concrete storage pad. This analytical approach is the same approach used for the tip-over analyses for the CC1 and CC2 concrete cask, which were previously reviewed and approved by the staff in amendment no. 0 (ML090350509).

The applicant calculated the peak accelerations of 25.5g and 27.2g for the top of the basket and the TSC, respectively. The comparison in table 4.3 below of the calculated peak accelerations with the design-basis accelerations shows that the calculated accelerations for CC7 are bounded by the peak accelerations reported in SAR section 4.7.3.7.

The staff reviewed the applicant's evaluation of the CC7 under a non-mechanistic tip-over event. Based on the staff's review of NAC's analyses, the staff finds it acceptable because the analysis followed a similar method previously used, which is applicable to CC7, and the results are bounded by those from CC1.

Table 4.3 – g-load at the Top of the Fuel Basket and TSC

Cask Type	**Method**	**Fuel Basket (Design Base = 35.0g)**	**TSC (Design Base = 40.0g)**
CC1 and CC2	LS-DYNA	26.4g	29.5g
CC5	LS-DYNA	25.8g	26.7g
CC7	LS-DYNA	25.5g	27.2g

4.3.3 Evaluation for Fuel Basket and Damaged Fuel Can

4.3.3.1 PWR Basket and PWR Damaged Fuel Basket

The applicant evaluated the PWR and PWR damaged fuel baskets for higher thermal stresses, which are produced by the increased heat load and the different loading patterns of fuel within the basket, as described in section 4.11 of the SAR, Rev. 20A. The applicant calculated the combined normal handling plus thermal stress for the PWR and PWR damaged fuel basket. The maximum stress occurred in the fuel tube and was equal to 6.0 thousand pounds per square inch (ksi). The allowable stress per the ASME BPV Code is 3 times the primary membrane stress ($3\times S_m$), which is equal to 62.6 ksi (SA-537 Class 1 steel at 750°F). Therefore, a large margin exists between the maximum calculated stress and the maximum allowable stress in the ASME BPV Code. Further, the applicant stated that the calculated relative thermal expansion between the adjacent tubes is approximately 0.07 inch, which is less than the pin gap of 0.10 inch. Hence, there are no axial thermal stresses produced by the axial expansion of the basket.

The staff reviewed the applicant's analysis and results for the PWR basket and the PWR damaged fuel basket, and the staff finds the analysis and results acceptable because the calculated stresses were below the maximum allowable stresses in the ASME BPV Code.

4.3.3.2 BWR Fuel Basket and BWR Damaged Fuel Basket

The applicant evaluated the BWR and BWR damaged fuel baskets for higher thermal stress, which is produced by the increased heat load and the different loading pattern in the basket, as described in section 4.11 of the SAR, Rev. 20A. The BWR damaged fuel basket with square pins was analyzed for normal, off-normal, and accident conditions of storage, including a hypothetical tip-over accident. The results of the analysis show that the BWR damaged fuel basket satisfies the design criteria set forth in ASME BPV Section III-NG and ASME BPV Section III Appendix F. In the SAR and the calculation package no. 71160-2033, Rev. 0, "BWR DF Basket Structural Evaluation for Storage Condition," the applicant presented the results for all of the structural components of the BWR and BWR damaged fuel basket. The results showed that the factors of safety are greater than 1.0 as set forth by the MAGNASTOR® design criteria.

The staff reviewed NAC's analysis and results for the BWR basket and the BWR damaged fuel basket, and the staff finds the analysis and results are acceptable because the calculated stresses were below the maximum allowable stresses in the ASME BPV Code which result in factors of safety greater than 1.

4.3.3.3 BWR Damaged Fuel Can

The applicant proposed to add a BWR DFC which holds the equivalent mass of a damaged fuel assembly in the BWR damaged fuel basket. The primary function of the DFC is to confine the fuel material within the can to minimize the potential for dispersal of the fuel material into the TSC cavity. In the normal condition of storage, the DFC is in a vertical orientation. The weight of the contents in the can is transferred directly through the bottom plate of the can to the TSC bottom plate, and the DFC is subjected to its self-weight only.

The applicant evaluated the BWR DFC in section 3.4.3.4.2, "BWR DFC Lift Evaluation," of the SAR, Rev. 20A for three structural components (lifting tab, tube body, and bottom plate) of the BWR DFC. The analysis method for the evaluation was identical to the analysis method for the evaluation of the PWR DFC that the staff previously reviewed and approved in amendment no. 3 (ML13207A245). The lifting tab component of the DFC was evaluated using the criteria defined in section 5.1.6 (3) (b) of NUREG-0612, and the other two components (tube body and bottom plate) were evaluated using the criteria defined in ASME BPV, Section III, Subsection NG.

The applicant presented the results of the analyses for the structural components of the BWR DFC in the SAR and in the calculation package no. 71160-2033, Rev. 0. The results showed that the calculated factors of safety are greater than 1.0 as set forth by the MAGNASTOR® design criteria.

The staff reviewed NAC's analysis and results for the DFC, and staff finds that the results of the analysis are acceptable because the calculated stresses in the lifting tab, tube body and bottom plate were below the maximum allowable stresses in the ASME BPV Code.

4.4 Concrete Cask Lid

4.4.1 Concrete Lid Construction

The NRC staff reviewed the proposed changes to technical specification A4.2 for the concrete cask lid and noted that the proposed technical specification includes the following controls for the concrete used in the construction of the concrete cask lid:

- The concrete shall, at minimum, be a commercial grade ready-mix type that can develop a density of 140 pounds per cubic foot (pcf).
- The concrete mix and batching should meet the purchaser's requirement for density and any additional purchaser-indicated attributes, such as air content, as allowed by ASTM International (ASTM) Standard ASTM C94, "Standard Specification for Ready-Mixed Concrete."
- The density of the concrete can be verified by either test method ASTM C138, "Standard Test Method for Density (Unit Weight), Yield, and Air Content (Gravimetric) of Concrete," or an approved shop fabrication procedure by following the equation for density, where density is equal to weight divided by volume. The shop procedure shall include steps to weigh the lid before and after concrete placement and in calculating the actual volume of the cavity to be filled with a record of the weight of concrete placed into the cavity.
- The concrete placement shall be in a dry and clean cavity or form with procedures and equipment that ensure the concrete placed is thoroughly consolidated and worked around any reinforcement and/or embedded fixtures and into the corners of the cavity or form.
- The concrete shall be protected from the environment during curing to minimize development of cracks by one or more of various methods such as moist cure or liquid membrane forming chemicals. Type II Portland cement may be substituted by an alternate cement type for the concrete if the above density requirement can be met.

The staff noted that the current MAGNASTOR® technical specifications (i.e., prior to the submittal of this amendment) do not differentiate between the concrete cask and the concrete cask lid, and they require that the entirety of ACI 349 and ACI 318 govern the design and construction of the entire cask (including the lid). These ACI standards provide comprehensive and detailed requirements for the design and construction of structural concrete. The staff considered whether the more limited set of requirements proposed in technical specification A4.2 for the concrete in the cask lid, in lieu of the requirements of ACI 349 and ACI 318, are suitable to ensure that the lid can adequately perform its required safety function.

As addressed in the SER section below regarding the concrete cask lid structural function, the concrete used in the construction of the concrete cask lid has a radiation shielding safety function but no structural strength requirements. Section R3.3.1 of ACI 349 states that shielding requirements for concrete components are dependent on the density of the concrete. Section R1.4 of ACI 349 cites ANSI/American Nuclear Society (ANSI/ANS) 6.4, "Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants," as specific guidance for evaluating the radiation shielding effectiveness of concrete components. The staff verified that the technical specification minimum density of 140 pcf meets the minimum density requirement of ANSI/ANS6.4 for concrete that performs a radiation shielding function.

The staff also verified that the proposed technical specification requirements for determining concrete density will ensure the measurement of concrete weight and volume are correctly performed and that the density is correctly calculated based on the measured weight divided by the measured volume. The staff noted that these methods are sufficient to ensure that the density of the concrete in the lid meets the technical specification requirement of 140 pcf. Therefore, the staff determined that the proposed technical specification is acceptable for ensuring that the concrete in the lid will have the density needed to adequately perform its radiation shielding safety function.

For the commercial grade concrete in the lid to maintain physical characteristics needed to adequately perform the radiation shielding function, the staff identified that the finished concrete should not undergo unacceptable shrinkage, and it should remain free of significant defects (such as voids and cracks) that could cause unacceptable radiation streaming through the concrete in the lid. Therefore, in addition to density, the staff considered whether the proposed technical specification for construction of the concrete cask lid are adequate to ensure that the concrete can maintain the physical properties (i.e., lack of unacceptable shrinkage and lack of significant voids or cracks) needed to perform its radiation shielding safety function during the operating life of cask. The NRC staff's evaluation of concrete shrinkage that may result in a loss of radiation shielding performance is provided in the SER section below.

4.4.2 Concrete Cask Lid Structural Function

The lid to the storage cask of the MAGNASTOR® storage system provides protection to the TSC within from the external environment including any postulated tornado missiles strike. The lid, in addition, provides protection against sky shine radiation.

The minimum lid thickness (¾ inch) carbon steel cover plate spans the entire opening. The complete lid assembly is bolted to the top of the concrete cask. Drawing 71160-561, Rev. 9, Sheet 3 of 5 section D-D shows a cross-section of the cask lid assembly with the alternate arrangement shown in drawing no. 71160-561, Rev. 9 Sheet 5 of 5 section E-E. The third lid configuration used with concrete cask number 6 is shown in drawing no. 71160-664, Rev. 1P, Sheet 2 of 2.

The staff's evaluation of tornado missiles is in section 3.5.2 "Tornado Wind and Tornado-Driven Missiles" of the SER amendment no. 0 (ML090350589). In that SER, the staff agreed with the FSAR conclusion that for a 280 lb, 8-inch diameter armor piercing shell traveling at 185 ft/sec impacting the 3/4-inch carbon steel top plate lid assembly, the carbon steel lid is adequate in preventing plate perforation with a FS of 1.15 (0.75/0.65 =1.15). The upper plate for the lid on concrete cask number 6 is 1 inch thick, which would yield a higher FS.

The staff confirmed that there is no structural strength demand on the concrete in the cask closure lid whose sole function is to provide radiation shielding.

4.4.3 Concrete Shrinkage

Concrete shrinkage is a reduction in the dimensions of a formed concrete component that occurs when hardened concrete dries from a saturated condition, as discussed in NUREG-2214, "Managing Aging Processes in Storage (MAPS) Report." For concrete components of certain dimensions that are relied upon to provide radiation shielding (i.e., to reduce external dose rates to acceptable levels), concrete shrinkage may have the potential to cause a reduction in the dimensions of the component by an amount that results in unacceptable radiation streaming and unacceptable external dose rates. Concrete shrinkage occurs initially during curing and can be controlled through concrete formulation. According to ACI 209R, "Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures," over 90 percent of the shrinkage occurs during the first year, reaching 98 percent by the end of the first 5 years. Thus, concrete shrinkage is the most significant degradation mechanism that may impact radiation shielding performance of a concrete component during the initial years of storage following concrete fabrication, when the radioactivity of the spent fuel in dry storage is the highest.

The March 18, 2022, supplement for the MAGNASTOR® amendment no. 12 application included an evaluation of the potential for concrete shrinkage. The applicant evaluated the potential effects that radial concrete shrinkage would have on the lid's ability to perform its radiation shielding safety function. The applicant's evaluation calculated an expected radial gap around the edge of the concrete cask lid due to shrinkage. The NRC staff compared the applicant's calculation of the expected radial gap around the edge of the lid (0.02 inches) due to shrinkage to the data regarding maximum concrete shrinkage from NUREG-2214 and found the licensee's calculation of the expected radial gap due to concrete shrinkage to be acceptable for the initial storage term. The staff determined that the proposed technical specification criteria for the concrete in the lid, including the requirement that the concrete mix and batching should meet the purchaser's indicated attributes, as allowed by ASTM C94 for commercial grade ready-mixed concrete, are sufficient to ensure that the applicant's calculation of expected radial shrinkage is a credible estimate of the actual shrinkage behavior that may be expected during the initial storage term.

The applicant also evaluated top surface dose rate profiles associated with conservative hypothetical radial gaps around the edge of the concrete cask lid due to shrinkage. The staff confirmed that the hypothetical radial gap values of 0.04 and 0.08 inches used for evaluating the top surface dose rate profiles are sufficiently conservative since they are well in excess of the expected radial shrinkage value of 0.02 inches that was calculated by the applicant, as discussed above. The NRC staff's review of the applicant's radiation shielding evaluation for calculating the top surface dose rates is documented in the SER section below.

4.4.4 Other Concrete Degradation Mechanisms

4.4.4.1 Storage Cask Lids with Concrete Encased in Steel

Over time, the concrete cask lid may be prone to other degradation mechanisms, in addition to shrinkage, which could potentially have adverse effects on its ability to perform its radiation shielding function. Since there is no structural strength demand for the concrete used in the lid, other degradation mechanisms of potential concern are those that could cause the concrete in the lid to develop flaws, such as voids or cracks, that could potentially cause an increase in radiation dose rates through the concrete in the lid.

Section 3.5.1, “Concrete,” of NUREG-2214 provides a generic evaluation of potential aging degradation mechanisms and associated aging effects for concrete used in storage overpacks. While NUREG-2214 is typically used for addressing aging during renewals, the staff considered the information on concrete degradation mechanisms to assess whether the proposed changes to the technical specification for the concrete in the cask lid could potentially result in increased susceptibility to deterioration that could have adverse effects on the ability of the lid to perform its radiation shielding safety function during the initial storage term.

Given that the concrete in two of the three cask lid designs is encased in carbon steel, there is very little potential for intrusion of significant water, moisture, and dissolved compounds into the concrete in these two designs due to exposure of the lid top surface to weather and debris. The only degradation mechanisms that are potentially credible for non-structural encased concrete used for radiation shielding are shrinkage (addressed in the SER section above), dehydration at high temperature, and delayed ettringite formation (DEF). Dehydration at high temperature could potentially contribute to cracking and may further exacerbate concrete shrinkage at sufficiently high temperatures if the concrete is not adequately fabricated. Considering the limit on the maximum bulk concrete temperature specified in the SAR, and the fact that fuel temperature decreases over time, the staff confirmed that the proposed technical specification criteria for the concrete are sufficient to ensure that the concrete in the cask lid will not be prone to unacceptable cracking or additional shrinkage beyond that already addressed above. DEF is a degradation mechanism characterized by the early-stage conversion of the mineral ettringite to monosulfoaluminate during curing at sufficiently high temperatures (greater than about 158°F), and subsequent reconversion back to ettringite after the concrete hardens. This degradation mechanism may lead to concrete volume expansion and increased internal residual stresses, which could result in concrete cracking and spalling. As addressed in NUREG-2214, DEF of concrete is not considered credible for dry storage casks in outdoor, sheltered, below-grade, and fully encased environments, provided that adequate concrete placement and curing standards, such as those in ACI 349 and ACI 318, are followed. While the proposed technical specification change removes these ACI standards, the staff confirmed that the specification of ASTM C94 for ready-mixed concrete and the additional specification that concrete shall be protected from the environment during curing are sufficient to ensure that DEF is unlikely to cause degradation that results in unacceptable loss of radiation shielding performance during the operating life of the cask.

4.4.4.2 Storage Cask Lid with Exposed Concrete

Over time, the concrete cask lid in CC7 may be prone to other degradation mechanisms, in addition to shrinkage, which could potentially have adverse effects on its ability to perform its radiation shielding function since it is not encased in steel and therefore exposed to the environment. Since there is no structural strength demand for the concrete used in the lid, other degradation mechanisms of potential concern are those that could cause the concrete in the lid to develop flaws, such as voids, cracks, and/or surface damage (e.g., chipping, scaling, spalling) that could potentially cause an increase in radiation dose rates through the concrete in the lid.

Section 3.5.1, “Concrete,” of NUREG-2214 provides a generic evaluation of potential aging degradation mechanisms and associated aging effects for concrete used in storage overpacks. While NUREG-2214 is typically used for evaluating aging during renewal periods, the staff considered the information on concrete degradation mechanisms to assess whether the proposed changes to the technical specification for the concrete in the cask lid could potentially result in increased susceptibility to deterioration that could have adverse effects on the ability of the lid to perform its radiation shielding safety function over the initial storage term.

Since the concrete sides and top of the lid for CC7 are exposed to outdoor air, there is some potential for intrusion of rainwater and dissolved compounds into the concrete lid due to exposure of the concrete lid to weather and debris. Based on consideration of the concrete degradation mechanisms in NUREG-2214, the staff noted that potential degradation mechanisms related to rainwater intrusion for the exposed concrete lid include freeze and thaw, reaction with aggregates, chemical attack, leaching of calcium hydroxide, and salt scaling. Non-moisture-related degradation mechanisms of potential concern are dehydration at high temperature and delayed ettringite formation. Since there is no structural strength requirement for the concrete lid, the staff noted that the above-listed degradation mechanisms are only a concern if they result in the formation of significant voids, cracks, and/or surface damage that could cause an increase in radiation dose rates. The staff also noted that SAR section 10.2.4, “Shielding Tests,” states that the shielding materials of the concrete cask “are designed for long-term use with negligible degradation over time as a result of normal operations.” This SAR section also states that “[c]hipping, spalling, or other defects of the concrete cask surface shall be identified by annual visual inspection, and “[r]epairs to defects larger than approximately one-inch deep or square shall be performed using grout repair materials applied in accordance with the manufacturer’s instructions.” The NRC staff reviewed this information and confirmed that any significant deterioration of the concrete lid that could cause an increase in radiation dose rates would likely be detectable initially though the annual visual exams of the top surface of the lid since these exams would be able to detect surface damage such as chipping, spalling, or scaling. Based on the specification of ASTM C94 for ready-mixed concrete used in the as-built lids, the additional care utilized during fabrication to ensure the concrete was protected from the environment during curing, and consideration of the SAR visual inspection criteria for detecting surface degradation and potential repair, the staff determined that the above-listed degradation mechanisms are unlikely to cause degradation that results in unacceptable loss of radiation shielding performance during the operating life of these five storage casks.

Considering the potential degradation mechanisms, the staff determined that the proposed technical specification for the concrete cask lid is sufficient to ensure that the concrete in the lid will maintain the physical properties needed to adequately perform its radiation shielding safety function.

4.5 Changes to Technical Specifications

The staff reviewed the proposed changes to the technical specifications for the MAGNASTOR® system to accommodate the addition of the LMTC and addition of a new design of concrete cask, CC7. Revisions to the technical specifications in appendix A and B of the amendment application included the new LMTC, BWR 89 fuel basket, BWR damaged fuel basket, and DFC. The staff concludes that the proposed changes to the technical specifications reflect the limitations of the structural evaluations supporting this amendment, and therefore, the staff finds the technical specifications acceptable.

4.6 Evaluation Findings

F4.1 The applicant has met the requirements of 10 CFR 72.124(b). The SSCs that are important to safety of the MAGNASTOR® system with the LMTC and CC7, including the TSC, are designed to provide favorable geometry or permanently fixed neutron-absorbing materials.

F4.2 The applicant has met the requirements of 10 CFR 72.236(b). The SSCs that are important to safety of the LMTC and CC7 are designed to accommodate the combined loads of normal, off-normal, accidents, and natural phenomena events with an adequate margin of safety. Stresses at various locations of the cask under various design loads are determined by analyses. Total stresses for the combined loads of normal, off-normal, accidents, and natural phenomena events are acceptable and are found to be within the limits of applicable codes, standards, and specifications.

F4.3 The applicant has met the requirements of 10 CFR 72.236(c) for maintaining structural design and fabrication of the MAGNASTOR® system with the LMTC and CC7 by including 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.

F4.4 The applicant has met the specific requirements of 10 CFR 72.236(m). In the SAR, NAC considered the design of the spent fuel storage cask for compatibility with the removal of the stored spent fuel from a reactor site, transportation, and ultimate disposition by the Department of Energy.

