May 2019

Study of the ACI 349.3R-02 Tier 2 (i.e., Section 5.2.1) Criteria Impacts on Dose Rates for Several Spent Nuclear Fuel Dry Storage System Designs

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Purpose: To demonstrate whether or not, for the purposes of aging management for spent nuclear fuel dry storage systems, inspections that use the Tier 2 concrete evaluation criteria in American Concrete Institute (ACI) 349.3R-02 may be sufficient to ensure adequate performance of the shielding function of the dry storage systems during the 20- to 60-year period of extended operation, or if additional activities, such as dose rate surveys and monitoring, are needed as part of an aging management program.

This study was performed in support of responses to public comments received on NUREG–2214, "Managing Aging Processes In Storage (MAPS) Report, Draft Report for Comment" (Agencywide Documents Access and Management System (ADAMS) Accession No. ML19072A016).

INTRODUCTION

Guidance for the review of applications to renew the licenses for independent spent fuel storage installations (ISFSIs) using dry storage technology and the certificates of compliance (CoCs) for spent nuclear fuel dry storage systems is described in NUREG–1927, Revision 1, "Standard Review Plan for Renewal of Specific Licenses and Certificates of Compliance for Dry Storage of Spent Nuclear Fuel." The staff is also publishing a related document, NUREG–2214, "Managing Aging Processes In Storage (MAPS) Report," that provides guidance and the technical basis for the safety review of renewal applications for specific licenses of independent spent fuel storage installations and CoCs for spent nuclear fuel dry storage systems. The report includes examples of aging management programs (AMPs) that are considered generically acceptable for addressing credible aging effects and ensuring the licensed facility's or the certified system's design bases will be maintained.

For dry storage systems that use reinforced concrete components, the guidance in NUREG–1927 and NUREG–2214 refers to ACI 349.3R-02, "Evaluation of Existing Nuclear Safety-Related Concrete Structures," for the inspection evaluation criteria. In particular, the Tier 2 criteria from that standard, "Acceptance after review," are recommended as AMP acceptance criteria. It is noted, however, that these criteria only relate to the structural capability, or function, of the component.

The concrete also provides an important shielding function for the spent nuclear fuel storage system. The guidance in NUREG–2214 also includes performance of dose rate surveys and related criteria as part of a means to ensure the concrete component continues to adequately perform its shielding function. It has been considered that inspections using the ACI 349.3R criteria may be sufficient to also ensure that the shielding function of the reinforced concrete component of the dry storage system is adequately maintained. Thus, this study was conducted to demonstrate whether or not that is possible, which would thus alleviate the need for additional actions and criteria specifically related to the shielding function in an AMP.

The premise of this study is that the 10 CFR 72.104 (and 10 CFR 72.236(d)) analyses and the dose rate limits in the technical specifications, if any, for the systems are based on the design basis source terms at the time of system loading and that separate analyses and limits were determined for these systems for the different design basis contents (e.g., PWR vs. BWR, low burnup vs. high burnup). Thus, only calculations that showed that dose rates after 20 years of storage, considering the degree of concrete degradation allowed by the ACI 349.3R inspection evaluation criteria, would be less than dose rates at the time of loading were considered to demonstrate that the ACI 349.3R criteria are sufficient to ensure the storage system's shielding function will be adequately maintained during the period of extended storage.

Given the assumptions and simplifications that were used in the calculations for this study, the dose rate values themselves should not be viewed as estimates of actual storage system dose rates. This study focuses only on the relative changes in dose rates from the time of system loading (which in this study is same as the minimum cooling time allowed in the CoC, including the technical specifications, or the cooling time in the design basis shielding analysis) to a time 20 years later, representing 20 years in dry storage. The staff considered that the assumptions and simplifications used in the calculations would not significantly influence the relative change in dose rates between these two times. This premise is important to understanding the information, including the calculation results, presented in this report.

ACI 349.3R TIER 2 EVALUATION CRITERIA

The standard's Tier 2 criteria identify conditions that, if exceeded, should be considered unacceptable and in need of further technical evaluation. These criteria include:

- Appearance of leaching or chemical attack
- Areas of abrasion, erosion, cavitation degradation
- Drummy areas exceeding the cover concrete thickness in depth
- Popouts and voids less than 2 inches in diameter or equivalent surface area
- Scaling less than 1-1/8 inches in depth
- Spalling less than 3/4 inches in depth and 8 inches in any dimension
- Corrosion staining on concrete surfaces
- Passive cracks less than 0.04 inches (1 mm) in maximum width
- Passive settlements or deflections within the original design limits

Some of the criteria, such as those related to leaching and corrosion staining, do not appear to involve loss of material in any significant way that would affect shielding. These indications are anticipated to be precursors to any such material loss. Therefore, the impact of degradation related to these criteria would negligibly impact dose rates. Criteria that involve loss of material or would result in streaming paths through the concrete are of more significance for the shielding function.

Loss of Material

To assess the impact of loss of shielding function as a result of concrete aging, this analysis considered the criterion regarding drummy concrete. While the drummy area criterion is not necessarily a criterion involving material loss, it may be an area where the concrete is not properly mixed or cured. Thus, the properties in such areas could have an impact on dose rates. The staff understands that having drummy areas is a fabrication issue and is not an aging or degradation issue. However, it may serve to capture the effects of aging and degradation mechanisms. The cover concrete thickness is the depth into the concrete from the surface where the rebar is placed. Based on the standard ACI 318-11, "Building Code Requirements for Structural Concrete," this depth is expected to typically be 2 inches for spent nuclear fuel dry storage systems.

