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

**SAFETY EVALUATION REPORT
DOCKET NOS. 72-1031 and 72-44
EXEMPTION REQUEST FOR
ARIZONA PUBLIC SERVICE COMPANY
PALO VERDE NUCLEAR GENERATING STATION
INDEPENDENT SPENT FUEL STORAGE INSTALLATION**

SUMMARY

On January 21, 2022 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML22021B305), the NRC issued a Severity Level IV violation to NAC International (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 Nos. 0 – 9 requires that the "American Concrete Institute Specifications ACI 349 and ACI 318 govern the CONCRETE CASK design and construction, respectively." NAC has submitted an amendment request to correct the violation.

By application dated May 23, 2023 (Agencywide Documents Access and Management System (ADAMS) No. ML23143A391), Arizona Public Service Company (APS or the licensee) requested an exemption under 10 CFR 72.7 for the Palo Verde Nuclear Generating Station (PVNGS) Independent spent fuel storage installation (ISFSI) to load and store spent fuel in MAGNASTOR® storage casks with lids that were not designed and fabricated in accordance with Technical Specification A4.2 as specified in Amendment No. 7 of the NAC Certificate of Compliance No. 1031 for the MAGNASTOR® storage system. Specifically, APS requested an exemption from the requirements of 10 CFR 72.212(a)(2), 10 CFR 72.212(b)(3), 10 CFR 72.212(b)(5)(i), 10 CFR 72.214, and the portion of 10 CFR 72.212(b)(11) that requires compliance with the terms and conditions set forth in the certificate of compliance for each spent fuel cask used by an ISFSI general licensee. These provisions require storage of spent nuclear fuel under a general license in dry storage casks approved under 10 CFR Part 72.

APS uses NAC's MAGNASTOR® storage cask utilizing Certificate of Compliance No. 1031, Amendment No. 7 for dry storage of spent fuel assemblies at the PVNGS ISFSI. APS stated that it has 10 casks to be loaded with lids fabricated by NAC that do not meet the requirement in Technical Specification, Appendix A, A4.2 to be designed and fabricated in accordance with the American Concrete Institute Specifications ACI 349, "Code Requirements for Nuclear Safety Related Concrete Structures," and ACI 318, "Building Code Requirements for Structural Concrete," which according to the technical specifications govern the concrete cask design and construction, respectively. APS has a loading campaign scheduled at the PVNGS ISFSI and is unable to load those 10 casks due to NAC's non-compliance. APS, therefore, submitted an exemption request to allow for the 10 casks to be loaded notwithstanding the NAC noncompliance.

Enclosure

This safety evaluation report (SER) documents the NRC staff's review and evaluation of APS' exemption request for PVNGS ISFSI. Based upon the staff's evaluation, the NRC has determined that, pursuant to 10 CFR 72.7, the exemption is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest.

A. Authorized by Law

The Commission has the legal authority to issue exemptions from the requirements of 10 CFR Part 72 as provided in 10 CFR 72.7. Issuance of this exemption is consistent with the Atomic Energy Act of 1954, as amended, and not otherwise inconsistent with NRC regulations or other applicable laws. Therefore, issuance of the exemption is authorized by law.

B. Will Not Endanger Life, Property or the Common Defense and Security

In support of its exemption request, APS provided NAC International's request for MAGNASTOR® Amendment No. 12, dated January 24, 2022 (ADAMS Accession No. ML22024A374), as supplemented on March 18, 2022 (ADAMS Accession No. ML22077A769) and April 18, 2022 (ADAMS Accession No. ML22108A197), which includes the technical basis for removing the reliance on ACI 318 and ACI 349 for design and fabrication of the concrete in the storage cask lids for the NAC certificate of compliance for the MAGNASTOR® system. While this exemption request is specific to the casks at PVNGS, the technical basis provided by NAC applies to the certificate of compliance for the MAGNASTOR® system for which APS seeks an exemption. Therefore, the technical basis is useful for this exemption request. In its March 18, 2022, supplement, NAC requested that all MAGNASTOR® Amendments be revised to include the same technical specification revision[1]. The staff reviewed APS' exemption request for the Palo Verde ISFSI to determine whether the proposed exemption will endanger life, property, or the common defense and security. The staff followed the guidance of NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities - Final Report," issued in April 2020.

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) which 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.

[1] Revisions and amendments of the certificate of compliance are not considered in this exemption.

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.

Concrete Cask Lid Construction

In its April 28, 2023 (ML22118B162), supplement to the exemption issued on May 13, 2023 (ML22111A282), APS stated that the cask lids approved in the previous exemption met the changes NAC proposed for MAGNASTOR Amendment No. 12 in Technical Specification A4.2. In the current exemption request, APS alluded to meeting the proposed changes in Technical Specification A4.2 but did not specifically state that its current set of lids meets those proposed technical specifications. Therefore, the NRC staff is making the following controls for the concrete used in the concrete cask lid from Technical Specification A4.2 a condition of this exemption:

- 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. 7 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 not a 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. APS stated that the 10 already-fabricated lids for casks number 176 through 185 meet the proposed TS requirements for the proposed revision to MAGNASTOR® Amendment No. 7.

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 Nuclear Society an American National Standard (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 lid to the storage cask for Amendment No. 7 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 concrete cask lid has a ¾-inch-thick (minimum) carbon steel plate that spans the entire opening. The complete lid assembly is bolted to the top of the concrete cask. Drawing No. 71160-561, Rev. 9 Sheet 3 of 5 Section D-D shows a cross-section of a cask lid assembly with the alternate arrangement shown in Drawing 71160-561, Rev. 9 Sheet 5 of 5 Section E-E. The third lid configuration with a 1-inch-thick carbon steel cover plate, approved in Amendment No. 9, is not applicable to this exemption.