Based on the statements and representations in the application, as supplemented, the staff concludes that the structural properties of the MAGNASTOR® system with the LMTC and CC7 are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the structural properties provides reasonable assurance that the MAGNASTOR® system with the LMTC and CC7 will allow safe storage of spent nuclear fuel for the certified term of 20 years. This finding is reached on the basis of a review that considered applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

4.7 References

1. ANSI/ANS 57.9, “Design Criteria for an Independent Spent Fuel Storage Installation (Dry Type).”

2. ANSI N14.6, “Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More.”

3. NUREG-0612, “Control of Heavy Loads at Power Plants: Resolution of Generic Technical Activity A-36,” July 1980 (ML070250180).

4. American Society of Civil Engineers Standard 7-93 “Minimum Design Loads for Buildings and Other Structures, 1993.

5. ASTM C94, “Standard Specification for Ready-Mixed Concrete.”

6. ASTM C138, “Standard Test Method for Density (Unit Weight), Yield, and Air Content (Gravimetric) of Concrete.”

7. NUREG-2214, “Managing Aging Processes in Storage (MAPS) Report,” July 2019 (ML19214A111).

8. ACI 209R, “Prediction of Creep, Shrinkage, and Temperature Effects in Concrete Structures.”

9. ANSI/ANS 6.4, “Guidelines on the Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants,” 2006.

Chapter 5 THERMAL EVALUATION

5.0 Introduction

The thermal review of amendment no. 11 for the MAGNASTOR® cask system provides reasonable assurance that the cask components and fuel material temperatures will remain within allowable values under normal, off-normal, and accident conditions. This review includes confirmation that the fuel clad temperatures for fuel assemblies stored in the MAGNASTOR® cask system will be maintained below specified limits throughout the storage period in order to protect the cladding against degradation that could lead to gross ruptures. This portion of the review also confirms that the cask thermal design has been evaluated using acceptable analytical techniques and/or testing methods.

This review was conducted under the regulations described in 10 CFR 72.236, which identify the specific requirements for the regulatory approval, fabrication, and operation of spent fuel storage cask designs. The unique characteristics of the spent fuel to be stored in the MAGNASTOR® cask system are identified, as required by 10 CFR 72.236(a), so that the design-basis and the design criteria that must be provided for the SSCs important to safety can be assessed under the requirements of 10 CFR 72.236(b).

This application was also reviewed to determine whether the MAGNASTOR® design fulfills the acceptance criteria listed in sections 3, 5, and 12 of NUREG-2215, which provide one method of satisfying the relevant regulatory requirements.

The following changes, proposed under amendment no. 11 to the MAGNASTOR® cask system, are applicable to the thermal evaluation:

1. The addition of a new transfer cask known as the lightweight MAGNASTOR® Transfer Cask (LMTC),
2. The addition of a new concrete cask design known as CC7,
3. Increasing the maximum system heat load capacity when using the LMTC and CC7,
4. Adding new loading patterns I, J, and K as provided in technical specification appendix A, table B2-2 for the PWR basket assembly,
5. Adding new loading patterns B, C, and D as provided in technical specification appendix B, table B2-10a for the new BWR 89-assembly basket,
6. Adding new loading patterns A, B, and C in technical specification appendix B, table B2-10b for the new BWR 81-assembly basket,
7. Removing the previously licensed BWR basket configurations since they are bounded by the new 89 and 81 configurations. (Note: the SAR analysis and licensing drawings remain as they partially support the evaluations justifying the approval of the new 89 and 81 configurations),
8. Addition of new BWR fuel basket design permits loading up to 89 undamaged BWR fuel assemblies with increased heat load capacity,
9. Addition of a new BWR damaged fuel basket design with a capacity of up to 81 undamaged BWR fuel assemblies, which includes 12 DFC locations with increased heat load capacity,
10. Adding a new DFC for BWR fuel,
11. Adding new and revised (previously approved) drawings for the LMTC, BWR 89 fuel basket, BWR damaged fuel basket, and DFC,

12. Technical specification, appendix A revisions to include the new LMTC, BWR 89 fuel basket, BWR damaged fuel basket, and DFC; including increased heat loads and loading patterns, and
13. Technical specification, appendix B revisions to include the new LMTC, BWR 89 fuel basket, BWR damaged fuel basket, and DFC; including increased heat loads and loading patterns.

Chapter 1 of the SAR provides an introduction to the MAGNASTOR® system and includes general descriptions of the various components needed to load the MAGNASTOR® system with spent fuel (section 1.3.1, figures 1.3-1 to 1.3-4, and table 1.3-1), including a new TSC, TSC5, which features an 8-inch thick solid stainless-steel closure lid assembly (section 1.3.1.1), the proposed new BWR basket designs for standard and damaged fuel (section 1.3.1.2), a new BWR DFC (section 1.3.1.5), a new concrete cask CC7 (section 1.3.1.3), and a new lightweight MAGNASTOR® Transfer Cask (LMTC[1]) intended for use at facilities with limited crane capacity and for TSCs with high heat loads (section 1.3.1.4). Operational features and general loading of the MAGNASTOR® system for storage are summarized in section 1.3.2. The proposed contents of the MAGNASTOR® system are described in section 1.4 of the SAR. Finally, licensing drawings are provided in section 1.8 of the SAR. Drawing no. 71160-590, Rev. 10, Sheet 2 of 2, provides a depiction of the loaded MAGNASTOR® concrete cask system.

5.1 MAGNASTOR® System Thermal Model

The applicant used a combination of the ANSYS® FEA and ANSYS FLUENT® finite volume computational fluid dynamics (CFD) computer-based programs to evaluate the thermal performance of the MAGNASTOR® spent fuel storage system for the added concrete cask and transfer cask designs as well as for the new fuel baskets and fuel loading configurations.

The evaluation of the designs for the “high heat load” variations of the MAGNASTOR® requested in this amendment, including the LMTC, are provided in a new section of the SAR, as indicated on Page 4.1-4 of the SAR: “Section 4.11 presents the thermal evaluation of the MAGNASTOR® system with heat load over 35.5 kW (up to 42.5 kW) for the PWR system and heat load over 33 kW (up to 42 kW) for the BWR system. Both the standard and damaged fuel PWR and BWR basket configurations are considered. The LMTC is used for evaluation of the transfer operations.”

5.1.1 Thermal Model Development

SAR section 4.11.1 provides a general description of the thermal models developed for “high heat load Configurations” and submitted as part of this application. In line with the applicant’s modeling approach in previous submittals, and as generally described in SAR section 4.4.1, “Thermal Analysis Models,” the applicant developed two-dimensional (2D) axisymmetric models for the TSC and concrete cask (for storage conditions) to generate the boundary conditions for three-dimensional (3D) one-quarter and one-eighth-symmetry models of the TSC with PWR fuel and BWR fuel, respectively. The 2D axisymmetric models were assembled using ANSYS FLUENT®, as described in section 4.11.1.1, and are used to perform steady-state analyses for normal, off-normal, and accident conditions of storage. The TSC temperature profiles derived from these analyses are then applied as boundary conditions on the 3D TSC models for storage

[1] “The LMTC includes a demineralized water-filled neutron shield tank that can be drained for pool loading operations in order to reduce the hook wet weight, then refilled to restore neutron shielding prior to performing canister draining, drying, and closure operations.” [from Page 4.1-2 of the SAR].

conditions, as described in section 4.11.1.2, for the PWR, BWR, and BWR damaged fuel configurations. Similarly, section 4.11.1.3 presents 3D models for transient analyses of the blocked inlets (accident) condition.

The applicant's evaluation of the TSC design for transfer conditions for PWR, BWR, and BWR damaged fuel configurations utilize two analysis models: (1) a 3D ANSYS FLUENT® model including the transfer cask and TSC, described in section 4.11.1.4 of the SAR and (2) a 3D ANSYS FEA model of the loaded TSC as described in section 4.11.1.5 in the SAR.

The ANSYS FLUENT® models are used to perform either steady-state or transient analyses for water or helium backfilled phases during transfer of the TSC. The ANSYS FEA model is then used to perform a transient analysis for vacuum drying conditions in the TSC during loading operations. Note that the thermal models consider a water inlet temperature of 70°F (21°C) and a flowrate of 40 GPM (upflow) for the Annulus Circulating Water Cooling System (ACWS) used during the TSC transfer.

5.1.1.1 Fuel and Fuel Basket Models

In the 2D ANSYS FLUENT® models described above and discussed in section 4.11.1.1 of the SAR, the fuel regions are modeled as homogeneous regions with effective thermal properties, represented by porous media constants simulating the flow resistance due to fuel rods, fuel assembly grids, and fuel basket walls. This approach has been implemented by the applicant in previous applications and has been reviewed and approved for its intended purpose by the staff in amendment no. 0 (ML090350509). These effective thermal properties and the porous media constants used in the fuel region are described in SAR sections 4.4.1.2 for the fuel basket and 4.4.1.3 for the fuel assemblies.

For the 3D ANSYS FLUENT® models described above and discussed in sections 4.11.1.2 through 4.11.1.4 of the SAR, fuel assemblies are also modeled as homogeneous regions with effective thermal properties as described in section 4.4.1.3 of the SAR. The porous media constants applied in these models are described in section 4.8.2 of the SAR. As with the 2D fuel models, this approach has been used in previous applications by the applicant and was reviewed and approved for its intended purpose by the staff.

For the 3D ANSYS FLUENT® models described above and discussed in sections 4.11.1.2 through 4.11.1.4 of the SAR, the basket fuel tube walls with neutron absorber are modeled using effective thermal properties which are determined using the two-dimensional fuel tube wall model as described in section 4.11.1.6 of the SAR.

Further, the fuel tube corners are also modeled using effective thermal properties to account for the gaps between the fuel tube corners. The effective properties for tube corners are determined using the two-dimensional fuel tube corner model as presented in section 4.11.1.7 of the SAR.

Finally, for the 3D ANSYS FEA models of the TSC used for the vacuum drying analyses of both the PWR and BWR configurations, as described above and discussed in section 4.11.1.5 of the SAR, effective thermal properties are used for both the fuel and neutron absorber regions, with the minimum effective thermal conductivity of the "Type 2" neutron absorber, as listed in table 8.3-27 of the SAR, used to determine the effective properties.

The staff reviewed the applicant's description of the MAGNASTOR® storage system thermal models based on the system components, proposed content, and fuel loading zone configurations. Based on the description of the revised thermal models, the staff determined that the application is consistent with guidance provided in NUREG-2215, section 5.5.4, *Analytical Methods, Models, and Calculations*. Therefore, the staff concludes that the descriptions of the thermal models are acceptable, as those descriptions are consistent with NUREG-2215 and satisfy the requirements of 10 CFR 72.236(b), 10 CFR 72.236(f), 10 CFR 72.236(g), and 10 CFR 72.236(h).

5.2 Thermal Evaluation for Normal Conditions of Storage

The applicant describes the analysis for normal storage conditions in section 4.11.2 of the SAR. The applicant uses the 2D and 3D models, described in sections 4.11.1.1 and 4.11.1.2 of the SAR, to determine temperature distributions under long-term normal storage conditions for both PWR and BWR fuel.

5.2.1 Normal Storage Conditions for PWR Fuel

The applicant performed thermal calculations for the three PWR heat load patterns, I, J, and K, shown in figures 4.11-1 through 4.11-3 of the SAR and as provided in technical specification appendix A, table B2-2. The applicant predicted maximum temperatures for fuel cladding, the fuel basket, TSC shell, and concrete (both local and bulk). As indicated in table 4.11.2.1-1 of the SAR, temperatures for these components remain below the allowable limits indicated in the table.

For the bounding fuel loading configuration (heat load pattern I), the maximum average helium temperature in the TSC is 461°F (238°C) and, as described in section 4.11.2.1 of the SAR, is bounded by a maximum average helium temperature of 485°F (252°C).

5.2.2 Normal Storage Conditions for BWR Fuel

The applicant performed thermal calculations for the three BWR heat load patterns, A, B, and C, shown in figures 4.11-4 through 4.11-6 of the SAR for the 89 cell BWR basket and figures 4.11-7 through 4.11-9 of the SAR for the 81 cell BWR basket for damaged fuel. The applicant predicted maximum temperatures for fuel cladding, the fuel basket, TSC shell, and concrete (both local and bulk). As indicated in tables 4.11.2.1-2 and 4.11.2.1-3 of the SAR, temperatures for these components remain below the allowable limits indicated in the table.

The maximum fuel cladding temperature was obtained for heat load pattern B in the BWR 89 basket, as shown in table 4.11.2.1-2 of the SAR and as shown in table 4.11.2.1-3, for heat load pattern A for the BWR 81 damaged fuel basket.

For the BWR bounding configuration (heat load pattern B), the maximum average helium temperature in the TSC is 461°F (238°C). For the BWR damaged fuel bounding configuration (heat load pattern A), the maximum average helium temperature in the TSC is 460°F (237°C).

5.2.3 Maximum Internal Pressure for Normal Conditions of Storage

The applicant calculated the maximum TSC internal pressures for normal storage conditions, using the method documented in section 4.4.4 of the SAR and a bounding helium temperature of 485°F (252°C), which resulted in a maximum TSC internal pressure of

105 psig (723.95 kiloPascal [kPa]) for the PWR configuration and 104 psig (717.05 kPa) for the BWR and BWR damaged fuel configurations.

Therefore, the maximum normal condition pressure for the TSC containing PWR fuel (heat load pattern “I”), BWR fuel (heat load pattern “B”), and BWR damaged fuel (heat load pattern “A”) is bounded by the maximum normal condition TSC design-basis internal pressure of 110 psig (758.42 kPa) system pressure used in chapter 3 of the SAR for normal condition structural evaluations.

5.2.4 Staff Review of Normal Conditions of Storage

The staff reviewed the applicant’s thermal evaluation of the MAGNASTOR® storage system during normal conditions of storage for the addition of the heat load patterns that may be loaded in the TSC for PWR and BWR fuel. Based on the information provided in the application regarding the thermal model and the evaluation of it, as described above, the staff determined that the application is consistent with guidance provided in NUREG-2215, section 5.5.4, *Analytical Methods, Models, and Calculations*. The applicant has demonstrated that the maximum temperatures resulting from an analysis of the proposed new heat load patterns are below the maximum temperatures allowed for each of the materials used in the storage system. As a result, the staff finds that the MAGNASTOR® storage system with the new heat loads proposed by the applicant, meets the requirements of 10 CFR § 72.236(f).

5.3 Thermal Evaluation for Short-Term Operations

5.3.1 Transfer Conditions

The applicant’s approach to the evaluation of transfer conditions is summarized in section 4.11.2.2 of the SAR. The applicant examines four phases of preparation for storage: water phase, vacuum drying phase, cooling/helium phase, and (on-site) transfer phase. The applicant has stated that the ACWS is operated in all phases except for the transfer phase.

As described in section 1.3.2 of the SAR, the relevant steps of the loading and transfer process for the MAGNASTOR® storage system are summarized below as they relate to the four phases of preparation for storage mentioned above. The steps highlighted below are not comprehensive and include only steps relevant to the review of the evaluation of loading and transfer conditions.

The applicant provides additional detail on the loading, closing, and transfer of the TSC in chapter 9 of the SAR “Operating Procedures,” specifically SAR sections 9.1.1 and 9.1.2 for the MTC, SAR sections 9.2.1 and 9.2.2 for the PMTC, and SAR sections 9.3.1 and 9.3.2 for the LMTC. The staff’s review of the impact of each of the phases highlighted below on the thermal performance of the system is provided in subsequent SER sections.

<u>Water Phase</u>:

- Lift the transfer cask over the pool, start the flow of water to the transfer cask annulus, and lower the cask to the bottom of the pool.
- Load the selected spent fuel assemblies and damaged fuel cans (if applicable) into the TSC.
- Install the closure lid assembly.

- Remove the transfer cask from the pool and place it in the cask preparation workstation, or place cask on the in-pool shelf or in the cask loading pit (CLP).
- Decontaminate the transfer cask.
- Lower the TSC water level and weld the closure lid to the TSC shell.
- Complete a weld examination.
- Hydrostatically test the TSC.
- Install and weld the closure ring, followed by a weld examination.
- Drain the remaining pool water from the TSC.

Vacuum Drying Phase:
- Complete vacuum drying of the TSC cavity and verify cavity dryness.

Cooling/Helium Phase:
- Establish a helium backfill.
- Install and weld the inner vent and drain port covers and examine the welds.
- Helium leak test the inner vent and drain port covers.
- Install and weld the outer vent and drain port covers and examine the welds.

Transfer Phase:
- Install the TSC lifting system.
- Install the adapter plate on the concrete cask body.
- Lift and place the transfer cask on the transfer adapter.
- Attach the TSC lifting system to the crane hook and raise the TSC off of the shield doors.
- Open the shield doors.
- Lower the TSC into the concrete cask body (see figure 1.3-1 of the SAR).
- Remove the transfer cask, transfer adapter, and TSC lifting systems.
- Install the lid on the concrete cask body.
- Move the loaded concrete cask to the storage pad.
- Move the concrete cask to its designated location on the storage pad.

5.3.1.1 Water Phase Conditions for PWR Fuel Transfer

Temperatures for the steady-state water phase of PWR fuel transfer are calculated using the 3D transfer cask and TSC ANSYS FLUENT® models described in section 4.11.1.4.1 of the SAR. The applicant provides the maximum fuel cladding temperatures in table 4.11.2.2-1.

The staff reviewed the analysis models used by NAC and determined that they were adequately developed, appropriately representative of the system being modeled, and properly implemented for the water phase of PWR fuel transfer. The staff reviewed the temperature results provided by the applicant and confirmed that they were within the appropriate design limits imposed by the applicant.

5.3.1.2 Water Phase Conditions for BWR/BWR Damaged Fuel Transfer

Temperatures for the steady-state water phase of BWR/BWR damaged fuel transfer are calculated using the 3D transfer cask and TSC ANSYS FLUENT® models described in section 4.11.1.4.2 of the SAR, and the maximum fuel temperatures are reported in table 4.11.2.2-7 and table 4.11.2.2-13 for BWR damaged fuel.

The staff reviewed the analysis models used by NAC and determined that they were adequately developed, appropriately represented the system being modeled and were correctly implemented for the water phase of BWR and BWR damaged fuel transfer. The staff reviewed the temperature results provided by the applicant and confirmed that they were within the appropriate design limits imposed by the applicant.

5.3.1.3 Vacuum Drying Phase for PWR Fuel Transfer

The applicant performed vacuum drying calculations for three heat load patterns (I, J, and K) using the 3D TSC ANSYS model described in section 4.11.5.1 of the SAR. The initial condition for the vacuum drying phase is the TSC backfilled with helium. The maximum temperatures of the fuel and the basket at the end of vacuum drying for these heat load patterns are reported in SAR table 4.11.2.2-3. As shown in this table, heat load pattern K results in the highest fuel temperature at the end of vacuum drying, but the predicted temperature remains below the allowable limit described in the SAR.

The applicant determined, based on maximum fuel temperatures after the first vacuum drying cycle and the temperatures at the end 24-hour cooldown, that the allowable time limits for second vacuum drying (if needed) are 9, 15, and 19 hours for heat loading patterns I, J, and K, respectively (shown in table 4.11.2.2-5).

As reported in SAR tables 4.11.2.2-4 and 4.11.2.2-6, the predicted results during cyclic vacuum drying show that temperature variations exceed the 117°F (65°C) threshold recommended in NUREG-2215 (specifically section 5.4.2, “Material and Design Limits”). As described in SAR section 4.11.2.2.1, the fuel will experience a minimum of four temperature cycles in excess of the recommended temperature threshold mentioned above, from the initial evacuation of the TSC to the transfer of the TSC to the VCC. The applicant has indicated that thermal cycles during system drying operations that exceed 117°F (65°C) will be restricted to no more than 10 cycles for spent fuel with burnup greater than 45 GigaWatt-days per metric ton Uranium (GWd/MTU) (i.e., High Burnup Fuel), as required by technical specification 5.2.c, in appendix A of the CoC. NRC staff determined that, as discussed in chapter 8, “Materials Evaluation” (section 8.9) of this SER, exceeding the temperature variation threshold is acceptable in this instance.

The staff reviewed the analysis models used by NAC and determined that they were adequately developed, appropriately represented the system being modeled and were correctly implemented for the vacuum drying phase for PWR fuel transfer. The staff reviewed the temperature results provided by the applicant and confirmed that they were within the appropriate design limits imposed by the applicant.