While the evaluation criteria also include a criterion regarding settlements and deflections within the original design limits, the staff does not expect that the original design limits for these systems will exceed the cover concrete depth. Thus, the staff expects the criterion regarding settlements and deflections to be bounded by the drummy area criterion, as applied for this shielding evaluation. The cover concrete depth also is greater in depth than the allowable depth of scaling in the ACI 349.3R Tier 2 criteria. Thus, to bound the effects of material loss, this analysis used the drummy concrete criterion, assuming a loss of concrete to the allowable depth defined in the standard's Tier 2 criteria. For cases where the conservative configuration with the cover concrete entirely gone resulted in dose rates that exceeded the dose rates calculated for the system at the time of loading, additional analyses were performed that considered other configurations with a lesser amount of conservatism. These configurations include one with the drummy area having half of the normal concrete density (rather than assuming the concrete in the drummy area is completely absent) and another with the reduced cover concrete density in combination with a loss of concrete occurring to a depth of the allowable scaling (1-1/8 inches in depth). Although this latter case is less bounding than the case where the cover concrete is entirely gone, it is still considered to be consistent with the standard's Tier 2 criteria.

It is recognized that, particularly for NUHOMS-type systems, the reinforced concrete is exposed to air on both the exterior and the interior surfaces (though the interior surfaces are in a sheltered environment). Thus, the cover concrete and other issues could also affect the interior concrete surfaces of the NUHOMS-type systems. Conversely, for most vertical overpack systems, the interior concrete surface is covered by a steel shell. This analysis only considered the potential degradation of the exterior concrete surfaces, assuming no degradation of the interior surfaces.

Streaming Paths

The ACI 349.3R criterion related to passive cracks represents a potential for streaming paths to exist in the concrete. However, in terms of this criterion, the staff expects that the allowable crack size should not have an effect on dose rates. The crack width is quite small, and the staff does not expect that a crack of that size would afford a streaming path of significance that would be detectable using a radiation survey meter of any typical dimension. Furthermore, any crack in the reinforced concrete of any depth would follow a torturous path through the concrete as a result of the presence of aggregate in the concrete and other relevant concrete properties. Thus, there will be significant concrete material directly beneath where a crack appears on the concrete structure's surface, which will provide shielding. Plus, the tortuous nature of the crack's path will also ensure any radiation (gamma or neutron) will interact in and be shielded by the concrete surrounding the crack. Based on the preceding considerations, the staff concludes that the cracks allowed by the ACI 349.3R criteria will not affect dose rates from the reinforced concrete structures; therefore, the staff did not perform any calculations to assess dose rates from cracks in these structures.

ANALYZED SYSTEMS AND MODELS

First, the staff identified all the certified dry storage systems (see 10 CFR 72.214) that have reinforced concrete components that are directly exposed to the outside environment. These systems only include those owned by Energy Solutions, Orano, and NAC; the Holtec systems use plain concrete encased in a steel shell, or the concrete component is below grade, as in the UMAX system (excluding consideration of the concrete portions of the design that are at or above grade). The staff identified the various dimensional and material properties of the different storage system components (including minimum cover concrete thickness), focusing on the storage overpack, or module, and the canister used to store the spent nuclear fuel in the overpack or module. The staff also considered the different design basis spent fuel contents for each of these systems. Based on consideration of these characteristics, the staff selected the MAGNASTOR, NAC-MPC, and NUHOMS systems to include in the analysis.

For the NUHOMS system, since there are several certified systems of this name (e.g., Standardized, Advanced), the NUHOMS system in this analysis is a composite design that is based on characteristics that the staff expects to provide bounding dose rates or to bound the material loss effects on dose rates. Also, early NUHOMS module designs included a gap of several inches between adjacent modules. The more recent designs have no, or very little, gap between adjacent modules. Both early and recent models include an end shield wall for the modules that are on the end of a row of modules and a rear shield wall for single rows of modules. Since the early NUHOMS modules have significant gaps between neighboring modules, the analysis has two separate models for the NUHOMS system. One model uses only the concrete thickness for a single module. This model represents or bounds the early NUHOMS modules. The second model includes the end shield wall's concrete and represents or bounds the more recent NUHOMS modules. The radiation source terms for these different models is kept consistent with what were allowed contents for the respective module designs (e.g., spent fuel with lower burnup in the earlier designs and higher burnup in the recent designs).

COMPUTER CODES

The staff used the SCALE5.1 computer code's SAS2H sequence to calculate the radiation source terms for the selected spent nuclear fuel contents. The SAS2H sequence has been available in the SCALE code system for many years and has been a reliable source term calculation tool. While it is not in more recent versions of the SCALE code, a sequence that is basically the same as SAS2H continues to be included in the SCALE code. SAS2H is a one-dimensional depletion code, which is adequate for the purposes of determining the radiation source term of spent nuclear fuel for shielding calculations.

The staff is aware of the limitations associated with using the SAS2H code for evaluation of radiation source terms from high burnup fuel in shielding analyses for spent nuclear fuel dry storage systems. In using SAS2H for this particular evaluation, the staff remains cognizant of those limitations. However, the focus of this evaluation is on the relative change in dose rates for spent fuel of a particular burnup and enrichment resulting from the loss of concrete and the change in post-irradiation cooling time from the time of loading into the cask to a time 20 years later in dry storage. The identified limitations of SAS2H for source term calculations occur in the depletion analysis in the code. As far as the decay analysis, the code is the same as other more recent codes such as TRITON. Thus, for a particular comparison, the burnup and enrichment are constant, and any uncertainties in the high burnup fuel source terms derived with SAS2H should be consistent for the source term at the time of cask loading and for the source term at 20 years after cask loading. Therefore, the uncertainties should not influence the relative change in dose rates between these two times.[1]

The staff used the MCNP6, Version 1.0 computer code to calculate gamma and neutron dose rates on the selected dry storage systems' surfaces. The MCNP code is a code that has wide use throughout the nuclear industry. The code is capable of analyzing 3-dimensional problems, including streaming paths. It also includes validated cross section libraries. The code's variance reduction techniques enable improved efficiency of shielding calculations for deep penetration problems such as spent nuclear fuel dry storage systems. The analysis uses the continuous-energy cross sections in the code to calculate the dose rates and makes use of some of the code's variance reduction capabilities.

