FINAL SAFETY EVALUATION REPORT
NAC INTERNATIONAL, INC.
MAGNASTOR® STORAGE SYSTEM
DOCKET NO. 72-1031
REVISION TO AMENDMENT NO. 9

SUMMARY

On January 21, 2022 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML22021B305), the U. S. Nuclear Regulatory Commission (NRC) issued a Severity Level IV violation to NAC International Inc., (NAC) for performing an improper change to the safety analysis report (SAR) via the regulations in Title 10 of the *Code of Federal Regulations* (10 CFR), section 72.48, "Changes, Tests, and Experiments." The violation identified that NAC failed to obtain an amendment to Certificate of Compliance No. 1031 for changes to the SAR to remove the requirements in American Concrete Institute (ACI) Specification ACI 318 "Building Code Requirements for Structural Concrete" for fabrication of the MAGNASTOR® concrete cask lid. Technical Specification A4.2, paragraph 3, in Amendment No. 9 requires that the "American Concrete Institute Specifications ACI 349, ["Code Requirements for Nuclear Safety Related Concrete Structures,"] and ACI 318 govern the CONCRETE CASK design and construction, respectively." NAC has submitted an amendment request to correct the violation.

By application dated March 18, 2022 (ML22077A769), as supplemented on April 18, 2022 (ADAMS Accession No. ML22108A197), and August 4, 2022 (ADAMS Accession No. ML22216A110), NAC requested a revision to Amendment No. 9 for the Model No. MAGNASTOR® storage cask. NAC proposed to revise the certificate of compliance by adding a definition for the storage cask and concrete cask lid, revising the definition of concrete cask with conforming changes throughout appendix A of the technical specifications, and providing alternate fabrication criteria and techniques in technical specification A4.2 for the concrete cask lid.

The NRC staff reviewed the request for the revision to the certificates of compliance using guidance in NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities - Final Report," dated April 2020. 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 and the technical specifications, the staff concludes that the requested changes meet the requirements of 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."

MAGNASTOR® Storage System

General Description and Operational Features

The MAGNASTOR® system used to store spent fuel consists of a transfer cask, storage overpack, and a welded stainless steel transportable storage canister (TSC) that contains the spent fuel. 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 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.

Storage Overpack

The MAGNASTOR® storage system includes variations of the concrete overpack, with differing dimensions to accommodate slightly different TSC designs. The concrete cask body is a right circular cylinder with a reinforced, structural concrete shield wall and a carbon steel inner liner and base. The reinforcing steel rebar is encased within the concrete. The concrete overpack contains inlet air vents at the bottom and outlet air vents at the top for convective air flow. The convective air flow removes decay heat from the TSC shell. Amendment No. 9 incorporates the sixth version of the concrete cask (known as CC6 in the NAC final safety analysis report [FSAR]), which includes the variation of the concrete cask lid shown on Drawing No. 71160-664.

Transportable Storage Canister

The stainless steel TSC assembly holds the fuel basket structure and confines the contents. There are two TSC lengths that contain the fuel baskets to hold pressurized-water reactor (PWR) and boiling water reactor (BWR) fuel assemblies. The TSC is defined as the confinement boundary during storage.

There were no changes to the TSC in this amendment.

Drawings

NAC did not submit any revised drawings for this amendment.

Proposed Changes

NAC proposed to revise the certificates of compliance by adding a definition for the storage cask and concrete cask lid, revising the definition of concrete cask with conforming changes throughout appendix A of the technical specifications, and providing alternate fabrication criteria and techniques in Technical Specification A4.2 for the concrete cask lid. The new definitions treat the concrete cask and the concrete cask lid as separate components and allow separate fabrication requirements in the technical specifications. The storage cask is defined as the concrete cask and the concrete cask lid. The conforming changes throughout the appendix are to replace concrete cask with storage cask, when the action is on a loaded cask, to ensure that it includes both the lid and the cask body.

NAC revised technical specification appendix A, section 4.2 (technical specification A4.2) for the concrete cask lid includes the minimum concrete density, the allowable methods for measuring the density of the cask concrete lid, and requirements for the concrete mix, placement, and curing methods used in the construction of the concrete cask lid.

The technical specification changes are to ensure that the concrete in the cask lid has the physical properties (e.g., density, dimensions, geometry, internal structure) needed to perform its radiation shielding safety function as intended, while eliminating the requirement that the concrete in the cask lid comply with the ACI 349 and ACI 318. It should be emphasized that these ACI standards will remain in the technical specification for the design and fabrication of the concrete cask body.

The NRC staff's review of the application for revision to Amendment No. 9 addressed the following items for the concrete cask lid: construction criteria, structural function, concrete degradation mechanisms, and the radiation shielding evaluation accounting for shrinkage of the concrete in the cask lid. The staff's evaluation is addressed in the safety evaluation report (SER) sections below.