5.3.1.4 Vacuum Drying Phase for BWR/BWR Damaged Fuel Transfer

The applicant performed vacuum drying calculations for three heat load patterns (A, B, and C, as shown in table B2-10a for the undamaged 89-assembly basket, and table B2-10b, for the 81-assembly damaged basket) using the 3D TSC ANSYS model described in section 4.11.5.2 for BWR and section 4.11.5.3 of the SAR for BWR damaged fuel. The maximum temperatures of fuel and basket at the end of vacuum drying for these heat load patterns are reported in SAR tables 4.11.2.2-9 and 4.11.2.2-15 for BWR and BWR damaged fuel, respectively. As shown in these tables, heat load pattern B results in the highest fuel temperature at the end of vacuum drying (659°F or 348°C) for BWR fuel and BWR damaged fuel (660°F or 349°C), but the predicted temperatures remain below the allowable limit described in the SAR.

The applicant determined, based on maximum fuel temperatures after vacuum drying and the 24-hour cooldown, that the allowable time for second vacuum drying (if needed) for a BWR basket is 14, 25, and 25 hours for heat loading patterns A, B, and C, respectively (shown in table 4.11.2.2-11).

The applicant further determined, based on maximum fuel temperatures after vacuum drying and the 24-hour cooldown, that the allowable time for second vacuum drying (if needed) for a BWR damaged fuel basket is determined to be 13, 20, and 19 hours for heat loading patterns A, B, and C, respectively (as shown in table 4.11.2.2-17).

As reported by the applicant in SAR tables 4.11.2.2-9 and 4.11.2.2-11 (for BWR fuel) and tables 4.11.2.2-15 and 4.11.2.2-17 (for BWR damaged fuel), the predicted results during cyclic vacuum drying show that temperature variations exceed the 117°F (65°C) threshold recommended in NUREG-2215 (specifically section 5.4.2, “Material and Design Limits”). As described in SAR sections 4.11.2.2.2 and 4.11.2.2.3, the fuel will experience a minimum of four temperature cycles in excess of the recommended temperature threshold mentioned above, from the initial evacuation of the TSC to the transfer of the TSC to the VCC. The applicant has indicated that thermal cycles during system drying operations that exceed 117°F (65°C) will be restricted to no more than 10 cycles for spent fuel with burnup greater than 45 GWd/MTU (i.e., High Burnup Fuel), as required by technical specification 5.2.c, in appendix A of the CoC. NRC staff determined that, as discussed in chapter 8, “Materials Evaluation” (section 8.9) of this SER, exceeding the temperature variation threshold is acceptable in this instance.

The staff reviewed the analysis models used by NAC and determined that they were adequately developed, appropriately represented the system being modeled and were correctly implemented for the vacuum drying phase for BWR fuel. The staff reviewed the temperature results provided by the applicant and confirmed that they were within the appropriate design limits imposed by the applicant.

5.3.1.5 Cooling/Helium Backfill Phase for PWR Fuel

For a 24-hour “cooling phase” (cooldown) following the first vacuum drying stage, as described in section 4.11.2.2.1 of the SAR for PWR fuel, the TSC is backfilled with helium. In order to assess this condition, the applicant performed two transient analyses using the 3D transfer cask and TSC ANSYS FLUENT® model for loading patterns I and K (loading pattern J is bounded by loading pattern K).

The temperature profile applied at the end of vacuum drying, a conservative temperature profile (based on an extended vacuum drying time) that bounds the temperature profiles at the end of the vacuum drying phase (for both heat loading patterns I and K), is used as the initial condition of the transient analysis for the cooldown phase.

For the 24-hour cooldown, the maximum temperatures at the end of the cooldown, for heat loading patterns I and K, are presented in table 4.11.2.2-4. The bounding maximum average helium temperature is 460°F (238°C).

The applicant then applies the maximum fuel temperatures calculated for the 24-hour cooldown to all of the fuel in the TSC in order to determine the duration of the second vacuum drying phase. The second vacuum drying phase is followed by a 12-hour cooldown prior to transfer.

The staff reviewed the analysis models used by NAC and determined that they were adequately developed, appropriately represented the system being modeled and were correctly implemented for the cooling/helium backfill phase for PWR fuel. The staff reviewed the temperature results provided by the applicant and confirmed that they were within the appropriate design limits imposed by the applicant.

5.3.1.6 Cooling/Helium Backfill Phase for BWR Fuel

For a 24-hour "cooling phase" following the first vacuum drying stage, as described in SAR sections 4.11.2.2.2 for BWR fuel and 4.11.2.2.3 for BWR damaged fuel, the TSC is backfilled with helium. In order to assess this condition, the applicant performed two transient analyses using the 3D transfer cask and TSC ANSYS FLUENT® model for loading patterns A and B (loading pattern C is bounded by loading pattern B) as shown in table B2-10a for the undamaged 89-assembly basket, and table B2-10b, for the 81-assembly damaged basket.

The temperature profile applied at the end of vacuum drying, a conservative temperature profile (based on an extended vacuum drying time) that bounds the temperature profiles at the end of the vacuum drying phase (for both heat loading patterns A and B), is used as the initial condition of the transient analysis for the cooldown phase.

For BWR fuel, the maximum fuel temperatures at the end of the 24-hour cooldown, for heat loading patterns A and B, are presented in tables 4.11.2.2-10 and 4.11.2.2-16 for the BWR and BWR damaged fuel baskets, respectively. The bounding maximum average helium temperature during the 24-hour cooling is provided in section 4.11.2.2.2 and 4.11.2.2.3 for BWR fuel and damaged BWR fuel, respectively, while the maximum fuel temperatures are shown in table 4.11.2.2-17 at the end of the 24-hour cooldown, for heat loading patterns A and B, respectively, for both BWR and BWR damaged fuel baskets.

The applicant then applies the maximum fuel temperatures calculated for the 24-hour cooldown to all of the fuel in the TSC in order to determine the duration of the second vacuum drying phase. The second vacuum drying phase is followed by a 12-hour cooldown prior to transfer.

The staff reviewed the analysis models used by NAC and determined that they were adequately developed, appropriately represented the system being modeled and were correctly implemented for the cooling/helium backfill phase for BWR fuel. The staff reviewed the temperature results provided by the applicant and confirmed that they were within the appropriate design limits imposed by the applicant.

5.3.1.7 On-Site Transfer for TSC with PWR Fuel

The applicant's methodology for analysis of the MAGNASTOR® system for onsite transfer is described in section 4.11.2.2.1 of the SAR and summarized below.

Upon completion of vacuum drying (described above), the system is cooled for 12 hours, and the TSC is transferred to the concrete cask within an administrative time limit of 16 hours (see technical specifications, LCO 3.1.1). The applicant performed a transient analysis of this evolution using the 3D transfer cask and TSC ANSYS FLUENT® models for heat loading patterns I and K only, as the loading pattern J is bounded by loading pattern K.

As shown in table 4.11.2.2-6, predicted maximum fuel temperatures for heat loading pattern K at the end of the transfer condition remain below the allowable limit described in the SAR for the administrative time used during transfer of the TSC to complete the operation. The bounding maximum average helium temperature during the 16-hour transfer is reported in section 4.11.2.2.1 of the SAR.

The staff reviewed the analysis models used by NAC and determined that they were adequately developed, appropriately represented the system being modeled and were correctly implemented for the onsite transfer phase for PWR fuel. The staff reviewed the temperature results provided by the applicant and confirmed that they were within the appropriate design limits imposed by the applicant.

5.3.1.8 On-Site Transfer for TSC with BWR Fuel

The applicant uses the same method described for PWR fuel (in SER section 4.3.1.7 above) for onsite transfer of BWR fuel as described in section 4.11.2.2.2 of the SAR and for the BWR damaged fuel basket as described in section 4.11.2.2.3 of the SAR. The applicant's method is also summarized below.

Upon completion of vacuum drying (as described above), the system is cooled for 12 hours, and the TSC is transferred to the concrete cask within an administrative time limit of 22 hours. The applicant performed a transient analysis of this evolution using the 3D transfer cask and TSC ANSYS FLUENT® models for heat loading patterns A and B, as shown in table B2-10a for the undamaged 89-assembly basket, and table B2-10b, for the 81-assembly damaged basket, with air flow in the annulus (loading pattern C is bounded by loading pattern B). This approach applies to TSC loaded with either BWR or BWR damaged fuel baskets.

The predicted maximum fuel temperatures for BWR fuel, for heat loading patterns A, B, and C at the end of the transfer condition are shown in SAR table 4.11.2.2-12. SAR table 4.11.2.2-18 shows the identical results for the BWR damaged fuel basket calculation. These temperatures are below the allowable limit described in the SAR for the 22-hour administrative limit for transfer for both BWR fuel and BWR damaged fuel baskets.

The staff reviewed the analysis models used by NAC and determined that they were adequately developed, appropriately represented the system being modeled and were correctly implemented for the onsite transfer phase for BWR fuel. The staff reviewed the temperature results for transfer of the TSC to the concrete cask provided by the applicant and confirmed that the average helium temperatures reported were essentially equivalent to, or less than, those reported for normal conditions. Further, the canister pressures that would be associated with those reported temperatures would fall within the appropriate design limits imposed by the applicant.

5.3.2 Staff Review of Transfer Conditions for PWR, BWR, and BWR Damaged Fuel Baskets

The staff reviewed the applicant's thermal evaluation of on-site transfer operations for MAGNASTOR® storage system, including the vacuum drying evolution. Based on the information provided in the application regarding the thermal analysis model, evaluation, and reported temperatures and associated pressures, the staff determined that the application is consistent with guidance provided in NUREG-2215, section 5.5.4, *Analytical Methods, Models, and Calculations* and, therefore, meets the requirements of 10 CFR § 72.236(f).

5.4 Off-Normal and Accident Events

5.4.1 Off-Normal Events

The applicant describes their evaluation of off-normal storage events, including extreme ambient temperatures (106°F and -40°F) and partial inlet vent blockage conditions, in section 4.11.3 of the SAR.

5.4.1.1 Off-Normal Ambient Temperatures

The variation in ambient temperatures for the off-normal event evaluation required a change to the boundary condition temperatures for the model evaluations. For the partial blocked air inlet condition, the air inlet model is modified to permit air flow through only half of the inlet area of the concrete overpack. The applicant reports the temperatures for the fuel cladding, fuel basket, TSC shell, and concrete for off-normal storage conditions in SAR section 4.11.3.1.

The applicant used heat load pattern "I" for the PWR fuel basket to perform these analyses, since it is the bounding heat load pattern. The applicant reported the following peak cladding temperatures: 775°F (413°C) for the 106°F ambient, 620°F (327°C) for the -40°F ambient, and 752°F (400°C) for the partially blocked air vents condition.

For BWR fuel the heat load pattern "B" is used, while for BWR damaged fuel the heat load pattern "A" is used, as those are the bounding heat load patterns for the BWR configurations. The applicant reported the following peak cladding temperatures: 753°F (400.5°C) for the 106°F ambient, 604°F 318°C) for the -40°F ambient, and 731°F (388°C) for the partially blocked air vents condition for the BWR fuel basket and 785°F (418°C) for the 106°F ambient, 618°F (325.5°C) for the -40°F ambient, and 742°F (394°C) for the partially blocked air vents condition for the BWR damaged fuel basket. The maximum average helium temperature is 504°F (262°C) for BWR and 505°F (263°C) for BWR damaged fuel for all off-normal conditions.

All component temperatures reported by the applicant in SAR section 4.11.3.1 for all fuel types and configurations remain below the allowable temperatures described in the SAR for off-normal conditions.

The staff reviewed the analysis models used by NAC and determined they were adequately developed, appropriately represented the system being modeled and were correctly implemented for off-normal conditions. The staff reviewed the temperature results provided by the applicant and confirmed that they were within the appropriate design limits imposed by the applicant.

5.4.1.2 Maximum Internal Pressure for Off-Normal Events

The applicant calculates the maximum TSC internal pressure for the off-normal events using the evaluation method documented in SAR section 4.5.2. A bounding average helium temperature of 521°F (272°C) is used and results in maximum TSC internal pressures of 119 psig (820 KPa) for the PWR and PWR damaged fuel configurations and 112 psig (772 KPa) for the BWR and BWR damaged fuel configurations.

The pressures calculated are less than the 130 psig (896 KPa) system pressure used in chapter 3 of the SAR (section 3.6.1) for off-normal operating event structural evaluations.

5.4.1.3 Staff Review of Off-Normal Events

The staff reviewed the applicant's thermal evaluation during off-normal events. Based on the information provided in the application regarding the thermal analysis model, evaluation, and temperatures, the staff determined that the application is consistent with guidance provided in NUREG-2215, sections 5.5.1, *Decay Heat Removal Systems*, and 5.5.4, *Analytical Methods, Models, and Calculations*, and, therefore, meets the requirements of 10 CFR § 72.236(f).

5.4.2 Accident Events

Accident events evaluated by the applicant are events with a low probability of occurring during the licensed storage period of a MAGNASTOR® system but that must be evaluated in accordance with 10 CFR §72.122(b). Three postulated thermal accident events were evaluated by the applicant including: the maximum anticipated ambient temperature, a fire accident, and full blockage of the air inlet vents on a loaded MAGNASTOR® concrete cask. The applicant's analyses of the accident events are described in section 4.11.4 of the SAR and are summarized below.

5.4.2.1 Maximum Anticipated Temperatures

As described in section 4.11.4.1 of the SAR, the applicant applied an accident condition ambient temperature of 133°F (56°C) to the 2D and 3D concrete cask and TSC models described in sections 4.11.1.1 and 4.11.1.2 of the SAR. The applicant reports the temperatures for the fuel cladding, fuel basket, TSC shell, and concrete for accident condition temperatures mentioned above. For the high heat PWR configurations the analysis was performed using bounding heat load pattern "I" with a maximum reported fuel cladding temperature of 803°F (428°C). A maximum concrete temperature of 272°F (133°C) was reported. The average helium temperature was reported to be 515°F (268°C) in the TSC.

For the high heat BWR and BWR damaged fuel configurations, the analyses were performed using the bounding heat load patterns B and A, respectively, with a maximum fuel cladding temperature of 774°F (412°C) and 805°F (429°C) reported for the BWR and BWR damaged fuel configurations, respectively. The maximum concrete temperature of 275°F (135°C) and an average helium temperature in the TSC of 527°F (275°C) were reported for both BWR configurations. All the temperatures determined by the applicant for the accident condition described above were within acceptable limits, as recorded in section 4.11.4.1 of the SAR.

5.4.2.2 Fire Accident

The applicant describes the postulated fire accident scenario for the MAGNASTOR® system in section 4.6.2 of the SAR as follows:

> "A fire may be caused by flammable material or by a transport vehicle. While it is possible that a transport vehicle could cause a fire while transferring a loaded storage cask at the ISFSI [independent spent fuel storage facility], this fire will be confined to the vehicle and will be rapidly extinguished by the persons performing the transfer operations or by the site fire crew. Fuel in the fuel tanks of the concrete cask transport vehicle and/or prime mover (maximum 50 gallons) is the

only flammable liquid that could be near a concrete cask, and potentially at, or above, the elevation of the surface on which the cask is supported. The fuel carried by other onsite vehicles or by other equipment used for ISFSI operations and maintenance, such as air compressors or electrical generators, is considered not to be within the proximity of a loaded cask on the ISFSI pad. Site-specific analysis of fire hazards will evaluate the specific equipment used at the ISFSI and determine any additional controls required."

The applicant's approach to the fire accident analysis for the high heat load PWR and BWR configurations is provided in section 4.11.4.2 of the SAR, in which the applicant describes a transient thermal analysis completed for PWR fuel in the concrete cask system with a design-basis heat load of 35.5 kW which was exposed to a 1,475°F (800°C) fire for 8 minutes resulting in a maximum fuel temperature increase of 3°F (~1.7°C).

The applicant determined the maximum fuel clad temperatures for the fire accident by adding 3°F to the normal condition temperature results presented in tables 4.11.2.1-1 through 4.11.2.1 3 for the PWR, BWR and BWR damaged fuel configurations, respectively. The applicant reported the maximum fuel cladding temperatures for the PWR, BWR, and BWR damaged fuel configurations as: 748°F (398°C), 711°F (377°C), and 722°F (383°C), respectively.

The applicant's conclusion, based on the transient analysis in section 4.6.2, was that the limited duration of the fire, the large thermal capacitance of the concrete cask, and the minimal thermal conductivity, limit the local region where the concrete temperatures exceed 300°F (149°C) to less than 10 inches above the top surface of the air inlets and, therefore, that the concrete cask is not adversely affected both during and after the fire accident condition.

The staff reviewed the applicant's approach and found it reasonable and, therefore, agrees with the applicant's conclusions regarding the effects of a fire exposure on the concrete surface of the cask.

5.4.2.3 Full Blockage of Concrete Cask Air Inlets

The applicant's approach employs the method described in SAR section 4.6.3 to complete a transient analysis of the full blockage of all the air inlets for the high heat load PWR and BWR configurations which is discussed in section 4.11.4.3 of the SAR. The applicant's evaluation was done at the normal storage condition ambient temperature of 76°F (24°C). The applicant utilized the 3D concrete cask and TSC models described in sections 4.11.1.2.1, using the bounding case of loading pattern "I" for PWR fuel. The 3D concrete cask and TSC models described in sections 4.11.1.3.2 and 4.11.1.3.3 were used for the evaluation of the BWR and BWR damaged fuel baskets, respectively, for a bounding case with bounding initial temperatures (loading pattern "B" for BWR and loading pattern "A" for BWR damaged fuel).

The applicant demonstrated that, following 60 hours of the vent blockage event, the maximum fuel temperature and bulk concrete temperature remain within the allowable accident temperature limits of 1058°F (570 °C) for fuel cladding and 350°F (177 °C) for concrete. The applicant reports that the maximum fuel temperature, bulk concrete temperature and average helium temperature in the TSC are 992°F (533°C), 253°F (123°C), and 726°F (385.5°C), respectively, for PWR fuel and 937°F (503°C), 249°F (120.5°C), and 724°F (384°C), respectively for BWR and 950°F (510°C), 237°F (114°C), and 697°F (369°C), respectively, for BWR damaged fuel.

The applicant's conclusion, for both the PWR and BWR configurations, was that there would be no adverse consequences due to this accident, provided that debris was cleared from at least two air inlets within 60 hours of the blockage, based on the steady-state evaluation of the half blocked air inlet condition in section 4.11.3 of the SAR, which is consistent with the technical specification A 3.1.2, "STORAGE CASK Heat Removal System" requirements.

5.4.2.4 Maximum TSC Internal Pressure for Accident Events

As described in section 4.11.4.4 of the SAR, the applicant calculates the maximum TSC internal pressure for accident events using the evaluation method documented in section 4.6.4 of the SAR. A bounding average helium temperature of 737°F (392°C) is used and results in maximum TSC internal pressures of 226 psig (1.55 MPa) for the PWR configuration and 161 psig (1.11 MPa) for the BWR and BWR damaged fuel configurations.

The pressures calculated are less than the 250 psig (1.72 MPa) system pressure used in chapter 3 (section 3.7.1) of the SAR for the storage accident condition structural evaluations.

5.4.2.5 Staff Review of Accident Events

The staff reviewed the applicant's thermal evaluation during accident events. Based on the information provided in the application regarding the thermal analysis model, evaluation, and temperatures, the staff determined that the application is consistent with guidance provided in NUREG-2215, section 5.5.4, *Analytical Methods, Models, and Calculations* and, therefore, meets the requirements of 10 CFR § 72.236(f).

5.5 Staff Review of Thermal Analysis Models and Confirmatory Analyses

5.5.1 Staff Review of Thermal Analysis Models

The staff completed an audit review of two of the applicant's thermal models: one, for the concrete cask and PWR canister thermal evaluation for high heat loads and the other, the transfer cask transient thermal analyses for BWR preferential loading, as described in section 4.11.1 of the SAR and the associated calculation packages. The staff checked the code input in the calculation packages submitted and confirmed that the proper material properties and boundary conditions were used. The staff verified that the applicant's selected code models and assumptions were adequate for the flow and heat transfer characteristics prevailing in the MAGNASTOR® geometry for the analyzed conditions.