SHIELDING MODELS

Shielding Configuration

Models were constructed for the selected systems to calculate dose rates at the time of system loading using a selected 'design basis' source term (burnup, enrichment, cooling time combination) with the storage system's components, including the reinforced concrete, at the design basis specifications in the system's design drawings and final safety analysis report (FSAR). The models were then modified to calculate dose rates for the selected system and source term after 20 years in dry storage as described below (see the “Material Loss Cases – Changes in the Cover Concrete” section of this report).

The analysis models were set up for analyzing radial dose rates and the changes in radial dose rates. One reason for taking this approach is that the concrete in the system lid or door is encased in steel for nearly all of the lids or doors for the analyzed systems in which concrete is part of the lid or door design, and this concrete is therefore considered to be less susceptible to degradation. As described below, models were also created to address the designs in which the lid or door concrete was not encased in steel. This consideration only affected the analysis for the NUHOMS systems since only the NUHOMS system has module designs with un-encased concrete in the door. Another reason is that the radiation source terms from the fuel are the main concern, and not the assembly hardware source terms, based on the characteristics described in the “DESIGN BASIS SOURCE TERMS” section of this report. The radial surface of the storage system extends along the length of the fuel and has the greatest surface area on which the radiation from the fuel impacts and through which it would pass.

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[1] For those licensees and CoC holders that have had to address the concerns and uncertainties of using SAS2H for calculating radiation source terms for high burnup spent nuclear fuel, the staff expects that the process by which the concerns and uncertainties were addressed should be part of any evaluation the licensee or CoC holder does to demonstrate that radiation surveys and monitoring are not needed in an AMP.

Each analysis model is cylindrical. For the MAGNASTOR and NAC-MPC systems, that is the actual shape of the storage overpack. For these systems, the concrete thickness and other radial dimensions in their respective models are those that are defined in the FSAR drawings for each system.

For the NUHOMS systems, the modules are rectangular in cross section. Thus, the cylindrical model is an approximation of the NUHOMS module. The radial features for a NUHOMS module include the side walls, the ISFSI basemat, and the module's roof. These module features have different concrete thicknesses and are at different distances from the canister stored in the module. In defining the radial dimensions of the NUHOMS analysis model, the thinnest gap between the spent nuclear fuel canister and the module walls is used as the inner radius of the model's cylindrical module.

Also, for the NUHOMS system analysis, the concrete thickness of module side walls and the roof were considered in determining the concrete thickness to use in the model. In addition, the thickness of the front and rear walls was also considered, which allowed for the model to be used to estimate the axial dose rate changes except for through the module door. Further, as previously stated, the model for the NUHOMS system is a composite across all of the different NUHOMS-type systems. The staff considered that the greatest increases in dose rates from the time of loading to the time at 20 years in dry storage would occur for designs with thinner concrete rather than thicker concrete. Thus, the staff selected the thinnest concrete specified for the module walls and roof from among the different module designs in existence. For the early module designs where a gap exists between modules, the module wall was thinner than the other walls and the roof. The analysis model used the thinnest side wall thickness from among the early module designs. For the more recent module designs, the front wall was thinner than the rest of the module; so, the model used the thinnest front wall thickness for the modeled radial concrete thickness.

The staff created additional NUHOMS models to estimate the effects of concrete loss in the module door to address the NUHOMS module designs that have doors with reinforced concrete that is not encased in steel. The doors for these module designs often include some thickness of steel, either on the inner face or the outer face of the concrete or on both faces. The gamma dose rates are the most significant component of the total dose rates in the system shielding analysis. Therefore, in selecting the door concrete thickness to use in the model, the staff also had to consider the gamma shielding effect of the steel in the door designs. Some designs had thick concrete with thin steel plates. Other designs had thin concrete with thick steel plates. Still others had some thickness of concrete with some thickness of steel that was in between these two designs. In choosing the concrete door thickness to use in its models, the staff selected the concrete thickness from the door design that had an overall minimum shielding capability from the combined shielding capability of the door's concrete and steel components. Thus, the modeled concrete is thicker than for some of the door designs; however, the staff considers that the selected thickness maximizes the effects of concrete loss in the module door in terms of dose rates.

The selected door concrete thickness was applied to the radial concrete in the models. Thus, the analysis approximates the impact of the door concrete loss in a configuration that is different from the actual configuration (i.e., the radial surface along the fuel length vs. the door at the axial end of the fuel in the module). While the door's concrete thickness was applied to the model's radial concrete, the steel from the door design was not included in the radial wall in the model.

The models include some additional simplifications for the overpacks and modules. Since the analysis focused on the concrete body and not streaming paths that are included by design (e.g., vent openings), the models do not include these designed streaming paths. For the NUHOMS modules, although the thin internal steel liners of the modules are included in the models, the support structure on which the storage canister rests (e.g., rails) and the materials of these items are neglected in the models. With the NUHOMS module being represented as a cylindrical cask, there is no room in the model for these items. Plus, they would not have a bearing on the radial or side dose rates for the module. For the vertical storage systems, the hardware in the bottom of the overpack is ignored, as is the concrete pad on which the overpack stands. The positioning of the storage canister in the overpack accounts for the presence of the hardware (e.g., pedestal) in the base of the overpack. Also, reinforcement steel in the concrete is ignored because this steel has little influence on the shielding performance of the concrete.

Source Configuration

Since, as described later in this report, the analysis is focused on the fuel's radiation source terms, the models are simplified to include only the active length of the fuel assemblies. For most of the analyses, the active fuel length is taken to be 144 inches, which is a typical active fuel length for commercial fuel assemblies. The active fuel lengths for the NAC-MPC contents are shorter and are in line with the active fuel lengths of the fuel assemblies approved for storage in that system. Thus, the storage canister's cavity is shortened to the active fuel length used in the model. The overpack, or module, is also truncated to keep the relative locations of system components with respect to each other (e.g., canister lid to overpack lid) consistent with what is shown in or determined from the design drawings in the respective system's FSAR. The spent fuel contents are modeled as a uniformly smeared mass of material that occupies the storage canister's entire cavity. The canister's fuel basket is ignored in the model.