Concrete Cask 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 the ASTM International 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 NRC staff noted that the MAGNASTOR® technical specifications for Amendment No. 9 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. As discussed in more detail below, because the concrete used in the construction of the concrete cask lid has a radiation shielding safety function but no structural safety function, 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 American National Standard Institute and 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/ANS-6.4 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 construction of the concrete cask lid is 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 the 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.

Concrete Cask Lid Structural Function

The storage cask lid provides protection to the TSC from the external environment including any postulated tornado missiles strike and provides protection against sky shine radiation. The lid configuration used in MAGNASTOR Amendment No. 9, which is shown in Drawing No. 71160 664, Rev. 1P, Sheet 2 of 2, is constructed of 1-inch-thick carbon steel.

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 (ADAMS Accession No. ML090350589). In that SER, the staff concurs with the SAR 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 top plate is adequate in preventing plate perforation with a factor of safety of 1.15 (0.75/0.65 =1.15). The 1-inch-thick carbon steel upper plate in the lid on concrete cask number 6 would yield a higher factor of safety for the same impact.

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.

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.

In its March 18, 2022 (ADAMS Accession No. ML22077A769), NAC included an evaluation of the potential for concrete shrinkage. NAC evaluated the potential effects that radial concrete shrinkage would have on the lid's ability to perform its radiation shielding safety function. NAC's evaluation calculated the expected radial shrinkage around the edge of the concrete cask lid. The NRC staff compared NAC's calculation of the expected radial shrinkage around the edge of the lid (0.02 inches for the smaller diameter concrete in the lids) to the data regarding maximum concrete shrinkage from NUREG-2214 and found the licensee's calculation of the expected radial concrete shrinkage to be acceptable for the initial storage term. The staff determined that since the as-fabricated lids meet the proposed technical specification criteria for the concrete in the lid, which included 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," they are sufficient to ensure that the NAC calculation methodology for expected radial shrinkage is a credible estimate of the actual shrinkage behavior that may be expected during the initial storage term.

NAC also evaluated top surface dose rate profiles associated with conservative hypothetical radial gaps (for concrete encased in steel) 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 NAC, as discussed above. The NRC staff's review of the NAC's radiation shielding evaluation for calculating the top surface dose rates and NRC's estimate for shrinkage of the larger concrete lid in Amendment No. 9 is documented in the "Shielding Evaluation" section of the SER section below.

Other Concrete Degradation Mechanisms

Over time, the concrete cask lid may be prone to other degradation mechanisms, in addition to shrinkage, that 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, 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 CC6 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 fabrication methods used for the five as-built concrete cask lids is sufficient to ensure that the concrete in the lid will maintain the physical properties needed to adequately perform its radiation shielding safety function.

Shielding Evaluation

NAC performed an evaluation to estimate 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 NAC's Supplemental Report No. W220318t024202, *"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 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. Staff notes that the concrete diameter in the lid for Amendment No. 9 is approximately 110 inches, which yields an estimated 0.028-inch-radius gap due to shrinkage of the concrete in the lid.

NAC performed dose rates analysis taking FSAR revision 0 (ADAMS Accession No. 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 FSAR.

NAC 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.

NAC's use of a 0.04- and 0.08-inch radial gap is well in excess of the 0.028-inch maximum expected radial gap, previously discussed in the first paragraph of this portion of the SER. In its analysis, NAC 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 to impact either occupational doses or site-boundary dose limits.

The staff reviewed the methodology employed by NAC and found them acceptable based on the fact that they used a radial gap that is larger than the estimated 0.028 in. gap, showing that it would not increase either the occupational doses or site-boundary limits calculated by NAC in any previous analysis. The staff agrees that the results fall within the statistical uncertainty band of the original solutions, and therefore no additional offsite or occupational dose analyses are needed. The staff also found acceptable the use of MCNP based on facts that this computer code has commonly been used previously in NRC approvals for the MAGNASTOR® system and is appropriate for this evaluation. NRC notes that the dose rate increase for this lid in which the concrete is not encased in steel would be bounded by the analysis above, because the location of the shrinkage for the concrete lid in Amendment No. 9 is further from the fueled area of the canister, which has lower dose rates.

The staff concludes that the shielding and radiation protection design features of the MAGNASTOR® system, including the changes to the requirements in appendix A of the technical specifications for the concrete cask lid, are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria continue to be satisfied. The evaluation of the concrete 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 applicable regulations, appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's evaluations, and acceptable engineering practices.

Conclusion

Based on the discussion above, the NRC concludes that the revised technical specifications in appendix A are sufficient to minimize concrete shrinkage and degradation such that the dose to any real individual beyond the controlled area and the occupational doses from the storage cask are bounded by previous analyses. Therefore, 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® system do not affect the ability of the storage cask system to meet the requirements of 10 CFR Part 72. Revision 1 to Amendment No 9 for the MAGNASTOR® system should be approved.

Issued with certificate of compliance no. 1031, amendment no. 9, revision no. 1, on September 12, 2023.