Engineering drawings were also consulted to verify that system geometry and dimensions were adequately translated to the thermal analysis models. The material properties presented in the SAR were reviewed to verify that they were appropriately referenced and applied. In addition, the staff performed appropriate sensitivity analysis calculations to verify that applicant's predicted results provide bounding predictions for all conditions analyzed in the application.

The staff found that the applicant's analyses and predicted results were generally acceptable for the configurations considered in the thermal models. The staff reviewed selected ANSYS FLUENT® models submitted by the applicant and provided two specific questions in a request for additional information (RAI) to the applicant (ML21047A209). The applicant provided a response to these questions in its supplement dated August 23, 2021 (ML21242A052).

The first RAI question related specifically to the transient analysis of the TSC conducted by the applicant during the helium cooldown phase. The temperatures reported by the applicant indicated that boiling could occur in the annular gap between the TSC and the transfer cask; however, boiling (2 Phase flow) was not modeled by the applicant.

The applicant performed a RELAP5 analysis (documented in NAC calculation package no. 71160-3056 Rev. 0) which indicated that the loss of cooling due to the steam condition in the annulus was limited to approximately 4 minutes at the beginning of a 12-hour cooling down period and therefore, has a negligible effect on the system temperatures. The staff found the applicant’s response in this case acceptable.

The second RAI question related to application of heat transfer coefficients from the 2D axisymmetric model of the TSC (for storage conditions) to the 3D quarter and eighth-symmetry models of the TSC with PWR and BWR fuel, respectively. The staff identified an apparent discrepancy between the total heat distribution between the 2D and 3D models.

The applicant revised their calculation package no. 71160-3085 for the PWR configuration to ensure that the heat transfer rates between the 2D and 3D models agreed for thermal evaluations of the TSC surfaces. For the BWR configuration, the applicant revised calculation package nos. 71160-3060 and 71160-3071 to include a sensitivity analysis that indicated that the fuel temperatures calculated using the existing temperature profile boundary conditions were conservative and, therefore, no revision to the SAR was required for the BWR configuration.

In their RAI response, the applicant also noted that SAR section 4.11 was updated to 15×15, 16×16, and 17×17 PWR fuel types for “high heat load configurations” and the thermal analyses were revised for the updated effective thermal properties for the fuel. The thermal results were also updated in the SAR as the applicant reported that the PWR 16×16 fuel assemblies were discovered to be the limiting fuel assembly for the MAGNASTOR® design.

This change by the applicant was not a direct result of the response to NRC staff RAIs; therefore, the staff investigated the source of the change and discovered that the changes to the models were made in response to a 10 CFR 72.242, “Reportable Licensing Basis Thermal Evaluation Deficiency,” which the applicant reported to NRC via a letter dated March 4, 2021 (ML21070A324). While this did change the “limiting” fuel for the PWR basket, and potentially increased reported fuel temperatures, the changes did not impact previous conclusions as to the acceptability of the thermal performance of the MAGNASTOR® storage system.

5.5.1.1 Review of Independent Report on the Performance Limits of the MAGNASTOR® Storage System

A study of the MAGNASTOR® storage system was conducted recently by the Pacific Northwest National Laboratory (PNNL) for the Department of Energy (DOE) to examine the thermal performance “envelope” of the MAGNASTOR® system. Given the applicant’s request for high heat load contents for the MAGNASTOR® system in this amendment, the staff considered the review of an independent technical analysis of the performance envelope of the MAGNASTOR® system as useful information to inform the review of the current amendment request and has treated it as such for this review. This report does not form the basis of any of the staff’s conclusions within this SER but is considered simply a single “data point” in the overall consideration of the performance this system.

The staff reviewed the report PNNL-28864, “Thermal Analysis of High Decay Heat Loading Strategies in the MAGNASTOR® System,” dated July 26, 2019, which was prepared for the DOE by PNNL. This report examined six high decay heat “zone” loading configurations in the MAGNASTOR® storage system for PWR fuel, as described in Rev. 1 of the MAGNASTOR® FSAR (ML21147A112) and compared them to a conservative base heat loading case provided by Duke Energy for an actual system loaded at the Catawba Nuclear Station in 2014. PNNL examined the 37-assembly fuel basket to determine the maximum decay heat capacity of the system while remaining below a peak clad temperature (PCT) of 400°C for the fuel.

PNNL found that a maximum single cell heat load of 2,860 Watts could be obtained for two assemblies, with a total decay heat loading of 33.5 kW. Using another “zone” approach, PNNL found that with a maximum single cell heat load of 1,416 W for 12 cells, with a total decay heat loading of 41.9 kW, which was found to be the maximum for the MAGNASTOR® system in the PNNL study.

While the PNNL study did not take into account the specific design changes proposed by the applicant in the current MAGNASTOR® application, the results from the PNNL study indicate that the results of the analysis provided by the applicant for the MAGNASTOR® system for a maximum single cell heat load of 3,250 kW for four assemblies, with a total decay heat loading of 42 kW (for preferential load pattern 37P-J) and a uniform single cell heat load of 1,149 Watts to yield a maximum total decay heat loading of 42.5 kW (uniform load pattern 37P-I) are comparable to the independent prediction of the performance limits of the MAGNASTOR® system for the 37-assembly PWR basket provided in the PNNL report.

5.5.3 Confirmatory Analysis

An independent confirmatory analysis was completed by PNNL using the COBRA-SFS thermal analysis code and applying what the applicant determined was the limiting heat load pattern (pattern “I” for PWR fuel), which was chosen because the high heat load per assembly in each cell would reduce heat transfer from the center assemblies compared to the more targeted high heat load assemblies of loading patterns “J” and “K”. In general, the applicant’s model results were within the expected range for the systems and heat loads being analyzed.

An existing COBRA-SFS model of the MAGNASTOR® system was used as a confirmatory check on the applicant’s results. Simulations were run with the limiting loading pattern “I” for both normal conditions and at the extreme heat (accident) condition, with the results presented in the table below. The initial COBRA-SFS for normal conditions results showed a PCT of 776.1°F (413.4°C) which was higher than the applicant’s result of 745.0°F (396°C) and higher than the fuel cladding temperature limit of 752°F (400°C). For the extreme heat condition there was a similar difference between the COBRA-SFS analysis and the applicant’s models, but that condition falls under the accident condition PCT limit of 1058°F (570°C) and with greater than 150°F of margin remaining, therefore, that case was not investigated in detail.

PNNL determined that the initial COBRA-SFS models for normal conditions used a canister wall emissivity value of 0.36, which was a value used by the applicant in prior applications. Using the applicant’s canister side wall emissivity value of 0.5, the COBRA-SFS confirmatory model resulted in a PCT of 744.9°F (396°C). This result is under the normal conditions limit and is within a degree of the applicant’s results. The confirmatory analysis results provide confidence that, under similar input conditions, the COBRA-SFS confirmatory model and the applicant’s models, as presented in the SAR, are behaving similarly. The results are summarized in the table below.

Case	NAC Result (°F)	COBRA-SFS initial Result (0.36 wall emissivity) (°F)	COBRA-SFS Result (0.5 wall emissivity) (°F)
Loading Pattern "I" Normal Conditions	745.0	776.1	744.9
Loading Pattern "I" Accident Extreme Heat (130°F Ambient)	796.0	832.3	N/A

The staff explored whether a 0.5 wall emissivity, as specified in the application, was reasonable and appropriate for this canister. The TSC is made of stainless-steel, which can have a wide variation in emissivity depending on surface conditions. The 0.36 value used previously by the applicant is considered conservative for an industrial application; however, it is possible that, over time, the emissivity of the material may increase. The staff requested a source from the applicant that would support the assertion that the emissivity of the TSC surface was at least 0.5 and the applicant supplied a report[2] that provided the appropriate demonstration of measured surface emissivity for the TSC. The staff reviewed the report and were satisfied with the test procedures and results provided in the report. Therefore, the staff accepts the applicant's proposed use of 0.5 for the TSC wall emissivity, as use of this value in the confirmatory models done by PNNL produce similar temperatures to the applicant's analysis models for the MAGNASTOR® system.

5.6 References

1. Jensen B. J., D. J. Richmond., "Thermal Analysis of High Decay Heat Loading Strategies in the MAGNASTOR® System" Pacific Northwest National Laboratory, PNNL-28864, July 2019 (ML22028A104).

5.7 Evaluation Findings

F5.1 SSCs important to safety are described in sufficient detail in the SAR to enable an evaluation of their thermal effectiveness in accordance with 10 CFR 72.236(f) and 10 CFR 72.236(h). Storage container SSCs important to safety remain within their operating temperature ranges in accordance with 10 CFR 72.236(a) and 10 CFR 72.236(b).

F5.2 The MAGNASTOR® storage system is designed with a heat-removal capability, verifiably and reliably consistent with its importance to safety. The storage container is designed to provide adequate heat removal capacity without active cooling systems in accordance with 10 CFR 72.236(f).

F5.3 The spent fuel cladding is protected against degradation leading to gross ruptures under normal conditions by maintaining the cladding temperature for 40 years below 752°F (400°C) in an inert helium environment. Protection of the cladding against degradation is expected to allow ready retrieval of the spent fuel for further processing or disposal in accordance with 10 CFR 72.236(g), 10 CFR 72.236(l), and 10 CFR 72.236(m).

F5.4 The spent fuel cladding is protected against degradation leading to gross ruptures under off-normal and accident conditions by maintaining the cladding temperature below

[2] Reference #42 from chapter 8 of the SAR: *Emissivity Report Form, Petersen Incorporated, Procedure Number PSP-100189-02, Revision* 2, February 5, 2018.

1058°F (570°C) in a helium environment. Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal in accordance with 10 CFR 72.236(g), 10 CFR 72.236(l), and 10 CFR 72.236(m).

5.8 Conclusions

The staff concludes that the thermal design of the MAGNASTOR® storage system complies with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the MAGNASTOR® will allow safe storage of spent fuel for the current duration of the licensed (certified) life for this design. This conclusion is reached on the basis of a review that considered applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

Chapter 6 SHIELDING EVALUATION

The objective of this evaluation is to determine whether the shielding design of the MAGNASTOR® dry cask spent fuel storage system with the requested amendment no. 11 will meet the regulatory requirements. The review seeks to ensure that the shielding design is reasonably capable of meeting the operational dose requirements of 10 CFR 72.104 and 10 CFR 72.106 in accordance with 10 CFR 72.236(d).

The staff reviewed the changes requested in the application. The staff's review includes the radiation source term determination and the radiation shielding design for all credible normal conditions and off-normal and accident events during loading, handling, on-site transfer, storage, and retrieval. This review also includes verification of computer modeling of the cask system for shielding analyses of the TSC with a shielded closure.

Overall, the staff reviewed whether the MAGNASTOR® dry cask spent fuel storage system with the requested amendments meets the radiation protection requirements set forth in 10 CFR Part 72 and 10 CFR Part 20, and whether the design and operation of the MAGNASTOR® storage system follows the "As Low as Reasonably Achievable" (ALARA) principle.

6.1 Shielding Design Description

6.1.1 Design Criteria

The MAGNASTOR® system is a spent fuel dry storage system designed by NAC International (NAC) and is previously certified by the U.S. NRC under docket no. 72-1031. The applicant submitted an application on March 24, 2020, to the NRC for amendment no. 11 to the CoC with several new loading patterns for the PWR and BWR fuels and other proposed changes. Specifically, the proposed changes to the system design that affect the shielding evaluation include:

1. A new transfer cask known as the lightweight MAGNASTOR® Transfer Cask (LMTC).
2. A new concrete cask design known as CC7.
3. New loading patterns I, J, and K for the PWR basket assembly.
4. Added new loading patterns A, B, and C for the new undamaged BWR 89-assembly fuel basket.
5. Added new loading patterns A, B, and C for the new damaged BWR 81-assembly fuel basket.
6. Added a new BWR fuel basket design which permits loading up to 89 undamaged BWR fuel assemblies with increased heat load capacity.
7. Added a new BWR damaged fuel basket design with a capacity of up to 81 undamaged BWR fuel assemblies, which includes 12 DFC locations with increased heat load capacity. Added a new DFC for BWR fuel.
8. Added a Non-fuel Hardware Component – Partial Length Shield Assemblies (PLSAs).

6.1.2 Design Features

The staff reviewed the shielding design of the MAGNASTOR® system and the analysis used to establish bounding radiation dose rates for the safe storage of up to 37 PWR fuel assemblies, including 4 DFC locations, up to 89 undamaged BWR fuel assemblies in the BWR basket, and up to 81 BWR fuel assemblies, including 12 DFC locations, in the BWR 81-assembly basket. Each DFC may contain an undamaged PWR or BWR fuel assembly or damaged fuel, which may be a damaged fuel assembly or fuel debris equivalent to an undamaged fuel assembly. Undamaged PWR or BWR fuel assemblies may be placed directly in the locations that are designed for damaged fuel in DFCs.

6.1.2.1 Lightweight MAGNASTOR® Transfer Cask and Concrete Cask Design

The MAGNASTOR® system is designed with four transfer casks and seven concrete cask configurations. The LMTC is detailed in section 5.12 of the SAR. The new concrete cask, CC7, is similar to previous concrete casks however it is designed with an augmented shielding, variable height configuration with a 3-inch thick liner, and a modified lid/outlet designed for high heat payloads (analysis in section 5.12 of the SAR for PWR and 5.14 of the SAR for BWR).

The LMTC is a shielded lifting device designed to hold the canister during loading, transfer, and unloading operations. The spent fuel will be stored in nine different preferential heat load configurations. The LMTC is designed with a variable thickness lead and water shield. The LMTC neutron shield is a water tank that can be emptied to decrease weight or filled to provide shielding. The LMTC uses lead for radial gamma shielding. The lead shield of the LMTC ranges in thickness from 2.5 to 4.0 inches, depending on crane lift capacity. LMTC dimensions used in the model are shown in figure 5.12.2-3 of the SAR and reflect dimensions on drawing nos. 71160-L257 and 71160-L258 and show the transfer cask in the maximum lead configuration (4.0 inches of radial gamma shielding). For conservatism, the expansion tank is not modeled in the applicant's shielding models. There are multiple combinations of flooded (W) and empty (D) conditions in the canister and neutron shield regions. Based on the operations for the LMTC, the possible combinations are W/W, W/D, and D/W, however, there is never a D/D combination. For determining maximum dose rates, the applicant states that the only relevant conditions are W/D and D/W. The W/W condition will always be bounded by W/D and D/W, but one case is included for comparison. The staff agrees with the applicant's conclusion that the W/D and D/W configurations are bounding, as the W/W configuration includes significantly more water for shielding.

6.1.2.2 PWR Shielding Evaluations for Load Patterns I, J, and K with CC3/CC7 and LMTC

The applicant proposed new loading patterns I, J, and K for PWR fuel assemblies. The method, which the NRC previously approved in amendment no. 3 (ML13207A245), from section 5.9 of the SAR was used in this amendment to evaluate PWR fuel, using a modified eight-zone pattern. The minimum cool time is 4 years for Pattern I and 2 years for Patterns J and K consistent with the primary intent of continued operations and decommissioning use, respectively. Included in the evaluations is an increase in burnup to 70,000 MWd/MTU versus the 62,500 MWd/MTU evaluated previously. Evaluations are performed for the contents within a CC3 or CC7. Like the CC3, the CC7 cask design contains a 3-inch liner, versus the 1.75-inch liner in the base cask design. A revised lid/top cask section design is implemented within the CC7 design. The revised lid/top cask section incorporates the outlet vent structure and has a variable thickness concrete section. Only the minimum lid thickness section is evaluated for the CC7 design.

6.1.2.3 Loading Patterns A, B, and C for the new BWR 89-Assembly Fuel Basket

The applicant proposed new loading patterns A, B, and C for BWR fuel assemblies. The method, which the NRC previously approved in amendment no. 3, from section 5.9 of the SAR was used in this amendment to evaluate BWR fuel, using a modified nine-zone pattern. The minimum cool time is 4 years for Pattern A and 2 years for Patterns B and C consistent with the primary intent of continued operations and decommissioning use, respectively.

There are three unique patterns summarized in section 5.14.4 of the SAR. The schematic of Nine-Zone 89 Assembly BWR Fuel Preferential Loading Pattern is shown in figure 5.14.4-1 of the SAR. The total cask heat load is pattern specific. The analyzed Pattern A, with 47.437 kW, was conservatively modeled with significantly more heat load than the requested 42 kW.

6.1.2.4 Loading Patterns A, B, and C for the new BWR 81-Assembly Fuel Basket

The applicant proposed new loading patterns for A, B, and C for damaged BWR fuel assemblies. The method, which has been previously approved, from section 5.9 of the SAR was used in this amendment to evaluate damaged BWR fuel, using a modified nine-zone pattern. The minimum cool time is 4 years for Pattern I and 2 years for Patterns J and K consistent with the primary intent of continued operations and decommissioning use, respectively.

There are three unique patterns summarized in section 5.14.5 of the SAR. The schematic of a nine-zone 81-Assembly BWR Fuel Preferential Loading Pattern is shown in figure 5.14.5-1 of the SAR. The total cask heat load is pattern specific. The analyzed Pattern A, with 47.385 kW, was conservatively modeled with significantly more heat load than the requested 42 kW.

6.1.2.5 New BWR Fuel Basket Design for Loading up to 89 Undamaged BWR Fuel Assemblies

Section 5.1 of the SAR contains BWR results and the supporting evaluations obtained from an 87-assembly configuration where two peripheral storage loading locations were left open. These locations were located below the vent and drain ports. The vent and drain ports were relocated to allow loading of these two locations in the updated 89-assembly basket configuration. As part of the design change, the port size was reduced from a 2-inch line to a 1-inch line. This change reduces the radiation streaming potential (reduces cross sectional area to ~1/4 of the original area). The applicant states that removing the two assemblies allowed for a view factor into the next inner assemblies with additional shielding limited to the thin-walled basket tube. Bottom dose rates are maximum at the basket center and would therefore not be impacted.

6.1.2.6 New BWR Fuel Basket Design for Loading up to 81 BWR Fuel Assemblies

The BWR damaged fuel basket contains 81 assemblies, with 12 basket locations designated for damaged fuel. The shorter canister also has an 8-inch lid, rather than a 9-inch-thick lid for TSCs 1 through 4, to accommodate the longer damaged fuel basket while conserving overall canister length. Based on a matrix of packing fractions from 30% to 75%, the mass and volume of damaged fuel in the top nozzle, plenum, and active fuel regions is calculated for the 12 damaged fuel locations in the basket. The active fuel region is divided into two axial zones, with the lower region containing only damaged fuel and the upper region containing homogenized grid material.

6.1.2.7 New Damaged Fuel Can for BWR fuel

Damaged fuel is required to be packaged in a DFC. The parameters used to construct the DFC model are shown in table 4.4-1 of calculation package no. 71160-5031, Rev.0, and based on the dimensions taken from drawing no. 71160-602, Rev. 1. The DFC is fabricated using stainless-steel. The bottom of the DFC is modeled as a 5/8-inch thick monolith with cutouts for the drain holes and corner “foot” voids.

The openings in the damaged fuel basket are constructed by assembling 33 undamaged fuel tubes and four damaged fuel corner weldments to create 37 fuel assembly locations.

The applicant stated that for the damaged fuel cases, the DFC is modeled, and the basket and canister are revised with the modified geometries of the components for damaged fuel. The damaged fuel assemblies are modeled by considering the fuel material collecting toward the bottom of the damaged fuel cans in various packing fractions, which is the same method as used in previously approved MAGNASTOR® analyses in amendment no. 3.

6.2 Radiation Source Definition

Source terms for the various vendor-supplied fuel types were generated by the applicant using SAS2H code sequence of the SCALE 4.4 package with the 44-group ENDF/B-V cross-section libraries. SAS2H includes an XSDRNPM neutronics model of the fuel assembly and the ORIGEN-S code for fuel depletion and source term calculations. Source terms are generated by the applicant for both UO_2 fuel and fuel assembly hardware.