Material Characteristics

The models assume all steel to be carbon steel, with the composition and density given in the SCALE computer code's manual (Section M8 of the SCALE5.1 manual). The models use the concrete composition specifications from the same SCALE manual. The concrete density is determined based on the density specified in the respective system's design drawings in the system's FSAR. Other materials from the system that are included in the models are based on the specifications for the materials that are provided in the respective system's FSAR.

The contents' material specifications are based on the contents occupying the storage canister's entire cavity as a homogeneous mass of fuel and cladding only (i.e., UO_2 and zirconium). No alloying elements are included with the zirconium. The fuel is assumed to be at 2 wt. percent enrichment in uranium-235. The density and composition specifications (i.e., weight fractions) of the contents are based on the modeled canister cavity, the cladding dimensions and density of zirconium, the mass of uranium per assembly, and the number of assemblies in the canister. For most systems and calculations, the PWR assemblies are assumed to be B&W 15x15 assemblies and the BWR assemblies are assumed to be GE 7x7 assemblies (or 7x7 assemblies similar to the GE design). The design specifications of these fuel types were used to determine the material properties of the modeled source. For the NAC-MPC, the contents are taken to be those described in the NAC-MPC's FSAR as the PWR and BWR contents that are allowed in that system and for which models were developed. Other materials in the canister (e.g., the canister basket components) were ignored. Since the purpose of the analysis is to determine the relative change in storage system dose rates, the staff considered that the effects of the other materials would not influence the relative change in dose rates; however, calculations were not performed to confirm that consideration.

Material Loss Cases - Changes in the Cover Concrete

Calculations of dose rates for the systems after 20 years of storage simply used the same models as described above for the system as designed but with the concrete thickness reduced by the system's specified minimum cover concrete thickness. Also, the source term used in the model was decayed for 20 years longer than the source term in the model of the as-designed system (i.e., the system at the time of loading). The cover concrete is removed from the entire axial length of the system. Loss of the entire thickness of cover concrete conservatively bounds any changes in the properties of the cover concrete and losses of concrete allowed by the ACI 349.3R Tier 2 criteria as used in the NUREG–2214 AMP for concrete degradation due to aging.

For some systems, additional calculations were done when the conservative configuration with the cover concrete entirely gone resulted in dose rates that exceeded the dose rates calculated for the system at the time of loading. These calculations include the configuration where the cover concrete was assumed to be present but with a density that was half the density specified in the system's design drawings and FSAR. In other calculations, the cover concrete was removed to the maximum allowable depth of scaling (1-1/8 inches) and the concrete in the remaining cover concrete thickness was set at half the design's specified concrete density. These configurations are less bounding than the configuration where the cover concrete is entirely gone, but are still consistent with the standard's Tier 2 criteria. Of these two configurations, the latter configuration is the most appropriate for calculating dose rates after 20 years in storage to compare with dose rates at the time of loading in instances when the calculated dose rates for the primary post-20-year storage configuration (i.e., all cover concrete gone) exceed the calculated dose rates for the system at the time of loading.

DESIGN BASIS SOURCE TERMS

The intent of the analysis was to cover a range of contents and source terms with the objective of identifying patterns or trends in how dose rates change from the time of loading to 20 years in dry storage, including whether or not dose rates change in the same manner regardless of source term characteristics. Thus, the analysis included calculations for both PWR and BWR spent nuclear fuel. The analysis included some spent nuclear fuel with low burnup and some with high burnup. In each case, the dose rate calculations used the spectra generated with SCALE5.1's SAS2H, with the total source strength from SAS2H scaled up to the number of assemblies that can be stored in the storage system (e.g., 37 or 24 PWR assemblies, 69 or 87 BWR assemblies).

The focus of the analysis is on the radiation source term from the fuel and not the fuel hardware or any non-fuel hardware items (e.g., control components) that could also be present in a loaded storage system. The main nuclide of concern for such hardware is cobalt-60, which has a half-life of about 5.3 years. After 20 years in storage, the gamma source from this nuclide will have decayed for at least about 3.75 half-lives, reducing the source strength to about 7 percent of what it was at the time of loading into the dry storage system, which will more than compensate for any degradation of the storage system's concrete to the extent allowed in the

ACI 349.3R Tier 2 evaluation criteria. Therefore, radiation source terms for fuel hardware and non-fuel hardware were not calculated for or included in the analyses.

The behavior of the fuel source terms, particularly the gamma source terms, is more complex, with contributions coming from multiple nuclides of different half-lives. Plus, the neutron source derives from nuclides of longer half-lives. Thus, the analysis focused on the fuel's radiation source terms (both gamma and neutron).

Given the rapid decay of the source term for spent nuclear fuel that is loaded into dry storage within a few years of discharge from a reactor, contents specifications with short cooling times were typically not considered in the analysis. This is because the rapid decay was considered to ensure that the source term decay would more than compensate for the loss of cover concrete. Thus, specifications that included relatively long cooling times prior to loading into dry storage were typically selected. Consistent with typical shielding analyses, specifications with lower minimum enrichments were also typically chosen since the lower enrichments result in stronger source terms. This analysis did not include investigation of the impact of using a higher enrichment for the same burnup and cooling time. The source term calculations used the B&W 15x15 assemblies and GE 7x7 assemblies (or 7x7 assemblies similar to the GE design) because these have the highest uranium mass loadings, which results in stronger source terms.

Source terms were calculated for the burnup, enrichment, and cooling time combinations for the time of loading of the storage system. For source terms after 20 years in dry storage, the same source term calculations were re-performed with an additional 20 years. The burnup, enrichment, and cooling time combinations that were selected for the time at system loading and used in the analysis are not necessarily the design basis combinations for the analyzed systems. However, the combinations were chosen to be in line with the analyzed systems' allowable contents. Thus, if a system's allowable contents include high burnup fuel, then the system was analyzed with a high burnup source term as well as a low burnup source term. The staff's expectation is that, given the burnup, cooling time, and enrichment combinations that were used to derive source terms for this analysis, the trends in dose rate changes with time that were seen in this analysis should be the same for the actual design basis contents of the analyzed systems. Note that source term is scaled up so that the source strength in the model is the total source from the number of assemblies that can be loaded in the system's storage canister.