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 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, is adequate in preventing plate perforation with a factor of safety of 1.15 (0.75/0.65 =1.15).

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, submittal requesting revision of MAGNASTOR® Amendment No. 7, 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 an expected radial gap around the edge of the concrete cask lid due to shrinkage. The NRC staff compared NAC’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 NAC’s calculation of the expected radial gap due to 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,” are sufficient to ensure that the NAC calculation of 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 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 is documented in 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 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 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 over the initial storage term.

Given that the concrete in this cask lid design is encased in carbon steel, there is very little potential for intrusion of significant water, moisture, and dissolved compounds into the concrete 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 FSAR, 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 that adequate concrete placement and curing standards, such as those in ACI 349 and ACI 318, are followed. While the exemption request eliminates these ACI standards, the staff confirmed that the specification of ASTM C94 for ready mixed concrete used in the as-built lids and the additional care utilized during fabrication that the concrete was 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 these 10 storage casks.

Considering the potential degradation mechanisms, the staff determined that fabrication methods used for the 10 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.

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.

For Amendment No. 12, 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 0.02-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 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 0.02 in. gap, showing that the increase in dose rate is not significant enough that it would 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.

The staff concludes that the shielding and radiation protection design features of the MAGNASTOR® system, which include the 10 lids APS listed in requested exemption from the requirements in TS A4.2 for the concrete in the lid, are in compliance with the proposed Technical Specification A4.2, and with the dose rate requirements in 10 CFR Part 72. 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, NAC’s analyses, the staff’s evaluations, and acceptable engineering practices.

Based on these evaluations, the staff concluded that granting this exemption will be consistent with the requirements of 10 CFR Part 72 and will not endanger life or property or common defense and security.

C. Otherwise in the Public Interest

In determining whether granting the exemption is in the public interest, the staff considered the no-action alternative of denying the exemption request. Denial of the exemption request would cause APS to postpone loading of spent fuel in storage casks with non-compliant lids until it is approved in a revision to Amendment No. 7 for Certificate of Compliance No. 1031 or compliant lids could be fabricated and delivered to the PVNGS ISFSI.

For casks to be loaded, replacing the non-compliant lids would delay the upcoming loading campaign and cause APS to reduce open space in the spent fuel pool below prudent operating reserves. In addition, replacing the non-compliant lids would not improve the ability of the storage cask to perform its intended safety function, since the concrete in the MAGNASTOR® cask lids is only relied on for providing shielding and not relied on to provide a structural function.

The staff reviewed the information provided by APS and, based upon that information and the staff's technical review, concludes that granting the requested exemption is in the public interest.

D. Environmental Consideration

The NRC staff also considered in the review of this exemption request whether there would be any significant environmental impacts associated with the exemption. The NRC staff determined that this proposed action fits a category of actions that do not require an environmental assessment or environmental impact statement. Specifically, the exemption meets the categorical exclusion in 10 CFR 51.22(c)(25).

Both ACI 318 and ACI 349 state the design and fabrication criteria in the codes apply design and construction of concrete that serves a structural function. However, since Technical Specification A4.2, maintained the requirement that "American Concrete Institute Specifications ACI 349 and ACI 318 govern the CONCRETE CASK design and construction, respectively," NAC, as the fabricator, was still responsible for the code inspection or surveillance requirements associated with determining the minimum concrete strength contained in as-fabricated lids as required by these two codes. Since the concrete in the cask lid does not serve a structural function, it is not required to have a minimum strength by either ACI 318 or ACI 349, and therefore there is no criteria in either ACI code for comparison with the post-fabrication inspection results for the concrete strength. Therefore, granting this exemption from 10 CFR 72.212(a)(2), 72.212(b)(3), 72.212(b)(5)(i), 72.212(b)(11), and 72.214 only relieves APS from the inspection or surveillance requirements associated with ensuring that the concrete in the MAGNASTOR® lid meets the fabrication requirements of ACI 318 and ACI 349. A categorical exclusion for inspection or surveillance requirements is provided under 10 CFR 51.22(c)(25)(vi)(C) if the criteria in 10 CFR 51.22(c)(25)(i)-(v) are also satisfied. In its review of the exemption request, the NRC staff determined, as discussed above, that, under 10 CFR 51.22(c)(25):

(i) granting the exemption does not involve a significant hazards consideration because granting the exemption neither reduces a margin of safety, creates a new or different kind of accident from any accident previously evaluated, nor significantly increases either the probability or consequences of an accident previously evaluated,

(ii) granting the exemption would not produce a significant change in either the types or amounts of any effluents that may be released offsite because the requested exemption neither changes the effluents nor produces additional avenues of effluent release.

(iii) granting the exemption would not result in a significant increase in either occupational radiation exposure or public radiation exposure, because the requested exemption neither introduces new radiological hazards nor increases existing radiological hazards,

(iv) granting the exemption would not result in a significant construction impact, because there are no construction activities associated with the requested exemption, and,

(v) granting the exemption would not increase either the potential or consequences from radiological accidents such as a gross leak from the closure welds, because the exemption neither reduces the ability of the closure welds to confine radioactive material nor creates new accident precursors at the Palo Verde ISFSI.

Accordingly, this exemption meets the criteria for a categorical exclusion in 10 CFR 51.22(c)(25)(vi)(C).

CONCLUSION

Based on this evaluation and the statements and representations provided by APS in its exemption request, the NRC staff concludes that the proposed action to exempt the 10 reference cask lids from the current Technical Specification in A4.2 is authorized by law and will not endanger life, property, or the common defense and security, and is otherwise in the public interest and, therefore meets the exemption requirements in 10 CFR 72.7.