Reference documentation in FSAR Revision 8 (ML17038A506) indicated that the combination of the SCALE 4.4 SAS2H sequence and the 44-group ENDF/B-V cross-section library is applicable to light-water reactor fuel assembly source term generation for high burnup fuel. According to the applicant, due to the limited experimental PWR and BWR data available for the SAS2H sequence as applied to high burnup (> 45,000 MWd/MTU) fuel assemblies, a 5% decrement in heat load is applied at the high burnup fuel levels. The heat load decrement implies an extension in minimum cool time required for high burnup fuel and provides additional margin to account for any uncertainties in the source generation method. For this amendment, the evaluations for an increase in burnup to 70,000 MWd/MTU. versus the 62,500 MWd/MTU evaluated previously is included by the applicant. In section 5.2, the applicant stated that NUREG/CR-7012 contains the summary of various NUREGs that document publicly available comparisons of experimental to code generated isotope compositions based on the TRITON sequence of SCALE. Burnups included in the NUREG are very high burnups and cover a range of 8,000 to 79,000 MWd/MTU. The NUREG compares isotopes relevant to burnup credit, radiation protection and heat generation, and waste management. The comparison relies on the TRITON with NITAWL rather than the newer CENTRM sequence. Beyond the transport solution, which used NEWT in TRITON for a 2D solution rather than XSDRNPM in SAS2H for a 1-D solution, the NUREG and MAGNASTOR® analysis methods are very similar and are not expected to show divergent results in the analysis trend versus burnup. The conclusion in NUREG/CR-7012 is that there is no code bias trend of the depletion-generated isotopics and sources versus burnup level. In particular, there is no significant trend for very high burnup fuels.

While absolute differences between SAS2H and TRITON are expected due to neutron transport method differences, trending is not expected for SAS2H. This conclusion is confirmed by reevaluating cases from NUREG/CR-6968, NUREG/CR-6969 and NUREG/CR-7013 using SAS2H. SAS2H modeled cases go up to 70,000 MWd/MTU. Similarly, to the TRITON cases in NUREG/CR-7012, the SAS2H result differences from experimental data are closer related to uncertainties within the experimental data (e.g., isotope measurement, depletion model inputs) and the uniqueness of the geometry or material composition (e.g., Gadolinium poisoned fuel rods) than to burnup levels. There was no significant trending of the SAS2H results as a function of fuel burnup. As such, the SAS2H/44GROUPNDF5 sequence is applicable to the high burnup fuel evaluated. While NAC appeared to validate the source term analysis for SAS2H up to 70,000 MWd/MTU, the NRC did not perform an in-depth review this validation, since NAC proposed in Table B2-8 of the technical specifications to evaluate decay heat for spent fuel with burnup greater than 62,500 MWd/MTU in accordance with Regulatory Guide 3.54, revision 3, "Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," and after loading the dose rates are measured on the top, sides and vents as required by technical specification A3.3.1.

6.2.1 Non-fuel Hardware Component

Non-fuel hardware components only apply to PWR fuel assemblies. The source terms for burnable poison rod assemblies (BPRAs) and thimble plugs (TPs) are evaluated by the applicant using the methods from section 5.8.5 of the SAR, which NRC found acceptable in amendment no. 4 (ML15107A467). Consistent with section 5.8.5 of the SAR, a full cask load of 37 BPRAs or TPs is analyzed for LMTC dose rates.

Maximum and average dose rates are shown in table 5.13.1-1 of the SAR for the CC3, and CC7 air outlet and the LMTC. On the sides of the LMTC, the applicant shows that the addition of BPRAs or TPs does not significantly affect the maximum surface dose rate. LMTC radial dose rates for fuel containing BPRAs and TPs are plotted in figures 5.13.1-1 and 5.13.1-2 of the SAR, respectively. The figures illustrate the locations of the dose peaks for the non-fuel hardware. Dose rate peaks are adjacent to the primary source region, i.e., top end-fitting/plenum region.

Reactor control elements (e.g., Control Element Assemblies (CEAs) and Reactor Control Cluster Assemblies (RCCAs)) are evaluated by the applicant using the methods from section 5.8.6 of the SAR. Consistent with section 5.8.6, a maximum of nine CEAs or RCCAs are permitted in the center of the basket. On the bottom of the LMTC, loading of CEAs significantly increases the maximum dose rates as shown in table 5.13.2-1 of the SAR. Due to their interior basket location, CEAs will not make a significant contribution to radial dose rates. The staff found this approach acceptable because when the CEAs are located at the center of the basket, self-shielding will help to minimize the radial dose rates.

6.3 Shielding Model Specification

6.3.1 Configuration of the Shielding and Source

The applicant states that maximum dose rates were calculated through a combination of response function and direct dose rate calculations using a methodology previously approved in amendment no. 3. Response functions and dose rates are computed using MCNP6.2. Three-dimensional MCNP shielding evaluations provide dose rates for transfer and concrete casks at distances up to 4 meters. MCNP models of the fuel assembly, tube, basket, and canister are imported from previously approved methodology from calculation package no. 71160-5030. The neutron absorber plates and retainers are added to the fuel tube model, and the only change to the basket geometry is the addition of the support bars. The applicant also includes statistical uncertainty in the reported dose rates. The statistical uncertainty of the calculated dose rate is part of the computational results of the MCNP code that employs the Monte Carlo method for solving neutron and gamma shielding problems.

6.3.1.1 PWR Models for Concrete Cask Number 3, and 7

There are three preferential heat load patterns (Patterns I, J, and K) that have been evaluated by the applicant for PWR fuel. These patterns allow loading of PWR fuel with cooling times as low as 2 years. Pattern I was designed as an on-going operations pattern with fully burned fuel with typical cool times of 4 years or greater and is evaluated as a uniform pattern, while patterns J and K are primarily designed to allow flexibility in decommissioning scenarios and contain heat load variations (preferential loads). Evaluations were performed by the applicant for the contents within CC3, and CC7.

The preferential loading pattern method from section 5.9 of the SAR was used by the applicant to evaluate PWR fuel, modified to an eight-zone pattern. The minimum cool time is 2 years for Patterns J and K, consistent with anticipated decommissioning use.

The maximum and average dose rates for flexible uniform Pattern I (4-year minimum cool-time) analysis in CC3 and CC7 are shown in section 5.12.4 of the SAR. Pattern J (2 years minimum cooling time) is bounding for the radial and top surfaces, and Pattern I is bounding for the air inlets and outlets.

Maximum assembly average burnup considered within the enrichment/burnup/cool-time space is 70,000 MWd/MTU. The heat load in terms of Watts per assembly for the PWR fuel is shown in section 5.12.4, with heat load zones shown in figure 5.12.4-1 of the SAR. For Pattern I, the total heat load is 51 kW, for Pattern J and K, the total heat load is 42 kW. Pattern I was modeled significantly more heat load than the allowed 42 kW. The staff found this approach acceptable since the allowed heat load is 42 kW and using 51 kW will produce a conservative dose rate results for high burnup fuel above 60,000 MWd/MTU. Dose rates reported in this section are fuel assembly only. Presence of non-fuel hardware components are addressed in section 5.13 of the SAR.

Section 5.8.3.5 of the SAR presents the site boundary analysis. This analysis is based on 40 kW uniform loaded source for the standard (CC1/CC2) cask shield configuration. Site boundary and restricted area boundary dose rates are dominated by the total emissions from the side of the concrete cask, which can be characterized by the average dose rate. The average cask side dose rate of 56.3 mrem/hr is listed in section 5.12.4 for pattern J. the surface dose rates ~~current information~~ for the site boundary evaluation was obtained from a case with a 60 mrem/hr gamma dose rate (56 mrem/hr × 1.07 to account for non-fuel hardware) and 0.8 mrem/hr neutron dose rate for a total of 60.8 mrem/hr. The staff finds this approach acceptable since the CC3 and CC7 average radial dose rate is lower, therefore, the previous off-site dose analysis remains bounding.

NRC staff reviewed the applicant’s calculations for the new heat load Patterns I, J, and K and found them acceptable because the analysis performed by the applicant used accurate source terms and the models adequately represent the MAGNASTOR® shielding design features. Also, the analysis models for the Patterns I, J, and K include bounding physical distributions of the source terms.

6.3.1.2 Lightweight MAGNASTOR® Transfer Cask

The load patterns evaluated require the use of the LMTC due to its liquid neutron shield providing for enhanced heat transfer. The LMTC is designed with a variable thickness lead and water shield. The LMTC uses lead for side shielding and a neutron shield tank that can be drained to further reduce system weight. The lead shield of the LMTC ranges in thickness from 2.5 to 4.0 inches, depending on crane lift capacity. The total thickness of lead plus water is maintained; therefore, increased gamma shielding results in reduced neutron shielding.

LMTC design is implemented for the evaluation of the three high heat load patterns outlined in section 5.12.4 of the SAR. LMTC dimensions are shown in figure 5.12.2-3 and show the cask in the maximum lead configuration (4.0 inches of radial gamma shielding). For conservatism, the expansion tank is not modeled.

NRC staff reviewed the applicant source term calculations for the LMTC and found them acceptable based on bounding PWR fuel assembly parameters that maximize the source terms for the shielding evaluations.

6.3.1.3 Concrete Cask Number 7

The CC7 design contains a 3-inch carbon-steel liner, versus the 1.75-inch carbon steel liner in the base cask design. A revised lid/top cask section design is implemented by the applicant within the CC7 design. The revised lid/top cask section incorporates the outlet vent structure and has a variable thickness concrete section. Only the minimum lid thickness section is evaluated by the applicant for the CC7 design. The dimensions for CC7 are shown in figure 5.12.2-1 of the SAR for the cask body, with detail on the cask upper section/lid shown in figure 5.12.2-2 of the SAR. The results in NAC report no. 71160-5033 demonstrated that the CC7, which has a modified lid/outlet design as compared to the CC3/5, is bounding for top and air outlet dose rates with 891.6 mrem/hr for undamaged fuel. NAC established a technical specification limit of 900 mrem/hr for the dose rates at the top of the cask. The NRC staff reviewed the applicant's shielding calculations for the CC3 and CC7 designs and found them acceptable because the analyses performed by the applicant included models adequately representing the source and cask shielding design features. Also, the analysis models for the Patterns I, J, and K include bounding physical distributions of the source terms.

6.3.1.4 Non-fuel Hardware in the Concrete Cask Number 3, 7 (CC3, CC7) and Lightweight MAGNASTOR® Transfer Cask

Burnable poison rod assemblies and TPs are evaluated by the applicant using the methods from section 5.8.5 of the SAR, for a full cask load of 37 BPRAs or TPs. The radiation source for the BPRAs is dominated by Co-60 gammas. Therefore, the spectrum of the activated BPRAs is not decay time sensitive. As a result of the dominant Co-60 contribution, the burnup/cool-time loading table reflects the cool-time increase required to decay to a limiting Co-60 curie content for each assembly type at each burnup level. System users may choose to directly implement the burnup/cool-time tables on a generic fuel type basis or to determine site-specific minimum BPRA cool times based on the Co-60 curie limit in table 5.8.5-3 of the SAR, which is limited in Table B2-6 of Appendix B of the technical specifications. The radiation source for the TPs is dominated by Co-60 gammas. Therefore, the spectrum of the activated TPs is not decay time sensitive. As a result of the dominant Co-60 contribution, the burnup/cool-time loading table reflects the cool time increase required to decay to a limiting Co-60 curie content for each assembly type at each burnup level. System users may choose to directly implement the burnup/cool-time tables on a generic fuel type basis or to determine site-specific minimum thimble plug cool times based on the Co-60 curie limit in table 5.8.5-5 of the SAR. CC3, CC5, CC7 air outlet and LMTC results are shown in table 5.13.1-1 of the SAR.

LMTC radial dose rates BPRAs and TPs are plotted in figures 5.13.1-1 and 5.13.1-2 of the SAR, respectively. The figures illustrate the locations of the dose peaks for the non-fuel hardware. Dose peaks are adjacent to the primary source region, i.e., top end-fitting/plenum region.

The staff evaluated the dose rates on the sides of the LMTC and found that the addition of BPRAs or TPs does not significantly affect the maximum surface dose rate. This happens because the dose peaks are located at fuel region.

NRC staff reviewed the applicant calculations for the non-fuel hardware and found them acceptable because the analysis performed by the applicant included models representing the source and the cask shielding design features adequately. Also, the analysis models for the Patterns I, J, and K include bounding physical distributions of the source terms.

6.3.1.5 Non-fuel Hardware Components – Partial Length Shield Assemblies

The PLSAs are modeled by the applicant using the Westinghouse 15×15 fuel type. PLSAs contain 42-inch (106.68 cm) stainless-steel inserts in the bottom of each fuel rod and a natural uranium blanket for the top 6 inches of the active core. The 42-inch stainless-steel inserts at the bottom reduce the active fuel length and have no effect on the outside dimensions of the assembly. To account for the self-shielding of the stainless-steel portion of the PLSA, the applicant updated the occupied area fraction of the PLSAs and the fuel homogenization. The analysis details for the PLSA mass and source terms are presented in section 5.13.3 of the SAR. An additional fuel assembly universe with a cut plane for the PLSA is added to the MCNP model. The PLSAs are limited with maximum assembly average burnup of 40 GWd/MTU, minimum assembly average enrichment of 1.2 wt % 235U, and minimum cool time of 6.5 years. Loading is limited to 9 PLSAs in the center of the basket and the dose rates results are shown in table 5.13.3-1 of the SAR.

NRC staff reviewed the applicant calculations for the non-fuel hardware components for PLSAs and found them acceptable because the analysis performed by the applicant included maximum assembly burnup, minimum enrichment, and minimum cool time which gives the maximum source terms.

6.3.1.6 BWR for Load Patterns A, B, and C

There are three preferential heat load patterns (Patterns A, B, and C) which have been evaluated by the applicant for the LMTC and CC3, CC5, CC7 casks. These patterns allow loading of BWR fuel with cool times as low as 2 years. Pattern A was designed as an on-going operations pattern with fully burned fuel with typical cool times of 4 years or greater and is evaluated as a uniform pattern, while patterns B and C are primarily designed to allow flexibility in decommissioning scenarios and contain heat load variations (preferential loads). The three unique patterns are summarized in section 5.14.4 of the SAR. Pattern A has a total heat load of 47.437 kW/cask. Patterns B and C both have total heat loads of 42 kW/cask.

Dose rate evaluations were performed by the applicant for the contents within CC3, CC5, and CC7. The load patterns evaluated require the use of the LMTC due to its liquid neutron shield providing for enhanced heat transfer. The CC7 is a variable height cask to support loading of either BWR/2-3 (which includes a shorter fuel assembly) or BWR/4-6 fuel. For BWR/4-6 fuel, the cask height increases to 197.8 inches. The BWR hybrid fuel assemblies were used by the applicant in this evaluation. Hybrid means that the BWR fuel assemblies contain maximum fuel and hardware masses to maximize the source terms. The staff finds this approach acceptable mainly because the fuel assembly may contain less fuel and less hardware which makes this a conservative assumption. Section 5.8.1 of the SAR contains geometry data for the BWR hybrids. The applicant used a combination of direct solution cases and cases using the response function method to evaluate the loading of BWR fuel assemblies into the CC3, CC5, and CC7.

The dose rate evaluation is based on Pattern A loading, with 4 or 2.5 inches of radial lead shielding, and 4 years minimum cool time. The analyzed Pattern A conservatively models significantly more heat load than the allowed 42 kW. The evaluation of other configurations (Patterns B and C at 2 years cool-time) was performed to provide bounding estimates to ensure that dose rates during loading operations are ALARA. The maximum and average dose rates for the Pattern A (4-year minimum cool-time) analysis are shown in table 5.14.4-1 of the SAR for the LMTC. The maximum dose rates for the minimum lead configuration are produced by a wet canister with a dry (empty) neutron shield tank. The NRC staff reviewed the applicant's source terms, and dose rate analyses for the BWR for Load Patterns A, B, and C and found them acceptable because the analyses performed by the applicant included models adequately representing the source and cask shielding design features. The source properties (material and geometric) were appropriate or conservative for the contents for which the dose rates were calculated.

6.3.1.7 Damaged Fuel

The BWR damaged fuel basket contains up to 81 assemblies, with 12 basket locations designated for damaged fuel. The shorter canister also has an 8-inch lid to accommodate the longer damaged fuel basket. The minimum cool time is 4 years for Pattern A and 2 years for Patterns B and C. There are three unique patterns, which are summarized in section 5.14.5 of the SAR.

Packing fractions from 30% to 75% were used by the applicant for the mass and volume of damaged fuel in the top nozzle, plenum, and active fuel regions for the 12 damaged fuel locations in the basket. The active fuel region is divided into two axial zones, with the lower region containing only damaged fuel and the upper region containing homogenized grid material. The source profile peaking factors of 2.314 and 1.22 are used by the applicant to compute the tally multiplier for the 12 DFC locations for neutron and gamma sources, respectively. The staff finds this approach acceptable because it is consistent with the methodology presented in NUREG/CR-7203.

The maximum dose rates for damaged fuel are based on 69 undamaged fuel assemblies and 12 damaged fuel assemblies. There is an increment on the dose rates at the concrete cask (CC) top and outlet due to the 8-inch BWR DF lid. No increment of the dose rates at the top of the LMTC. The staff agreed with the applicant after reviewing the dose rates calculations by applicant.

Damaged BWR fuel was evaluated by the applicant for the CC3, CC5, and CC7 by assuming the entire fuel assembly collapsed. Packing fractions (PFs) up to a maximum of 0.75 were considered. The NRC staff finds this approach acceptable because it is consistent with the methodology presented in NUREG/CR-7203, "A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages."

For damaged fuel with tie rod, the applicant assumed that the tie rod was unable to hold the fuel assembly and all fuel assembly materials had collapsed. The staff reviewed this assumption and finds it to be conservative even for fuel assemblies that have tie rods to hold the damaged fuel. The additional self-shielding provided by the collapsed fuel is retained. The application included the shielding by the damaged fuel basket corner weldment and DFC itself in the damaged fuel model. In its model 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 impact of damaged fuel on system dose rates is summarized in section 5.12.5 of the SAR. There is an increment on the dose rates at the CC top and outlet due to the 8-inch BWR DF lid. No increment of the dose rates at the top of the LMTC. The staff agreed with the applicant because the upper region has only the grid materials which has less neutron contributions.

The damaged fuel material is homogenized over the inner width of the DFC. The source profile peaking factors of 2.314 and 1.22 are used by the applicant to compute the tally multiplier for the 12 DFC locations for neutron and gamma sources, respectively. Applying the axial peaking factor to the full damaged fuel mass artificially increases the total source in the TSC.

6.3.1.8 Shrinkage of Concrete in Cask Lid

To evaluate the concrete changes in the lid due to removal of the ACI codes, NAC evaluated the effects on dose rate due to concrete shrinkage on the MAGNASTOR® overpack lid. The evaluation focuses on the potential effects that any radial concrete shrinkage would have on the lid's ability to perform its radiation shielding safety function. In its supplement dated March 18, 2022, Enclosure 2, "Potential Concrete Radial Shrinkage," NAC stated that based on publicly available literature, for every 100 ft of concrete there is about 0.6 inches of shrinkage. Applying this approach to the MAGNASTOR® overpack concrete top lid for the two lids shown in drawing no. 71160-L261, Sheet 5, yields approximately 0.039 inches of shrinkage on the diameter [78.5 inches x (0.6 in./(100 ft x 12 inches)) = 0.039 inches], which is equivalent to about a 0.02-inch radial gap around the edge of the concrete cask lid. As discussed above in "Concrete Shrinkage," the staff finds this approach acceptable.

6.3.2 Materials Properties

Section 8.7 of the SAR states that MAGNASTOR® uses lead, concrete, and steel as the principal shielding materials. Lead and steel are the primary gamma radiation shielding materials in the transfer cask. Concrete provides neutron radiation shielding for the concrete cask based on the silicon and water content of the concrete. Silicon, hydrogen, and oxygen are low atomic number materials that are effective in thermalizing and capturing energetic neutrons. Since the density of these materials is a relatively fixed function of the concrete mix, the thickness of the concrete shell is designed to establish the required neutron shielding. The concrete is poured and cured in place around a reinforcing bar that provides structural rigidity.

6.4 Shielding Analyses

6.4.1 Computer Codes

NAC uses the SAS2H code sequence of the SCALE 4.4 package with the 44-group ENDF/B-V cross-section libraries to generate source terms for the shielding analysis. SAS2H includes an XSDRNPM neutronics model of the fuel assembly to resolve resonances and the ORIGEN-S code for fuel depletion and source term calculations. Source terms are generated for both UO_2 fuel and fuel assembly hardware. Source terms for the hybrid fuel assemblies are generated using the SCALE 4.4 sequence as discussed in section 5.2 of the SAR.