The source terms for the NAC-MPC system are done a little differently than for the other analyzed systems. In the case of the NAC-MPC system, the analyzed source terms are the design basis source terms for that system, as described in the NAC-MPC's FSAR (Yankee fuel for the PWR cases and LaCrosse fuel for the BWR cases) with their respective design basis burnup, enrichment, and cooling time combinations and 20 years additional cooling for the cases at 20 years of dry storage. Also, as noted previously, the earlier NUHOMS module designs which include large gaps between adjacent modules are limited to fuel that is in the lower range of burnup. So, the calculations for these early module designs are limited to low burnup fuel, and the later module designs are analyzed with high burnup fuel.

DOSE RATE CALCULATION

The dose rates were calculated using surface tallies, which are features in MCNP that are used to calculate radiation fluence (rate) averaged across a surface of specified size. With the simplified nature of the models, the surface tallies were determined to be appropriate for calculating the dose rates. Plus, the use of surface tallies helped ensure meaningful results were obtained in a relatively short calculation time. The surfaces were set to the location of the outer concrete surface at the design-specified concrete thickness or just off of that surface (by a few millimeters). For the aged system calculations, while the concrete surface changed, the location of the surface detectors was kept the same. The staff does not expect this to have much effect on the results of the analysis, though the staff did not do a calculation to confirm that expectation.

The calculations use flux to dose rate conversion factors from the 1977 revision of the ANSI/ANS 6.1.1, "Neutron and Gamma-Ray Flux-to-Dose-Rate Factors."

Calculations were performed to determine gamma dose rates. Separate calculations were done to determine neutron dose rates. However, the neutron calculations included determination of the contribution from secondary gammas, gammas arising from the neutron interactions within the storage system and contents materials.

The focus of the calculations is on the relative change in the dose rates from the time of loading to the point in time 20 years after storage as opposed to the dose rates themselves. This is because the question is whether or not the ACI 349.3R Tier 2 criteria are sufficient to ensure the shielding function is adequately maintained. The basis for the focus on the relative change in dose rates in relation to answering that question assumes that the design basis dose rates are derived from an analysis from which dose rate limits in technical specifications were derived, which analysis also supports the evaluation of compliance with 10 CFR 72.104, in accordance with the requirement in 10 CFR 72.236(d). Thus, if the dose rates after 20 years of storage with the degradation allowed by the ACI 349.3R Tier 2 criteria (as evaluated in this analysis) do not exceed the design basis dose rates (which would be the dose rates at the time of loading), then the evaluation of compliance with 10 CFR 72.104 in the FSAR remains valid and bounds the system's performance during extended operation with the AMPs only relying on the ACI 349.3R Tier 2 criteria as acceptance criteria.

It is important to note that, as used in the NUREG–2214 AMP, the Tier 2 criteria/AMP acceptance criteria identify the degree of degradation beyond which a licensee would need to evaluate the impact on the intended function and take corrective actions if necessary. That evaluation should include the effects of that condition on the concrete’s shielding function. Based on that evaluation, the licensee may determine that the condition is acceptable and does not, at that point in time, require repair. The same process would be followed for an AMP that relied on radiation surveys and monitoring to ensure the concrete's shielding function is adequately maintained. Thus, how an identified condition would be handled is not a factor in determining whether or not the ACI 349.3R Tier 2 criteria would be sufficient to ensure maintenance of the shielding function.

RESULTS

Table 1 shows the results of the calculations for loss of concrete that were performed for this analysis. The table includes the system name, information to describe the configuration represented by the model, and the dose rate results. The neutron and the gamma calculation results for each configuration are placed in the same row. Although they are two separate calculations, the configuration is the same; hence, the results are shown for both gamma and neutron calculations in a single row. As stated in the introduction section of this report, the relative changes in dose rates are the focus and the results for this study. The dose rate values themselves, seen in Table 1, are not to be construed as estimates of actual storage system dose rates given the assumptions and simplifications used in the calculations, which the staff considered would not significantly influence the relative changes in dose rates from the time of system loading to the time after 20 years in dry storage. The calculated dose rates shown in Table 1 are useful only in identifying the relative changes in dose rates between these two times for each analyzed system and each analyzed spent nuclear fuel contents.

Loss of Cover Concrete

A review of the Table 1 results shows that for the MAGNASTOR overpack and the recent NUHOMS module designs (adjacent modules touch each other without a gap) for which the modeled concrete thickness is based on module features other than the door, even for the bounding configuration with the cover concrete completely gone, the dose rates after 20 years in dry storage are less than at the time of loading. This is also true for the PWR contents in the NAC-MPC overpack. However, this was not the case in terms of the gamma dose rates for the NAC-MPC's BWR contents or the remaining NUHOMS configurations that were analyzed. For these other cases, the gamma dose rates after 20 years in dry storage exceed the dose rates at the time of loading. The most significant change was for the NAC-MPC with the BWR contents; the dose rates were about 62 percent higher versus the dose rates at the time of loading. The increases for the NUHOMS cases were all less than 10 percent.

Reduced Cover Concrete Density With and Without Scaling Loss

Because of the increases in dose rates for the cases described above, additional calculations were performed that reduced the conservatism inherent in completely removing the cover concrete. The gamma dose rates for all cases with the cover concrete present but at half of its design basis density resulted in gamma dose rates after 20 years of storage that were less than the dose rates at loading. The case that best captures the degradation allowed by the ACI 349.3R Tier 2 criteria, though it is still conservative, is the scaling case (i.e., loss of concrete to depth of 1-1/8 inches, with remaining cover concrete at half of the concrete's design basis density). The NUHOMS case with the greatest increase in gamma dose rates when the cover concrete was removed was run again in the scaling configuration. The resulting gamma dose rates were less than those at the time of loading. Since this case had this outcome and had a more significant increase in gamma dose rates than the other NUHOMS cases, the staff concluded that calculating the gamma dose rates for these other NUHOMS cases in the scaling configuration would also result in gamma dose rates that are less than those at the time of loading. The same outcome was not observed for the NAC-MPC containing the BWR spent nuclear fuel. The gamma dose rates in the scaling configuration were still about 15 percent higher than the dose rates at the time of loading.

Table 1. Calculation Results for Loss of Concrete in the Cover Concrete of the Storage Overpack, Module

Storage System	Cover Concrete Condition	Assembly type and number	Burnup (GWd/MTU), enrichment (wt. % ^{235}U)	Fuel Cooling time[A]	Primary Gamma		Neutron		Secondary Gamma		Total Dose Rate (mrem/hr)
					Dose Rate (mrem/hr)	Relative error (%)	Dose Rate (mrem/hr)	Relative error (%)	Dose Rate (mrem/hr)	Relative error (%)	
MAGNASTOR	Present	PWR, 37	62, 3.8	20yrs @ loading	36.01	2.94	2.33	1.18	3.35	0.24	41.69
	Gone	PWR, 37	62, 3.8	40yrs @ 20 yrs storage	32.91	2.47	1.65	0.85	2.20	0.21	36.76
	Present	BWR, 87	40, 1.8	16yrs @ loading	26.99	2.88	1.41	1.14	2.01	0.24	30.41
	Gone	BWR, 87	40, 1.8	36yrs @ 20 yrs storage	26.00	2.47	1.12	0.93	1.40	0.21	28.52
NAC-MPC	Present	PWR, 37	32, 3.5	8yrs @ loading	32.34	2.50	0.47	0.70	0.74	0.15	33.55
	Gone	PWR, 37	32, 3.5	28yrs @ 20 yrs storage	23.84	1.67	0.46	0.46	0.57	0.13	24.87
	Present	BWR, 68	22, 3.6	28yrs @ loading	14.71	1.72	0.07	0.66	0.08	0.15	14.86
	Gone	BWR, 68	22, 3.6	48yrs @ 20 yrs storage	23.89	0.97	0.08	0.43	0.08	0.13	24.05
	1/2 density	BWR, 68	22, 3.6	48yrs @ 20 yrs storage	14.39	1.32	--[C]	--	--	--	--
	Scaling[B]	BWR, 68	22, 3.6	48yrs @ 20 yrs storage	16.99	1.17	--	--	--	--	--
NUHOMS (early module design)	Present	PWR, 24	45, 1.5	15yrs @ loading	5199.25	0.24	44.25	0.34	18.63	0.15	5262.13
	Gone	PWR, 24	45, 1.5	35yrs @ 20 yrs storage	5437.08	0.19	32.52	0.27	11.51	0.14	5481.11
	1/2 density	PWR, 24	45, 1.5	35yrs @ 20 yrs storage	3877.76	0.21	--	--	--	--	--

	Present	BWR, 52	40, 1.8	16yrs @ loading	4204.37	0.23	22.58	0.33	9.25	0.15	4236.20
	Gone	BWR, 52	40, 1.8	36yrs @ 20 yrs storage	4575.46	0.16	16.60	0.26	5.72	0.13	4597.78
	1/2 density	BWR, 52	40, 1.8	36yrs @ 20 yrs storage	3263.79	0.21	--	--	--	--	--
	Scaling[B]	BWR, 52	40, 1.8	36yrs @ 20 yrs storage	3910.30	0.18	--	--	--	--	--
NUHOMS (recent module design)	Present	PWR, 37	62, 3.8	20 yrs @ loading	52.08	2.32	1.95	1.11	2.18	0.34	56.21
	Gone	PWR, 37	62, 3.8	40 yrs @ 20 yrs storage	49.51	1.65	1.42	0.69	1.45	0.24	52.38
	Present	BWR, 69	62, 3.8	16yrs @ loading	57.68	1.88	1.89	1.01	2.02	0.31	61.59
	Gone	BWR, 69	62, 3.8	36yrs @ 20 yrs storage	53.12	1.45	1.36	0.78	1.33	0.27	55.81
NUHOMS (recent module design, concrete at door thickness)	Present	BWR, 69	62, 3.8	16yrs @ loading	722.35	0.61	10.17	0.52	6.90	0.19	739.42
	Gone	BWR, 69	62, 3.8	36yrs @ 20 yrs storage	738.05	0.43	7.45	0.41	4.49	0.17	749.99

Table 1 Notes
A. This column gives the post-irradiation decay time of the fuel and whether the case is at the time of loading into the dry storage system or at 20 years of storage in the dry storage system.
B. This entry means the cover concrete was removed to a depth of 1.125 inches and the remaining cover concrete was at 1/2 the design basis density specified in the storage system's design drawings and FSAR.
C. A '--' anywhere in the table means that this dose rate (and its uncertainty) was not calculated.

CONCLUSIONS

Based on the results of this analysis, it appears that the ACI criteria are sufficient to ensure adequate performance of the reinforced concrete's shielding function for the MAGNASTOR, the NAC-MPC with its PWR contents, and the NUHOMS systems during the period of extended operation. The conclusion would also apply to systems that are similar to these systems and have the same, or similar, spent nuclear fuel contents. For the NAC-MPC with its BWR contents, and for similar systems with similar contents, additional justification appears to be necessary for accepting the ACI criteria as sufficient to ensure adequate performance of the shielding function of the reinforced concrete.