NAC used the three-dimensional MCNP code for the shielding evaluations to provide dose rates for transfer and concrete casks at distances up to 4 meters. NAC-CASC, a modified version of the SKYSHINE-III code, calculates site boundary dose rates for either a single cask or cask array. Section 5.6 of the SAR provides more detail on the shielding codes.

The staff found these codes acceptable because these codes are capable of handling the geometries and configurations of the dry storage system design features and the contents for normal, off-normal, and accident conditions. These computer codes are capable of analyzing storage containers (e.g., dry storage system) that have axial or radial variations in features relied on for shielding, inlet and outlet vents, and other features that can be streaming paths and, for a dry storage facility, variations in facility features that can affect dose rates. This also includes configurations of contents that result in variations in the physical distribution of the contents' source term, which can also affect dose rates.

6.4.2 Flux-to-Dose Rate Conversion Factors

The ANSI/ANS 6.1.1-1977, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," flux-to-dose rate conversion factors are used in all shielding evaluations. Neutron and gamma dose conversion factors are listed in table 5.6.5-1 and table 5.6.5-2 of the SAR, respectively.

6.4.3 Shielding Discussion and Dose Results

6.4.3.1 Concrete Cask Number 3, 5, 7 and Lightweight MAGNASTOR® Transfer Cask Dose Rates

The maximum and average dose rates for the Pattern I (4-year minimum cool-time) analysis for the LMTC are shown in table 5.12.4-1 of the SAR. The surface corresponding dose rate profiles are shown in figure 5.12.4-2 through figure 5.12.4-4 of the SAR. Radial dose rates for maximum and minimum lead thickness are displayed. Based on a fixed cask outer diameter, the thicker lead shield case increases gamma shielding (lead) and reduces neutron shielding (water) and vice versa. Top and bottom dose rates are not impacted by the radial shield configuration. Maximum dose rates for the maximum lead configuration are produced by a dry canister with a wet (filled) neutron shield tank.

The staff reviewed the applicant's shielding analysis for the transfer cask and found it acceptable for the changes identified in section 6.1.1 of this SER. The results cover all reasonably foreseeable wet and dry conditions for the LMTC canister.

Flexible uniform (Pattern I) and preferential load (Pattern J and K) patterns were evaluated in the CC3 for PWR, CC5, and CC7. The resulting maximum and average surface dose rates for the preferential pattern are shown in table 5.12.4-2 of the SAR. Pattern J is bounding for the radial and top surfaces, and Pattern I is bounding for the air inlets and outlets.

The staff reviewed the applicant's shielding analysis and found it acceptable for the changes identified in section 6.1.1 of this SER. The results cover all reasonably foreseeable wet and dry conditions for the LMTC.

6.4.3.2 Dose Rate Change Due to Potential Shrinkage of Concrete in the Lid

The applicant performed dose rates analysis taking FSAR Revision 0 (ML091030364) dose rates from NAC calculation no. 71160-5014, which used the MCNP5 computer code. Dose rates for base cases represent a full spectrum run and response solution, as described in the NAC SAR.

For amendment no. 12, the applicant used MCNP6 to calculate the dose rates at the top lid. MCNP is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. Specific areas of application include, but are not limited to, radiation protection and radiation shielding.

The applicant's use of a 0.04- and 0.08-inch radial gap is well in excess of the 0.02-inch maximum expected radial gap, previously discussed in the first paragraph of this portion of the SER. The applicant found that the shrinkage does result in an increase in the absolute value for dose rates. However, all dose rate results fall within the statistical uncertainty band of the original calculations and, therefore, are not significant enough that they would impact either occupational doses or site boundary dose limits.

The staff reviewed the methodology employed by the applicant and found them acceptable based on the facts that they used a radial gap which is larger than the 0.02 in, showing that the increase in dose rate is not significant enough that it would exceed either occupational doses or site boundary limits. The staff agrees that the results fall within the statistical uncertainty band of the original solutions, and therefore no additional off-site or occupational dose analyses are necessary. The staff also found acceptable the use of MCNP based on facts that this computer code has commonly been used in previous NRC approvals for the MAGNASTOR® system and is appropriate for this evaluation.

6.4.4 NRC Confirmatory Analyses

The staff independently calculated source terms for the bounding PWR WE14×14 fuel assemblies using combinations of different enrichments, burnups, and cooling times using ORIGEN/ARP, SCALE 6.1. The staff reviewed the applicant's analyses of the dose rates for the lightweight MAGNASTOR® transfer cask and found them acceptable because they were conservative and demonstrated that the system meets the off-site dose rate limits of 10 CFR 72.104. The staff also reviewed the applicant's analyses of the dose rates for the CC3 for PWR and, CC3/CC5 for BWR, and CC7 storage casks. Using irradiation parameter assumptions similar to the applicant's, the staff obtained bounding source terms that were similar to or bounded by those determined by the applicant and therefore finds the applicant's result acceptable. The staff finds that the applicant has correctly assessed the bounding dose rates for all proposed contents, as defined in tables 5.1.3-1 through 5.1.3.2 of their SAR. Based on this review and analyses, the staff concludes that the applicant has demonstrated that the MAGNASTOR® dry cask storage system meets the radiation protection requirements of 10 CFR 72.104, 72.106, 72.126, and 72.128.

6.5 References

1. NUREG/CR-7012, "Uncertainties in Predicted Isotopic Compositions for High Burnup PWR Spent Nuclear Fuel," ORNL/TM-2010/41, Oak Ridge National Laboratory, January 2011 (ML110140213).

2. NUREG/CR-7203, "A Quantitative Impact Assessment of Hypothetical Spent Fuel Reconfiguration in Spent Fuel Storage Casks and Transportation Packages," ORNL/TM-2013/92, Oak Ridge National Laboratory, September 2015 (ML15266A413).

3. American National Standards Institute (ANSI)/American Nuclear Society (ANS) 6.1.1, “Neutron and Gamma-Ray Flux-to-Dose-Rate Factors,” 1977.

6.6 Evaluation Findings

The staff reviewed the applicant’s shielding analyses for the amendment no. 11 to the MAGNASTOR® dry storage system design and finds that the approaches and methodologies used in these calculations and the results are acceptable for the LMTC and CC3 for PWR, CC3 and CC5 for BWR, and CC7 system design; new loading patterns I, J, and K for the PWR fuel basket; new loading patterns A, B, and C for the new undamaged BWR 89-assembly fuel basket; and the new damaged BWR 81-assembly fuel basket.

The staff concludes that the shielding and radiation protection design features of the MAGNASTOR® system, including the changes to the requirements in TS A4.2 for the top lid concrete cask comply with 10 CFR Part 72, and that the applicable design and acceptance criteria continue to be satisfied. The evaluation of the concrete top lid in terms of shielding and radiation protection design features provides reasonable assurance that the system will still provide shielding and radiation protection from the spent fuel. This finding is based on the appropriate regulatory guides, applicable codes and standards, the applicant’s analyses, the staff’s evaluations, and acceptable engineering practices.

Based on the information provided by the applicant on how the shielding evaluation was conducted, the staff concludes that the requested changes meet the regulatory limits and the acceptance criteria specified in NUREG-2215 and provides reasonable assurance of the safe transfer and storage of the spent fuel and non-fuel hardware as specified in the technical specifications for the MAGNASTOR® system. On these bases, the staff finds:

F6.1 Chapter 5 of the MAGNASTOR® SAR sufficiently describes the shielding design bases and design criteria for the SSCs important to safety.

F6.2 The MAGNASTOR® system radiation shielding features of CC3, CC5, and CC7 and the LMTC and their associated confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, and 10 CFR 72.236(d).

F6.3 The shielding and radiation protection design features of the MAGNASTOR® system, including the concrete cask, the transfer cask, and the TSC, are in compliance with 10 CFR Part 72, and the applicable design and acceptance criteria have been satisfied. The evaluation of the shielding and radiation protection design features provides reasonable assurance that the system will provide safe transfer and storage of spent fuels. This finding is based on a review that considered applicable regulations, the appropriate regulatory guides, applicable codes and standards, the applicant’s analyses, the staff’s confirmatory analyses, and acceptable engineering practices.

Chapter 7 CRITICALITY EVALUATION

Staff reviewed the amendment request to determine if the MAGNASTOR® system with the new contents and loading configurations continues to remain subcritical under all credible normal, off-normal, and accident events encountered during the handling, loading, transfer, and storage of spent nuclear fuel. Staff reviewed the applicant's criticality safety analysis to ensure that all credible bounding scenarios were adequately identified and any potential consequences on the criticality safety of the MAGNASTOR® dry cask storage system continues to meet the regulatory requirements of 10 CFR 72.124 and 72.236. The conclusions of the staff are based on the information provided by the applicant and the supporting calculations for the addition of:

- A new LMTC
- A new CC7
- New loading patterns and basket types for both PWR and BWR fuel types
- New BWR basket design allowing up to 89 undamaged BWR fuel assemblies
- New BWR damaged fuel basket design allowing up to 81 undamaged BWR fuel assemblies and up to 12 DFC locations
- New DFC for BWR fuel

The applicant also removed previously licensed BWR basket configurations since they are bounded by the new 89 and 81 configurations.

7.1 Criticality Design Criteria and Features

The MAGNASTOR® storage system consists of a TSC, a concrete and metal storage overpack, and a lead-shielded transfer cask. Criticality safety is provided by a combination of fissile mass and enrichment controls, geometry control, and fixed neutron absorbers contained in the basket. The MAGNASTOR® system may also contain DFCs to store damaged PWR and BWR fuel. Fixed neutron absorber sheets are attached to the walls of the fuel assembly tubes and sit between each fuel assembly in the basket. PWR fuel requires the use of soluble boron in the water that is used to flood the canister during loading and unloading operations. The minimum soluble boron content is based on the assembly type and the maximum initial assembly enrichment.

Since the previously approved MAGNASTOR® overpack design is not altered by this amendment, staff evaluated only the addition of the LMTC, CC7, new loading patterns for both PWR and BWR fuel assemblies, the new DFC for BWR fuel, and the two new BWR basket designs. The applicant made changes to the CoC and the technical specification to allow these additions to the MAGNASTOR® storage system.

Staff reviewed the applicant's model and assumptions and finds that they are consistent with the description of the design and contents provided in the SAR. Staff also evaluated the information the applicant provided in the amended SAR and found the criticality calculations were sufficiently detailed to support the staff evaluation. Based on this review, the staff finds that the applicant continues to meet the requirements of 10 CFR 72.236 and 10 CFR 72.124.

7.2 Fuel Specifications

Consistent with previous amendments to the FSAR, the applicant identified the fuel contents based on the specified fuel type and identified the conservative bounding values for the criticality significant parameters for each fuel. There are no new fuel types added by this amendment, only new configurations in the baskets. Staff reviewed the SAR and the technical specification and found the applicant adequately specified the proposed fuel specifications that could impact the criticality safety of the MAGNASTOR® in the new basket configurations, including DFCs.

7.3 Model Specifications

The applicant evaluated the storage of both damaged PWR fuel and undamaged PWR and BWR fuel assemblies in amendment no. 0 to the MAGNASTOR® system. The methodology for these evaluations are unchanged in this amendment. The applicant used the same criteria to evaluate the new models to support the new basket configurations, as well as the damaged BWR fuel. The applicant also uses multiple sizes of concrete and transfer casks to accommodate all of the PWR and BWR TSCs and added the CC7 and LMTC as part of this amendment.

The key model assumptions the applicant used in the analysis include:

- Modeling all of the assemblies as fresh fuel at a 96% theoretical density of 10.52 g/cm^3
- Neglecting the structural materials except for BWR fuel assembly channels
- Excluding integral fuel assembly neutron absorbers (i.e., BWR gadolinium rods, PWR erbium, etc.)
- Fixed neutron absorber sheets
- Structural integrity of the basket in normal, off-normal, or accident events

The boron loadings of the neutron absorber sheets specified in technical specifications, appendix A, 4.1.1, used in the MAGNASTOR® system are unchanged by this amendment. In addition, the structural integrity of the baskets is maintained as described in chapter 12 of the SAR and therefore does not alter the geometry necessary to maintain the relative position and geometry of the fuel assemblies.

Based on the applicant's use of conservative modeling assumptions listed above, all of which tend to drive the reactivity of the system higher, staff finds that the modeling assumptions used in the criticality analysis are conservative and adequate to evaluate the subcriticality of the changes to the MAGNASTOR® system proposed by this amendment.

7.4 Criticality Analysis

The applicant requested the addition of two new BWR basket designs to allow for the storage of up to 89 undamaged BWR fuel assemblies, or 81 fuel assemblies of which up to 12 may be damaged BWR fuel assemblies in DFCs. Damaged fuel must be loaded into DFCs within the 81-assembly basket, and each DFC may contain an undamaged fuel assembly, a damaged fuel assembly, or fuel debris not to exceed the fissile mass equivalent of one BWR fuel assembly.

The applicant also requested nine new loading patterns, three for PWR fuel (patterns I, J, and K), three for the BWR 89-assembly basket (patterns B, C, and D), and three for the BWR 81-assembly basket (patterns A, B, and C). All basket types rely on integrated neutron absorber sheets with a minimum ^{10}B loading of 0.036 g/cm^3 for the PWR basket and 0.027 g/cm^3 for the BWR baskets, and take either 75% or 90% credit based on type and payload as shown in table 6.1.1-5 of the SAR and technical specifications, appendix A, 4.1.1. The PWR basket also relies on soluble boron concentrations in the pool based on payload and are unchanged by this amendment. The soluble boron concentrations are provided in technical specifications, appendix A, 3.2 “MAGNASTOR SYSTEM Criticality Control for PWR Fuel.”

The applicant calculated the maximum k_{eff} + 2σ using conservative assumptions for undamaged fuel contained within both the concrete casks and the transfer casks. These results are presented in section 6.1.1.1 of the application and demonstrate that all undamaged configurations are below the upper safety limit of 0.9376 for k_{eff} + 2σ for all cask bodies. The applicant defined a maximum initial enrichment for each fuel type in section 6.7.6 that is unchanged by this amendment. Specific findings for each of the changes requested by this amendment are addressed in the subsections below.

7.4.1 PWR Fuel Analysis

The applicant evaluated the maximum enrichments for up to 37 PWR fuel assemblies, including up to four damaged PWR fuel assemblies, for storage in the MAGNASTOR® system using the ^{10}B contents for each fuel type and new loading configurations I, J, and K as stipulated in the SAR. The applicant determined these new loading patterns did not have an effect on the criticality safety of the package and continue to result in k_{eff} values below the upper subcritical limit (USL). Staff evaluated the new PWR fuel configurations and finds that the applicant’s analysis adequately bounds the new basket configurations by assuming the optimum moderation, maximum pitch, maximum fuel pellet outer diameter, minimum fuel rod outer diameter, and minimum cladding thickness to provide the most reactive fuel parameters. Staff also evaluated the applicant’s basket variations, including minimizing the tube interface width, minimizing the tube thickness, and shifting of fuel assemblies within the basket, and staff finds the variations acceptable since they maximize reactivity.

7.4.2 Undamaged BWR Fuel Analysis

The applicant evaluated the undamaged BWR fuel assemblies based on an optimum loading of BWR fuel assemblies at the most reactive lattice moderator (H/U) ratio as justified in section 6.7.5 of the SAR. The optimum system configurations are based on the 89-assembly basket configuration with a 4.0 wt% ^{235}U initial enrichment. As noted by the applicant, and discussed below, certain assembly types and configurations exceed the USL and require the use of an underload or preferential loading configuration.

The design options of the MAGNASTOR® cask permit the replacement or removal of up to 24 neutron absorber sheets. The applicant removed all sheets for all analyses with the exception of the underloading of 84-assemblies in the basket. In this case, the applicant kept the neutron absorber sheets with an area density of at least 0.027 g/cm^2 B-10 in place. Replacement sheets are composed of un-borated aluminum. Using the most reactive basket and fuel assembly shifting, the applicant analyzed each BWR fuel type as specified in section 6.7.5 of the SAR at 4.0 wt% ^{235}U for the 89-assembly configuration and 4.5 wt% ^{235}U for the 84-assembly configuration. The applicant demonstrated that there were no statistically significant reactivity changes associated with the neutron absorber sheet removal or replacement in the models. Since the applicant’s analysis indicated that the USL was exceeded for some of these configurations, the applicant left specific assembly locations vacant as specified in table 6.1.1-8, and specified technical specifications appendix B, table B2-9, item I.c, to allow for additional underloading options. Staff finds NAC’s proposal to leave assembly locations empty acceptable because this reduces the overall fissile material present in an underloaded basket, which allows the maximum k_{eff} to remain below the USL.

The applicant also evaluated a non-uniform preferential loading of the MAGNASTOR® basket. The applicant modeled 37 inner assemblies in the basket at a lower enrichment level than the 52 outer cell locations. Since neutron leakage is higher from the periphery of the basket, placing the lower enriched assemblies in the center allows for higher enrichment to be loaded in the cask. Staff evaluated the applicant's assessment and finds that the increased enrichment on the periphery of the basket is acceptable because the preferential loading will result in a k_{eff} below the USL.P

Based on the most reactive basket configuration, the applicant evaluated each of the BWR fuel assembly types at varying enrichment levels to determine the maximum enrichment where the resultant k_{eff} + 2σ remained below the USL. The allowable assemblies are listed in table 6.7.6-9 of the SAR (table B2-11 in appendix B of the technical specifications), with the maximum allowed planar average enrichments of fuel assemblies with and without partial length rods listed in table 6.7.6-8 (table B2-12 in appendix B of the technical specifications). The applicant evaluated the BWR fuel enrichments at a range of neutron absorber plate boron loading (0.027, 0.0225, and 0.020 ^{10}B g/cm^2) as listed in tables 6.7.6-13 and 6.7.6-14. Preferential loading enrichments are listed in table 6.7.6-15 and only apply to the neutron absorber plates at a minimum 0.027 10B g/cm^2. Preferential and underloading results are summarized in table 6.7.6-16 of the SAR.

Based on this analysis, the applicant determined that the maximum reactivity of the MAGNASTOR® storage system loaded with undamaged BWR fuel was a k_{eff} + 2σ of 0.93679 for the wet transfer cask configuration, and 0.43685 when stored in the concrete overpack in a dry condition. Since there are no design-basis off-normal or accident conditions that could affect the system reactivity, the maximum reactivity is the same for all cases. Based on the conservative assumptions used by the applicant described above, including using the most reactive fuel and most limiting basket configurations, staff finds this an acceptable approach for the undamaged BWR fuel analysis.

7.4.3 Damaged BWR Fuel Analysis

The MAGNASTOR® system also allows for damaged fuel assemblies to be located in the basket. Up to 12 damaged assemblies may be placed in DFCs, with the other 69 or more BWR fuel assemblies remaining intact. Each of the DFCs are screened to prevent fissile material release into the TSC cavity in the event of gross cladding failure of the fuel. The applicant establishing the maximum reactivity of the 81-assembly basket containing undamaged fuel and the maximum reactivity configuration for damaged fuel. The applicant also evaluated the optimum moderator density of both partial flooding and preferential flooding of the DFC And demonstrated that the maximum reactivity of the damaged fuel configurations remains below the USL by comparing the two cases.

Each DFC is comprised of a stainless-steel rectangular box with a screened lid and bottom plate which contains four, 1.3-inch diameter screened holes and is designed to freely drain and fill to avoid the preferential flooding of a DFC. DFC contents include undamaged, clad fuel assemblies with functional grids (i.e., fuel assemblies retaining cladding and fuel rods in their as-designed configuration), unclad fuel, and a homogenized fuel/water mixture. Unclad fuel assemblies are modeled with no cladding or end-fitting. Homogenized fuel is modeled as a rectangular volume with the optimal mixture height determined by analysis.

The applicant assumed that a floating array of fuel pellets, or a homogeneous mixture of water and fuel would represent a conservative physical configuration of any fuel in a DFC. The higher density of the fuel would result in unclad fuel settling to the bottom of a DFC, and re-suspension of the fuel is not credible since moderator flow into a DFC would not have the force to do so by any normal means. As a result, staff finds this assumption acceptable.