The necessary justification may depend on how the system was analyzed and the basis for any license or CoC conditions (e.g., dose rate limits in the technical specifications). For example, if a system's analysis for demonstrating compliance with 10 CFR 72.104 (per 10 CFR 72.236(d) for certified storage systems) and the system's technical specification dose rate limits, if any, are based on the most bounding system contents allowed in the system, then the added justification may include pointing to the margin between the limits in the technical specifications, or the design basis dose rates for the bounding system contents, and the design basis dose rates for the contents for which dose rates were estimated to increase after 20 years in storage. If the identified increase in dose rates does not exceed the identified margin or result in dose rates after 20 years of storage that exceed the technical specification limits, then the ACI criteria may be considered sufficient. If the increase is greater than the identified margin, or the dose rates after 20 years in storage exceed the technical specification limits, or the technical specification limits and 10 CFR 72.104 analysis is based on the contents for which dose rates increased, then this kind of justification cannot be made. Instead, more rigorous calculations or some other justification may be needed to demonstrate the ACI criteria are sufficient. Otherwise, the AMP may also need to include radiation surveys and monitoring.

In the case of the NAC-MPC with its design basis BWR fuel, this analysis indicated that the dose rates increase after 20 years of storage. However, if the design basis dose rates for the system’s PWR contents are bounding for all of the system’s contents, both PWR and BWR fuel, and the design basis 10 CFR 72.104 analysis and technical specification dose rate limits, if there are any, are based on the NAC-MPC's design basis PWR contents (vs. a separate analysis and separate technical specifications for PWR and BWR fuel contents), then justification of reliance on the ACI criteria could involve looking at the margin between the dose rate limits in the technical specifications, or the design basis dose rates for the PWR contents, and the design basis dose rates for the BWR contents. If the increase in the BWR fuel's dose rates after 20 years of storage does not result in dose rates that exceed those limits or challenge the 10 CFR 72.104 analysis, then the ACI criteria would be deemed sufficient. If that is not the case, or a separate 10 CFR 72.104 analysis was performed and separate dose rate limits established for the NAC-MPC with its design basis BWR contents, then some other kind of justification would be needed. Otherwise, the AMP may also need to include radiation surveys and monitoring.

The calculations in this analysis show that if burnup is low enough or cooling time is long enough, the ACI criteria may not be enough. That is the case for the NAC-MPC with its design basis BWR fuel. That fuel had a design basis burnup of 22 GWd/MTU and a design basis cooling time at the time of loading of 28 years, thus the further decay of the source after 20 years in storage would not be expected to be significant. Thus, for systems where the design basis contents are more like the design basis BWR contents in the NAC-MPC, justifying that the ACI criteria alone are sufficient to ensure the reinforced concrete's shielding function will be adequately maintained will likely require something more than what was done in this analysis. It is true that all systems may load spent nuclear fuel with such burnups and cooling times; however, the staff expects that the design basis contents specifications result in dose rates that are significantly greater than the dose rates from this low burnup, long cooling time fuel. Since analyses for compliance with 10 CFR 72.104 and 10 CFR 72.236(d) and for establishing any technical specification dose rates limits would be based on the design basis contents, the staff expects there will be enough margin in the analysis and technical specification limits to compensate for the dose rate increases that could be predicted by analysis for these low burnup, long cooling time fuel contents.

Additionally, the calculations show that the cover concrete thickness can also be more significant when the original, design thickness of the overpack or module concrete is relatively thin. This is seen in the trends for the NUHOMS cases with the recent module designs versus those same designs when the radial concrete is set to the thickness of concrete in the module door. Thus, justification of the adequacy of the ACI criteria should also consider the effect of concrete thickness, particularly if the certified system has multiple overpack or module designs that vary in concrete thickness, whether in the radial or side concrete or in the lid or door concrete. For horizontal modules, the consideration should also extend to the axial front and rear concrete.

As stated in an earlier section of this report, the staff expects that the allowable crack size should not have an effect on dose rates. The crack width is quite small, and the staff does not expect that a crack of that size would afford a streaming path of significance that would be detectable using a radiation survey meter of any typical dimension. Furthermore, any crack in the reinforced concrete of any depth would follow a torturous path through the concrete as a result of the presence of aggregate in the concrete and other relevant concrete properties. Thus, there will be significant concrete material directly beneath where a crack appears on the concrete structure's surface, which will provide shielding. Plus, the tortuous nature of the crack's path will also ensure any radiation (gamma or neutron) will interact in and by shielding by the concrete surrounding the crack. Based on these considerations, the staff concludes that the cracks allowed by the ACI 349.3R criteria will not affect dose rates from the reinforced concrete structures and so the staff did not perform any calculations to assess dose rates from cracks in these structures.

As noted previously, this analysis only considered degradation of the exterior surface of the reinforced concrete. For NUHOMS-type systems, the inner surface of the reinforced concrete is also exposed (i.e., it is not covered by a steel shell like is the case for vertical overpack systems) and, theoretically, could also experience degradation to the level of the ACI 349-3R Tier 2 criteria. The results and conclusions of this analysis do not apply for such a case. However, operating experience from internal concrete inspections to date has not identified degradation that is considered to be capable of affecting the shielding function of concrete overpacks and modules. As a result, for the purpose of this study, using a configuration where only the external concrete is degraded is considered to be a reasonable approach to evaluate whether inspections per the ACI criteria, rather than radiation monitoring, can be supported. Nevertheless, the NUREG–2214 AMP includes periodic inspections of internal concrete surfaces, and results from those inspections will be used to verify the validity of the methodology of this study.

As is described throughout this report, this analysis is based on some assumptions and considerations regarding whether or not certain features were included or neglected in the models. Features that were neglected were considered to not have an effect on the relative change in dose rates; however, no calculations were performed to confirm this determination. These assumptions and considerations are summarized here and include:

1. The storage canister's basket materials were ignored.

2. For NUHOMS models, the concrete thicknesses of the module's front wall and door (for doors with reinforced concrete) were used in some models as the radial concrete thickness to estimate the effects of the loss of axial concrete.

3. All models are cylindrical; for the NUHOMS models, the inner radius of the module is the smallest distance between the storage canister and the module's wall components.

4. All vents and other 'designed in' streaming paths were ignored.

5. Support structures for the storage canister in the NUHOMS modules were ignored while the thin internal steel liners were included.

6. The materials of the overpack's base components in vertical storage systems were ignored; however, their impact on the axial positioning of the storage canister was included. The steel shells on the inside of the overpack concrete were included in the models.