The applicant also evaluated BWR fuel affected by crud induced localized corrosion (CILC), which affects some older generations of BWR fuel assemblies. These assemblies may contain localized cladding failure. The applicant performed a bounding analysis for BWR channeled fuel to cover CILC fuel by removing all of the cladding while maintaining pellet spacing. Since removal of the cladding drives the reactivity of the assembly up, and as a result, drives down the maximum enrichment that is allowable in each basket as shown in tables 6.7.10-1 and 6.7.10-2 (table B2-12h in appendix B of the technical specifications), and since the resultant multiplication factors are below the USL, staff finds this assumption acceptable. Unchanneled fuel assemblies containing CILC fuel, or other fuel types that have damaged cladding identified, must be evaluated under the standard damaged fuel definition in appendix A of the technical specifications, and may require placement in a DFC.

7.5 Criticality Evaluation Summary

All of the applicant's models for the new basket configurations for both PWR and BWR fuels are based on the engineering drawings in the SAR and models submitted as part of associated calculations. The design-basis, off-normal, and accident events do not affect the design of the cask with regards to maintaining the MAGNASTOR® system in a subcritical configuration. This means that calculation models for the normal, off-normal, and accident conditions are the same. Staff imported sample input files provided by the applicant in support of its supplemental calculations to confirm the results provided by the applicant. For these reasons, staff finds that the applicant's evaluation of the criticality design demonstrates that the MAGNASTOR® storage system will continue to allow for the safe storage of spent fuel. This finding is reached on the basis of a review that considered applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7.6 Evaluations Findings

Staff reviewed the information provided in amendment no. 11 of the MAGNASTOR® system application and determined that it is in compliance with the requirements in 10 CFR 72.124 and 10 CFR 72.236(c). Staff also determined that the results of the applicant's evaluation of the new fuel loading configurations, as described in this application, show that the k_{eff}s remain less than the USL of 0.9376 for each of the evaluated cases. The applicant incorporated a number of conservative assumptions and evaluated the fuel configurations over a range of bounding credible scenarios. Limits are imposed in the technical specifications as stated above for each fuel in regard to the minimum ^{10}B concentration in the absorber sheets, soluble boron concentrations in the pool, and allowable enrichments. As a result, staff has reasonable assurance that the MAGNASTOR® spent fuel dry cask storage system, as described in this amendment, will remain subcritical while in storage. Specifically, the applicant's nuclear criticality safety evaluation demonstrates that the MAGNASTOR® spent fuel dry cask storage system will continue to meet the relevant regulatory requirements, and the staff finds the following:

F7.1 The applicant described SSCs important to criticality safety in sufficient detail in the SAR to enable an evaluation of their effectiveness.

F7.2 The cask and its spent fuel transfer systems, including the new CC7 and LMTC are designed to be subcritical under all credible conditions.

F7.3 The criticality design is based on favorable geometry, fixed neutron poisons, and soluble poisons of the spent fuel pool. An appraisal of the fixed neutron poisons has shown that they will remain effective for the term requested in the application and there is no credible way for the fixed neutron poisons to significantly degrade during the requested term in the application; therefore, there is no need to provide a positive means to verify their continued efficacy as required by 10 CFR 72.124(b).

F7.4 The applicant's analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the storage of spent fuel for the term requested in the application.

Chapter 8 MATERIALS EVALUATION

The staff evaluated the materials performance of the new LMTC, CC7, BWR undamaged fuel basket, damaged BWR fuel basket, and DFC to ensure the requirements of 10 CFR Part 72 are met. The LMTC is designed to transfer the TSC between workstations and the concrete cask when limited crane capacity exists or for TSCs with high heat loads. The CC7 is designed to hold the TSC during long-term storage. The BWR undamaged fuel basket is designed to position and support up to 89 BWR fuel assemblies. The BWR damaged fuel basket is designed to position and support up to 81 BWR fuel assemblies which contains up to 12 BWR DFCs. The BWR DFC is designed to confine the BWR fuel material within the can and minimize dispersal into the TSC cavity.

8.1 Materials of Construction

8.1.1 LMTC

As described in SAR sections 3.1.2 and 8.1, SAR table 1.3-1, and the licensing drawings, the LMTC is fabricated from stainless-steel structural components, with the inner and outer shell being ASTM A240 Type 304 stainless-steel, and the top and bottom forging from ASTM A182 Type F304 stainless-steel. The LMTC includes a demineralized water-filled neutron shield tank. The retaining ring, used to prevent a loaded TSC from being inadvertently lifted through its top opening, is fabricated from ASTM A240 stainless-steel. The bottom shield doors are fabricated from ASTM A240/A182 stainless-steel. The transfer cask annulus is isolated using ethylene propylene diene monomer (EPDM) inflatable seals. ASTM B29 Chemical Copper grade bricks are provided for gamma shielding.

8.1.2 Concrete Cask Number 7

As described in SAR sections 3.1.2 and 8.1, SAR table 1.3-1, and the licensing drawings, the CC7 is a reinforced concrete cylinder fabricated from ASTM C150 Type II or I/II Portland Cement (with the concrete mix meeting the requirements of ACI 318) and ASTM A615/A615M carbon steel reinforcing bar. The internal cavity is lined by ASTM A36 carbon steel. Standoffs, fabricated with ASTM A36 or A992 carbon steels, are welded to the cask inner liner and are responsible for guiding the TSC into the concrete cask. An optional heat shield, fabricated from ASTM A36 carbon steel, is available for higher heat loads to reduce concrete temperatures. The top of the concrete cask is closed by one of two upper segment options, fabricated from A36 carbon steel and a cylindrical concrete plug, and attached via eight ASTM A354 Grade BC bolts.

8.1.3 BWR Fuel Basket, Damaged Fuel Basket, Damaged Fuel Can

As described in SAR sections 3.1.2 and 8.1, SAR table 1.3-1, and the licensing drawings, the BWR fuel basket and BWR damaged fuel basket are comprised of fuel tube assemblies, corner support weldments, and side support weldments that are fabricated from SA537 Class 1 Carbon Steel. Each fuel tube supports borated aluminum metal matrix or composite neutron absorber sheets, on up to four interior sides of the tube, that are covered by stainless-steel.

As described in SAR sections 1.3.1.5 and SAR table 1.3-1, and the licensing drawings, the BWR DFC is fabricated from Type 304 stainless-steel. The fuel can is comprised of upper side plates and tube body walls fabricated of ASME SA240 Type 304 stainless-steel, a lid assembly fabricated of ASME SA240/SA479 Type 304 stainless-steel, and a bottom plate fabricated of ASME SA240 XM-19.

As described above, the staff reviewed the information provided on materials of construction and verified sufficient detail exists to support a safety finding. Therefore, the staff finds the applicant's description of the materials of construction to be acceptable.

8.2 Drawings

The applicant provided new drawings in section 1.8 of the SAR to incorporate the LMTC, CC7, and BWR fuel basket, damaged fuel basket, and DFC. The drawings include a parts list that provides the material specification of each component, and they also provide the welding, examination, and coating requirements. The staff notes that the level of detail in the new and revised drawings are consistent with those of the previously approved drawings in the MAGNASTOR® System FSAR Revision 0 (ML091030628). The staff reviewed the drawing content with respect to the guidance in NUREG-2215 section 8.5.1, "Drawings" and NUREG/CR-5502, "Engineering Drawings for 10 CFR Part 71 Package Approvals." The staff confirmed that the drawings provide an adequate description of the materials, fabrication, and examination requirements to assess their properties. Therefore, the staff finds the drawings to be acceptable.

8.3 Codes and Standards

8.3.1 Lightweight MAGNASTOR® Transfer Cask

The staff verified that the LMTC uses the same ASTM International steel materials as the previously approved transfer casks in the MAGNASTOR® System FSAR Revision 5 (ML17132A265). As described in section 3.1 of the SAR, the LMTC and lifting devices are designed, fabricated, and load-tested to the requirements of ANSI N14.6 as well as the NRC guidance in NUREG–0612, "Control of Heavy Loads at Nuclear Power Plants."

8.3.2 Concrete Cask Number 7

The staff verified that the CC7 cask uses the same ACI construction codes for the cask body and ASTM International steel materials as the previously approved cask versions in the MAGNASTOR® System FSAR Revision 5. The staff notes that the cited standards are consistent with NRC guidance in NUREG-2215, which states that concrete structures designs may use ACI codes and that structural components of the overpack may be constructed of ASTM materials. The staff's structural evaluation of the CC7 cask is included in SER chapter 4.

8.3.3 BWR Fuel Basket, Damaged Fuel Basket, Damaged Fuel Can

The staff verified that the BWR fuel basket, BWR damaged fuel basket, and BWR DFC use the same ASME standards as the previously approved fuel baskets and damaged fuel cans in the MAGNASTOR® System FSAR Revision 5. The staff notes that the cited standards are consistent with NRC guidance in NUREG-2215, which states that fuel basket structures may be fabricated in accordance with ASME BPV Code Section III, Subsection NG, "Core Supports." Similarly, minor changes to other components (e.g., dimensions) did not affect the applicable codes and standards.

The staff finds that the identified codes and standards are appropriate for the material control of the components. Therefore, the staff finds the materials codes and standards to be acceptable.

8.4 Welding

8.4.1 Lightweight MAGNASTOR® Transfer Cask

The LMTC uses the same welding codes and standards as the previously approved transfer cask (TC) designs in the MAGNASTOR® System FSAR Revision 5. The weld design and non-destructive examination (NDE) will be in accordance with ASME BPV Code Subsection NF, and the welding procedures, processes, and welder qualifications will be in accordance with either ASME BPV Code Section IX or American Welding Society (AWS) D1.1, “Structural Welding Code—Steel.” The visual examinations of the TC welds shall be performed in accordance with ASME Code, Section V, or AWS D1.1, with acceptance per Section III, Subsection NF, Article NF-5360. Critical load-bearing welds shall be either dye penetrant (PT) or magnetic particle (MT) examined in accordance with ASME Code, Section V.

8.4.2 Concrete Cask Number 7

The CC7 concrete cask uses the same welding codes and standards as the concrete cask design NRC approved in amendment no. 0 (ML090350509). The weld design and NDE will be in accordance with ASME BPV Code, Section VIII, Division 1, Part UW, and the welding procedures, processes, and welder qualifications will be in accordance with either ASME BPV Code Section IX or AWS D1.1. Inspections of the welded steel components of the concrete cask shall be in accordance with ASME Code, Section VIII or AWS D1.1.

8.4.3 BWR Fuel Basket, Damaged Fuel Basket, Damaged Fuel Can

The BWR fuel basket, BWR damaged fuel basket, and BWR DFC use the same welding codes and standards as the previously approved designs in the MAGNASTOR® System FSAR Revision 5. The weld design and NDE will be in accordance with ASME BPV Code Subsection NG, and the welding procedures, processes, and welder qualifications will be in accordance with ASME BPV Code Section IX. The visual examinations of the welds will be performed in accordance with ASME BPV Code, Section V, Articles 1 and 9, with acceptance per Section III, Subsection NG, Article NG-5360.

As described above, the staff reviewed ASME and AWS codes for the design, fabrication, and examination of the welds in this application and determined that the codes identified are consistent with the guidance in NUREG-2215. Therefore, the staff finds applicant’s approach for welding and NDE to be acceptable.

8.5 Material Properties

The applicant did not add additional materials or make any changes to the mechanical properties and thermal properties used in the structural analyses and thermal analysis. However, the staff reviewed the applicant’s new thermal analysis to ensure that those material properties remain valid under the service conditions associated with the new LMTC, new CC7, increased system heat load capacity, new fuel zoned loading patterns, and new undamaged BWR fuel basket, BWR damaged fuel basket, and BWR DFC that were added in this amendment. In SAR section 4.11, the applicant evaluated the maximum temperatures of the fuel cladding, fuel basket, canister shell, and cask concrete under normal, off-normal, and accident conditions. The staff reviewed the applicant's analysis and verified that component temperatures remain below each of the material's allowable service temperatures. Therefore, the staff finds the mechanical and thermal properties used in the applicant's structural and thermal analysis to be acceptable.

8.6 Radiation Shielding Materials

8.6.1 Lightweight MAGNASTOR® Transfer Cask

As described in SAR section 8.7, the primary gamma shield materials in the LMTC are lead and steel, the same ASTM materials as the previously approved casks in the MAGNASTOR® System FSAR Revision 5. The lead gamma shield is comprised of ASTM B29 Chemical Copper grade lead bricks designed to "nest" to eliminate both horizontal and vertical gaps. For neutron shielding, the LMTC includes a demineralized water-filled shield tank that can be drained for pool loading operations to reduce the hook wet weight, then refilled to restore neutron shielding prior to performing canister draining, drying, and closure operations. The shell of the LMTC, made of ASTM A240 stainless-steel, is also appropriately accounted for in the shielding analysis.

8.6.2 Concrete Cask Number 7

As described in SAR section 8.7, the ASTM A36 carbon steel liner provides the primary gamma radiation shielding for the CC7 concrete cask, with the ASTM C150 concrete and ASTM A615/A615M carbon steel reinforcing bar also providing measurable gamma radiation shielding. The concrete also provides the neutron shielding for the cask. As stated in section 8.4.2, above, the concrete and carbon steels are the same ACI and ASTM materials as the previously approved overpack in the MAGNASTOR® System FSAR Revision 5.

8.6.3 BWR Fuel Basket, Damaged Fuel Basket, Damaged Fuel Can

The BWR fuel basket and BWR damaged fuel basket make use of borated aluminum metal matrix or composite neutron absorber sheets attached on up to four faces of the fuel tubes and are covered in stainless-steel. The staff verified that the new BWR fuel basket and BWR damaged fuel basket use the same neutron absorber materials as the previously approved fuel baskets in the MAGNASTOR® System FSAR Revision 0 (Boral, metal matrix composite, and borated aluminum have all been previously approved).

Per the above discussion, the staff finds that the applicant is using appropriate materials and provides an adequate description of the dimensions and geometries in the shielding analysis.

8.7 Concrete and Reinforcing Steel

The new CC7 cask body uses the same ASTM A615 carbon steel reinforcing bar as the previously approved casks in the MAGNASTOR® System FSAR Revision 0, complying with the requirements of ACI 349. As described in SAR table 2.1-1 and SAR section 10.1.1, the CC7 concrete cask body is designed in accordance with ACI-349, and construction and inspections in accordance with ACI-318. As described in SAR table 4.1-2, the CC7 cask body follows the temperature requirements of ACI-349. The staff reviewed the thermal analysis in SAR section 4.11 and verified the concrete temperature requirements of ACI-349 are met under the service conditions associated with increased system heat load capacity, new fuel zoned loading patterns, and new BWR fuel basket, BWR damaged fuel basket, and BWR DFC that were added in the amendment, under normal, off-normal, and accident conditions. The staff notes the applicant's uses of ACI-349 are consistent with the guidance of NUREG-2215. Therefore, the staff finds the concrete and reinforcing steel design of the CC7 to be acceptable.

8.8 Bolt Applications

8.8.1 Lightweight MAGNASTOR® Transfer Cask

The LMTC design uses bolts to attach the retaining ring to the top ring of the TC. The staff verified that the new LMTC uses the same bolts as previously approved transfer casks in the MAGNASTOR® System FSAR Revision 5 (ASME SA193, Grade B6 stainless-steel).

8.8.2 Concrete Cask Number 7

As described in SAR section 3.1.2, the top end of the CC7 concrete overpack is closed by an upper segment assembly that is attached by eight upper segment bolts. As described in SAR section 3.11.3.1, the CC7 is lifted using two lift lug assemblies that bolt to the upper forging of the cask liner weldment using lift lug bolts. The staff verified that the new CC7 design uses the same bolts as the previously approved casks in the MAGNASTOR® System FSAR Revision 11 (ML21054A003) (ASTM A354 Grade BC steel).

8.8.3 BWR Fuel Basket, Damaged Fuel Basket, Damaged Fuel Can

The BWR fuel basket and BWR damaged fuel basket use mounting bolts to attach the side and corner weldments to the fuel tube array. The staff verified that the new BWR fuel basket and BWR damaged fuel basket use the same mounting bolts as the previously approved baskets in the MAGNASTOR® System FSAR Revision 0 (ASME SA193, Grade B6 stainless-steel).

Per the above discussion, the staff finds the applicant's bolting materials specifications and mechanical properties to be acceptable.

8.9 Spent Fuel

In SAR section 4.11, the applicant revised the calculated fuel cladding temperatures during drying operations. The revisions include temperature changes that exceed the 65°C threshold recommended in NUREG-2215 to limit potential detrimental effects of hydride reorientation on cladding mechanical performance. However, the staff notes that the exceedance of the recommended thermal cycling threshold was previously reviewed and approved by the staff in CoC no. 1031, amendment no. 0 and is defined in the existing MAGNASTOR® technical specifications.

In addition, the staff notes that more recent research with high burnup, Zircaloy-4 clad PWR fuel with high cladding stresses subjected to multiple, more excessive, thermal cycles (between 100 °C and 230 °C) and a maximum cladding temperature of 400°C has found that hydride reorientation did not detrimentally affect fuel performance under the bending loads associated with drop accidents (Billone, 2014; NRC, 2017). The PWR fuels with cold-worked, stress-relieved (CWSR) claddings, such as standard and low-Tin Zircaloy-4, and Zirlo™, are unlikely to undergo significant hydride reorientation because cladding stresses at drying temperatures are below the threshold stresses needed for hydride reorientation in these alloys (EPRI, 2020). Although PWR M5 clad fuel and BWR CWSR and recrystallized Zircaloy-2 clad fuel may be susceptible to hydride reorientation, the mechanical response of the fuel rods with hydride reoriented cladding is expected to be similar as that observed with Zircaloy-4 clad fuel and not detrimental to fuel rod performance (NRC, 2020). For BWR fuels, the majority use Zircaloy-2 cladding with a zirconium liner that is not susceptible to hydride reorientation (EPRI, 2020). Therefore, based on the staff's prior approval of the applicant's thermal cycling criteria and more recent research that provides additional support for the use of that criteria, the staff finds the revised cladding temperature changes during fuel drying operations to be acceptable.

8.10 Corrosion Resistance and Content Reactions

The staff reviewed the amendment changes and verified that they do not introduce any adverse corrosive or other reactions that were not previously considered in the staff's prior review of the MAGNASTOR® CoC. The materials of construction and the service environments are bounded by those that were previously evaluated in the CoC. Therefore, the staff finds the applicant's evaluation of corrosion resistance and potential adverse reactions to be acceptable.

8.11 Protective Coatings

In the CC7 cask design, the applicant used the same coatings that have been previously approved for use in the MAGNASTOR® System FSAR Revision 0 to mitigate atmospheric corrosion of carbon steel. Therefore, the staff finds the coatings to be acceptable.

8.12 Evaluation Findings

F8.1 The applicant has met the requirements in 10 CFR 72.236(b). The applicant described the materials design criteria for SSCs important to safety in sufficient detail to support a safety finding.

F8.2 The applicant has met the requirements in 10 CFR 72.236(g). The properties of the materials in the storage system design have been demonstrated to support the safe storage of spent fuel.

F8.3 The applicant has met the requirements in 10 CFR 72.236(h). The materials of the spent fuel storage container are compatible with their operating environment such that there are no adverse degradation or significant chemical or other reactions.

F8.4 The applicant has met the requirements in 10 CFR 72.234(b). Quality assurance programs and control of special processes are demonstrated to be adequate to ensure that the design, testing, fabrication, and maintenance of materials support SSC intended functions.

The staff concludes that the MAGNASTOR® design adequately considers material properties, environmental degradation and other reactions, and material quality controls such that the design is in compliance with 10 CFR Part 72. This finding is reached on the basis of a review that considered applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

8.13 References

1. Billone, M.C., T.A. Burtseva, Z. Han, and Y.Y. Liu. 2014. "Effects of Multiple Drying Cycles on High-Burnup PWR Cladding Alloys, DOE Used Fuel Disposition Report," FCRD-UFD-2014-000052, ANL Report ANL-144/11.

2. Electric Power Research Institute (EPRI), “Effect of Hydride Reorientation in Spent Fuel Cladding – Status from Twenty Years of Research,” EPRI-3002016033, June 2020.

3. NRC, “Dry Storage and Transportation of High Burnup Spent Nuclear Fuel – Final Report,” NUREG-2224, Washington, DC, November 2020. ML20191A321.

4. NRC, “Control of Heavy Loads at Nuclear Power Plants – Resolution of Generic Technical Activity A-36,” NUREG-0612, Washington, DC, July 1980. ML070250180.

5. NRC, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities — Final Report,” NUREG-2215, Washington, DC, April 2020. ML20121A190.

Chapter 9 CONFINEMENT EVALUATION

The confinement review ensures that radiological releases from the storage system to the environment will be within the limits established by the regulations and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that might otherwise lead to gross ruptures.

Staff reviewed the information provided by the applicant and evaluated the following changes requested in amendment no. 11:

1. Addition of a new TC known as the lightweight MAGNASTOR® TC (LMTC).
2 Addition of a new CC design known as CC7.
3. Increased the maximum system heat load capacity when using the LMTC and CC7.
4. Addition of new loading patterns I, J, and K for the PWR basket assembly.
5. Addition of new loading patterns B, C, and D for the new BWR 89-assembly basket.
6. Addition of new loading patterns A, B, and C for the new BWR 81-assembly basket.
7. Addition of a new BWR fuel basket design permits loading up to 89 undamaged BWR fuel assemblies with increased heat load capacity.
8. Addition of a new BWR damaged fuel basket design with a capacity of up to 81 undamaged BWR fuel assemblies, which includes 12 DFC locations with increased heat load capacity.
9 Removed the previously licensed BWR basket configurations since they are bounded by the new 89 and 81 configurations; however, the SAR analysis and licensing drawings remain as they partially support the evaluations justifying the approval of the new 89 and 81 configurations.
10. Added a new DFC for BWR fuel.
11. Added new and revised previously approved drawings for the LMTC, BWR 89 fuel basket, BWR damaged fuel basket, and DFC.
12. Technical specification, appendix A revisions to include the new LMTC, BWR 89 fuel basket BWR damaged fuel basket, and DFC, including increased heat loads and loading patterns.
13. Technical specification, appendix B revisions to include the new LMTC, BWR 89 fuel basket, BWR damaged fuel basket, and DFC, including increased heat Loads and loading patterns.

Staff reviewed the information provided in this amendment application and finds that:

- the confinement design of the stainless-steel TSC remains unchanged, and the confinement performance is not affected by the proposed changes;
- the TSC confinement boundary components (TSC shell, bottom plate, closure lid, inner vent and drain port covers, and the welds that join these components) remain unchanged and continue their confinement functions; and
- the welded TSCs (TSC1 ~ TSC4) remain under the leaktight condition as addressed in the previous application approved by the NRC.

Staff determined that all the changes proposed in this amendment application do not impact the confinement design, boundary components, and performance. Therefore, staff concludes that the MAGNASTOR® TSCs meet the confinement requirements in 10 CFR 72.236 and will continue to fulfill the confinement acceptance criteria as listed in section 9.4, “Regulatory Requirements and Acceptance Criteria,” of NUREG 2215, “Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities.”

Staff also reviewed the editorial changes in SAR chapter 7, "Confinement Evaluation," chapter 9, "Operating Procedures," chapter 10, "Acceptance Criteria and Maintenance Program," and chapter 12, "Accident Analyses," and determined that these editorial changes do not affect the confinement evaluation and do not have negative impact to the confinement functions of the MAGNASTOR® cask system.

10.3 Findings

F9.1 NAC evaluated I confinement system of MAGNASTOR® casks for the proposed changes, and demonstrated that the design will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F9.2 On the basis of review of the proposed changes and the submitted documents, staff concludes that the proposed changes have no negative impact on the confinement system and that the MAGNASTOR® System (CoC 1031) continues to meet the confinement requirements of 10 CFR Part 72.

Chapter 10 RADIATION PROTECTION EVALUATION

The purpose of this review is to: (1) evaluate adequacy of the radiation protection capabilities of the MAGNASTOR® system for the proposed new contents and the new storage cask and TC designs to ensure that the NRC's regulations pertaining to direct radiation are met, (2) determine that the proposed changes in this amendment will maintain workers' exposure ALARA, and (3) ensure that the radiation doses to workers and to the general public still meet the regulatory limits during both normal, off-normal, and accident conditions. 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 estimated exposures for operations and storage are based on the PWR or BWR contents that result in the highest dose rates. Transfer cask exposures are based on the cask configuration documented in section 6.1.1 of the SAR. Similarly, the LMTC exposures are based on the configuration documented in section 5.12.2 of the SAR and assume a dry canister cavity with a supplemental (weld) shield in place for both the maximum and minimum radial lead thicknesses, 4-inch and 2.5-inch-thick lead shields, respectively.

10.1.2 Design Features

The principal radiation protection design is based on the placement of penetrations near the edge of the TSC lid to reduce operator exposure and improve access, and the use of the weld shield for work on and around the closure lid. Use of the weld shield reduces operator exposure during the welding, inspection, draining, drying and helium backfilling operations. The radiation exposure rates at various work locations within the vicinity of a single transfer and concrete cask were determined using the MCNP code and the NAC-CASC (a modified SKYSHINE-III version) code for the dose at the controlled area boundary for a hypothetical ISFSI (i.e., an array of concrete casks) as well as the dose rate as a function of distance from the ISFSI to the controlled area boundary. These codes generated bounding dose rate profiles at various distances from the transfer and concrete cask, which are used to estimate the operator exposures for loading and routine operations.

10.2 Occupational Dose Evaluations

The estimate occupational radiation exposures to personnel (person-rem) during the loading and transfer to pad for the MAGNASTOR® storage system are taken from NAC Report 71160-5053, Rev.0. Dose rates are taken from the high heat LMTC and CC shielding evaluations in 71160-5031 Rev.0, -5032 Rev.0, -5033 Rev. 0, and -5035 Rev. 0. Specifically, the evaluation estimates the occupational radiation exposures incurred during fuel loading, TSC sealing, TSC transfer, and cask pad placement. These estimates are input to the SAR radiation protection chapter.

10.2.1 Estimated Dose Due to Loading Operations

The applicant evaluated the estimated dose for loading the high heat LMTC using the same method and operations which produce table 11.3-1 and table 11.3-2 of the SAR. Adding all the subtasks, exposure duration and average dose rates, table 11.3-1 showed that the total person-mrem for loading operation of the PWR System resulted in 712 mrem. Adding all the subtasks, exposure duration, and average dose rates, table 11.3-2 showed that the total person-mrem for loading operation of the BWR resulted in 913 mrem. The exposure estimates for the PWR and BWR high heat LMTC systems are shown in table 11.3-3, "Estimated Person-mrem Exposure for Loading Operations of the High Heat LMTC System." Exposures due to LMTC loading operations are based on the uniform loading patterns (PWR Pattern I outlined in section 5.12.4 and BWR Pattern A outlined in section 5.14.4) which was modeled with higher than allowable total cask heat load. The estimated dose exposures are shown for both the maximum 4-inch and minimum 2.5-inch radial lead shield. The staff found that the estimated dose exposures envelop the dose exposures from the 37-assembly undamaged and damaged fuel PWR baskets as well as the 89-assembly undamaged BWR and 81-assembly damaged fuel BWR baskets. The NRC finds this acceptable because the number of persons allocated to task completion is generally the minimum number of actual operators required for the task and excludes supervisory, health physics, security, and other non-operating personnel.

10.2.2 Estimated Dose Due to Routine Operations

The applicant considered in the annual dose evaluations the tasks that are anticipated to be representative of an operational facility. Exposure due to certain events, such as clearing the material blocking the air vents, was also considered. Storage operation exposures for a 2x10 array of either PWR or BWR concrete casks loaded with TSCs containing bounding fuel assembly sources are presented in table 11.3-4, "Estimate of Annual Exposures Due to Routine Operations for a PWR 20-Cask Array." From table 11.3-4, the total person-rem average dose per cask resulted in 34 mrem. Table 11.3-5, "Estimate of Annual Exposures Due to Routine Operations for a BWR 20-Cask Array," showed that the total Person-mrem average dose per cask resulted in 32 mrem.

The staff reviewed the estimated dose due to routine operations and found them acceptable based on the fact that the applicant estimated of occupancy times for personnel involved in these functions, including the maximum expected total hours per year for any individual and total person-hours per year for all personnel. Also, the applicant estimated the annual collective doses associated with each major function and each radiation area and showed that the individual doses to workers are below the dose limits specified in 10 CFR 20.1201, "Occupational dose limits for adults."

10.3 Off-Site Dose Evaluation

The applicant stated that contributions from concrete casks to site radiation dose are either from the radiation emitted from the concrete cask surface (via skyshine) or a hypothetical release of surface contamination from the TSC. The applicant used NAC-CASC, a modified version of the SKYSHINE-III code, to calculate site boundary dose rates for either a single cask or cask array. See section 5.6 of the SAR for more detail on the shielding codes. The analysis presented in section 5.6.5 of the SAR estimates a total dose of less than 0.1 mrem at 100 meters from a design-basis concrete cask. The analysis demonstrates that the off-site radiological consequences from the release of TSC surface contamination is negligible, and all applicable regulatory criteria are met for an ISFSI array. ISFSI-specific allowable dose rates will be calculated on a site-specific basis to conform to 10 CFR Part 72. As documented in section 5.6.5 of the SAR, there is no significant site dose effect from the expected surface contamination of the system. There is no credible leakage from the system, and no significant effluent source can be released from the TSC contents because the TSCs are comprised of a welded shell, bottom plate, and lid structure and the vent and drain ports in the lid are covered by redundant welded plates.

The NRC staff reviewed the evaluations presented by the applicant and finds the reported dose rate calculations and the MCNP shielding analysis model for undamaged fuel, damaged fuel, and non-fuel hardware to be acceptable because site boundary and restricted area boundary dose rates are dominated by the total emissions from the side of the concrete cask, which can be characterized by the average dose. The applicant evaluations included both a single cask and a 2x10 array of casks for site exposure evaluations. Each cask in the array is assigned the maximum dose (surface current) source allowed by the cask loading tables. A combination of the maximum cask side and top dose cases provides for an estimate on the controlled area boundary exposure, since the different fuel types produce the highest cask surface dose components. Off-site dose calculations for amendment no. 0 had higher doses and bound the off-site dose rates for this amendment. Accident dose analyses are bounded by the off-site dose rates determined for amendment no. 0.

10.4 Radiological Consequences from Off-Normal Events

In section 12.1.2.5 of the SAR, the applicant states that there are no significant radiological consequences from off-normal events for the one-half of the air inlets blocked event. Personnel will be subject to an estimated maximum contact dose rate of 448 mrem/hr when clearing the inlet screens of a concrete cask containing a conservative 37 kW payload of PWR fuel. If it is assumed that a worker kneeling, with his hands at the inlet screens, would require 15 minutes to clear the screens, the estimated maximum extremity dose is 112 mrem.

For clearing the inlet screens of a concrete cask containing a conservative 35 kW payload of BWR fuel, the maximum contact dose rate and the maximum extremity dose are estimated to be 364 mrem/hr and 91 mrem, respectively. The whole-body dose in both the PWR and the BWR cases will be significantly less than the extremity doses. The staff finds the radiological consequences from off-normal events acceptable because dose rates were based on previously approved cask models in revision no. 5 (ML15180A364) without inlet shield, and bound dose rates calculated for the remaining concrete cask types (CC3, CC7), even those evaluated with higher heat loads, as all other cask types contain inlet shields as a required, not optional, component.

The staff finds this approach acceptable because it follows the ALARA principles. When used in accordance with its design, MAGNASTOR® maintains occupational radiation exposures ALARA while meeting overall system performance objectives. The following specific design features demonstrate the ALARA philosophy:

1) Material selection and surface preparation that facilitate decontamination,
2) Basket configurations that allow spent fuel loading using accepted standard practices,
3) Positive clean water flow in the transfer cask/TSC annulus to minimize the potential for contamination of the TSC surfaces during in-pool loading,
4) Passive confinement, thermal, criticality, and shielding systems that require no maintenance,
5) Thick steel and concrete shells in the storage system, and a steel/lead/neutron shield/steel, configuration in the transfer system,
6) Nonplanar cooling air pathways with respect to the spent fuel assembly source regions to minimize radiation streaming at the concrete cask inlets and outlets,

7) Provision of shield blocks below the vent and drain openings in the closure lid to minimize streaming,
8) Provision of quick-release devices for use on the transfer cask doorstops and retaining blocks, and
9) Optional use of remote, automated outlet air temperature measurement to reduce surveillance time.

10.5 ALARA

The onsite collective dose assessment estimates allow the user to perform ALARA evaluations on MAGNASTOR® implementation and use and to establish personnel exposure guidelines for operating personnel. The personnel exposure estimates associated with loading and routine operations are presented in table 11.3-1 through table 11.3-4 of the SAR. The estimated durations, task sequences, and personnel requirements are based on the MAGNASTOR® design features, operational experiences in loading systems of similar design, and operational and equipment improvements based on previous experience.

These estimates are provided to allow the user to perform ALARA evaluations on MAGNASTOR® implementation and use and to establish personnel exposure guidelines for operating personnel.

The staff found the ALARA evaluations acceptable because the applicant identified the collective and individual doses associated with all operations involved with placing one full storage container in the storage position according to the associated function. Also, the applicant provided estimates of the annual collective and individual doses by multiplying the single-storage container dose by the maximum annual placement rate of containers into storage. This estimation made by the applicant assumed that the same personnel will be involved in the same operations for each container to ensure that the doses not exceed those allowed by 10 CFR 20.1201(a).

10.6 Evaluation Findings

F10.1 The SAR sufficiently describes the radiation protection design bases and design criteria for the proposed changes to the MAGNASTOR® storage system.

10.2 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 requirements in 10 CFR 72.236(d).

F10.2 The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

Based on its review of the information presented in the SAR, the staff concludes that the design of the radiation protection system for the MAGNASTOR® storage system with the proposed changes described in section 6.1.1 of the SAR is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the MAGNASTOR® storage system will provide safe storage of spent fuel. This finding is based on a review that considered applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

Chapter 11 OPERATION PROCEDURES AND SYSTEMS EVALUATION

Chapter 9 of the MAGNASTOR® SAR describes the operating procedures for loading spent fuel, for removing a loaded TSC from a concrete cask, and for the wet unloading of fuel from a TSC. The procedures described in the SAR provide general guidance; the system user (reactor licensee) will develop more detailed, site-specific procedures for the actual loading, handling, transfer, storage, and unloading of the system.

The applicant made minor changes to the SAR chapter 9 for the MAGNASTOR® storage system. In addition to editorial changes unrelated to the changes requested in this amendment, NAC revised the operating procedures to incorporate the new LMTC, higher heat load fuel, and damaged PWR fuel, and added procedures for loading damaged BWR fuel.

11.1 Lightweight MAGNASTOR® Transfer Cask

NAC added section 9.3, “Loading MAGNASTOR Using LMTC”. The staff reviewed the operating procedures in section 9.3 for loading the LMTC and transfer to the MAGNASTOR® overpack and determined that the procedures are acceptable because they address TSC loading operations, including preparation of the LMTC and TSC, TSC fuel loading, TSC drying and backfilling, and TSC sealing operations. The staff verified that the LMTC loading operations are consistent with the approved operating procedures included in the FSAR Revision 12 (ML22187A045) for the MTC and PMTC. The staff also finds that the operating procedures include consideration of ALARA as required by 10 CFR 72.104(b). On these bases, the staff determined that the procedures are acceptable because they include the necessary steps for the described operations, are consistent with the technical analyses in the SAR, and maintain operations ALARA.

11.2 Evaluation Findings

F11.1 Section 11 of this SER assesses the radiological protection measures and operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the cask user/licensee.

F11.2 The staff concludes that the generic procedures and guidance for the operation of the MAGNASTOR® system comply with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the system will enable safe storage of spent fuel. This finding is based on a review that considered the applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

Chapter 12 CONDUCT OF OPERATIONS

The objective of the staff's review is to ensure that the applicant's SAR includes the appropriate acceptance tests and maintenance programs for the dry cask storage system. The applicant addressed this area in chapter 10 of the MAGNASTOR® system SAR, "Acceptance Criteria and Maintenance Program."

12.1 Acceptance Tests

Section 10.1.2 of the SAR describes the structural testing to be conducted on the load-bearing components of the TC and concrete cask, and the pressure testing of the TSC. The visual and non-destructive tests, and load testing of TCs are sufficiently inclusive to include the LMTC without modification since it uses the term TC rather than identifying a specific TC. NAC added section 10.1.2.6, "Pressure Testing of the LMTC," to provide specific pressure tests for the LMTC.

Following completion of the load tests for the LMTC, the neutron shield tank and the expansion tanks are hydrostatically tested. The minimum test pressure for the components is 20 psig (125% of the maximum operating pressure) for a 10 minute hold time, which is in accordance with the ASME BPV Code, Section VIII, Division 1. After completion of the 10 minute hold time, the LMTC is visually inspected for evidence of leakage and all accessible welds on the neutron shield structure are visually examined, dye penetrant tested, and accepted in accordance with the ASME BPV Code.

12.2 Maintenance Program

NAC updated table 10.2-1, "MAGNASTOR Maintenance Program Schedule" to include the same maintenance program activities and schedule for the LMTC as for the PMTC.

12.3 Evaluation Findings

F12.1 SSCs important to safety will be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function(s) they are intended to perform. The SAR identifies the safety importance of SSCs and presents the applicable standards for their design, fabrication, and testing in accordance with 10 CFR 72.82(d), 10 CFR 72.122(a), 10 CFR 72.122(f), 10 CFR 72.124(b), 10 CFR 72.162, 10 CFR 72.234(b), and 10 CFR 72.236(b), (g), (j), and (l).

F12.2 The applicant or licensee, as appropriate, will examine and test, as needed, the MAGNASTOR® SSCs and features to ensure they do not exhibit any defects that could significantly reduce their confinement effectiveness. Chapter 12 of the SAR describes this inspection and testing, in compliance with 10 CFR 72.236(l).

The staff concludes that the conduct of operations program complies with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the conduct of operations program provides reasonable assurance that the MAGNASTOR® storage system will allow for the safe storage of spent fuel throughout its licensed or certified period of storage. This finding is reached on the basis of a review that considered applicable regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

Chapter 15 QUALITY ASSURANCE EVALUATION

There were no requested changes to NAC's quality assurance program, and none of the changes requested by NAC affect the quality assurance program.

Chapter 16 ACCIDENT ANALYSES EVALUATION

NAC made minor changes in chapter 12 of the SAR to ensure the new CC7, LMTC, fuel baskets, and contents are incorporated into the evaluation of accidents. Since NAC showed in its SAR, and as NRC describes above in this SER, the evaluation results for these changes are either similar to or bounded by previous amendments. Therefore, the revisions requested by NAC do not affect the accident analysis evaluation for the system and do not alter the staff's previous evaluation of the accident analyses for the MAGNASTOR® storage system.

Chapter 17 TECHNICAL SPECIFICATIONS AND OPERATING CONTROLS AND LIMITS EVALUATION

NAC incorporated the changes listed in the Summary section above and its associated changes in the technical specifications. The staff reviewed the technical specifications and the associated operating controls and limits to ensure they meet the requirements of 10 CFR Part 72. The staff evaluation of the adequacy of the technical specifications is provided in chapters 2 through 8, above.

Chapter 18 CONCLUSIONS

The staff has performed a comprehensive review of the amendment application, during which it evaluated the following requested changes to the MAGNASTOR® storage system:

- add a seventh concrete overpack (CC7) and a lightweight MAGNASTOR® transfer cask (LMTC)
- increase the maximum heat load for the system when using CC7 and the LMTC
- new loading patterns
- add new 81 and 89 BWR spent fuel basket designs, and associated loading patterns
- remove existing 87-assembly and 82-assembly BWR basket designs
- add a new BWR damaged fuel basket design with a capacity of up to 81 undamaged BWR fuel assemblies
- add a new DFC for BWR fuel

Based on the statements and representations provided by the applicant in its amendment application, as supplemented, the staff concludes that the changes described above to the MAGNASTOR® storage system do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. amendment no. 11 for the MAGNASTOR® storage system should be approved.

Issued with CoC no. 1031, amendment no. 11,
on September 12, 2023.