7. The storage facility's basemat concrete was ignored.

8. Fuel assembly hardware and nonfuel hardware were ignored in the source term calculations and the shielding models.

9. Only the fuel source term was calculated, and the shielding models include only the active fuel zone of the assemblies. The storage canister cavity height is set to equal the active fuel length of the assemblies, and all other storage system component axial lengths are adjusted accordingly to maintain distances and positions relative to the storage canister components.

10. Material specifications for the assemblies only included the UO_2 and zirconium from the fuel rod cladding.

11. All modeled steel is assumed to be carbon steel at the density and with the material specifications described in the SCALE5.1 manual.

12. All modeled concrete is assumed to have the composition described in the SCALE 5.1 manual.

13. The design basis contents for the source term calculations and the shielding models use the B&W 15x15 assembly type for PWR contents and the GE 7x7 assembly type for BWR contents, except for the NAC-MPC analyses, which use the NAC-MPC's design basis assembly types.

14. The positions of the surface detectors remained constant for the calculations at the time of 20 years in dry storage versus their positions at the time of loading. In other words, the detector position was not adjusted based on the loss of cover concrete. Thus, the detector position was always located at the surface of the concrete structure's

design basis outer surface location whether or not the cover concrete was present in the model. Surface detectors were neither axially sub-divided nor azimuthally sub-divided. The detector's axial extent was at least the full length of the modeled overpack or module cavity.

15. Staff-identified limitations with using SAS2H for evaluation of radiation source terms from high burnup spent nuclear fuel are not expected to influence the relative change in dose rates from the time of loading to 20 years of dry storage for cases involving high burnup spent nuclear fuel since the objective of this study is to identify the relative changes of source terms at different cooling times. The burnup and enrichment are the same for any given comparison. (The staff expects that the process by which licensees and CoC holders addressed the concerns and uncertainties associated with using SAS2H for high burnup source term analyses for the their license or CoC should be part of any evaluations they do to demonstrate that radiation surveys and monitoring are not needed in an AMP.)

16. No adjustments were made to the source to account for the fuel's axial burnup profile. The fuel's source terms were uniformly applied axially and radially throughout the storage canister cavity. Maximum source strengths were the value in the SAS2H results multiplied by the number of assemblies in the storage canister.

17. The analysis premise is that the 10 CFR 72.104 (and 10 CFR 72.236(d)) analyses and the dose rate limits in the technical specifications, if any, for the systems are based on the design basis source terms at the time of system loading and that separate analyses and limits were determined for these systems for the different design basis contents (e.g., PWR vs. BWR, low burnup vs. high burnup). Thus, only calculations that showed dose rates after 20 years of storage would be less than at the time of loading were considered to demonstrate that inspections using the ACI 349.3R criteria are sufficient to ensure the storage system's reinforced concrete structures' shielding function will be adequately maintained during the period of extended storage.

FIGURES ILLUSTRATING THE MODELS

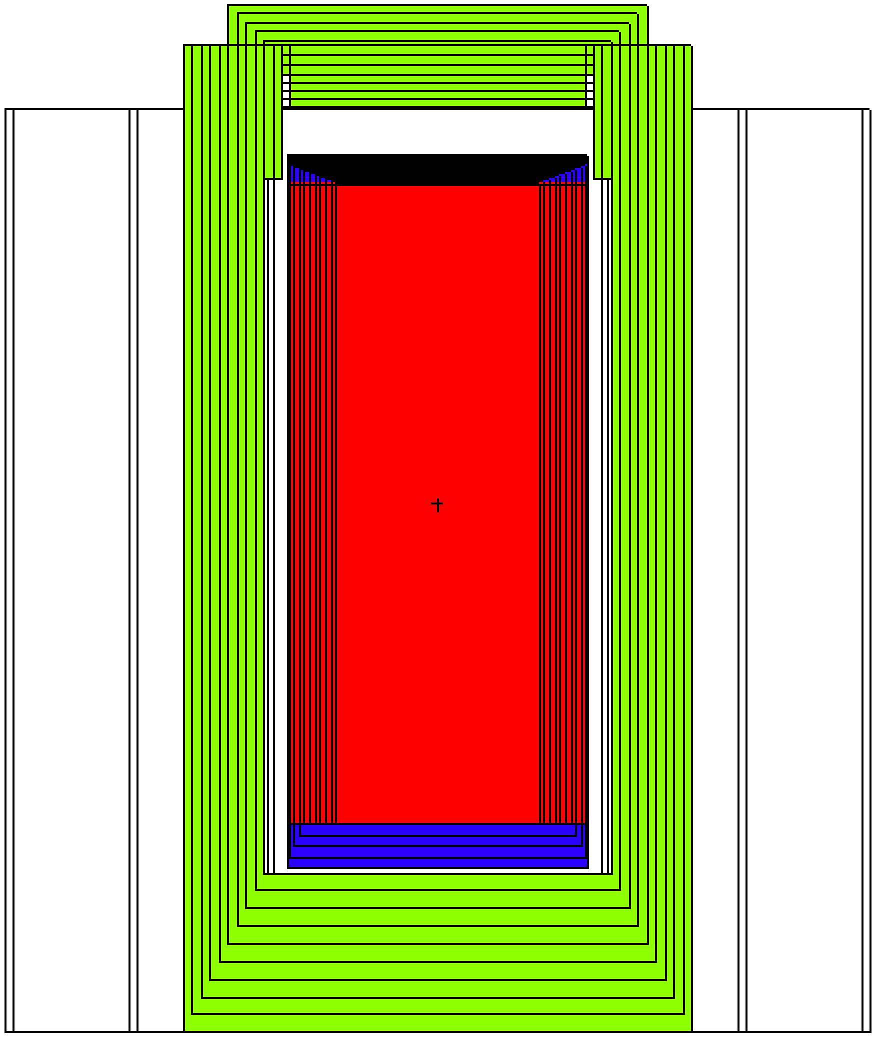

Figure 1. Low burnup PWR case in NUHOMS module (early module design).

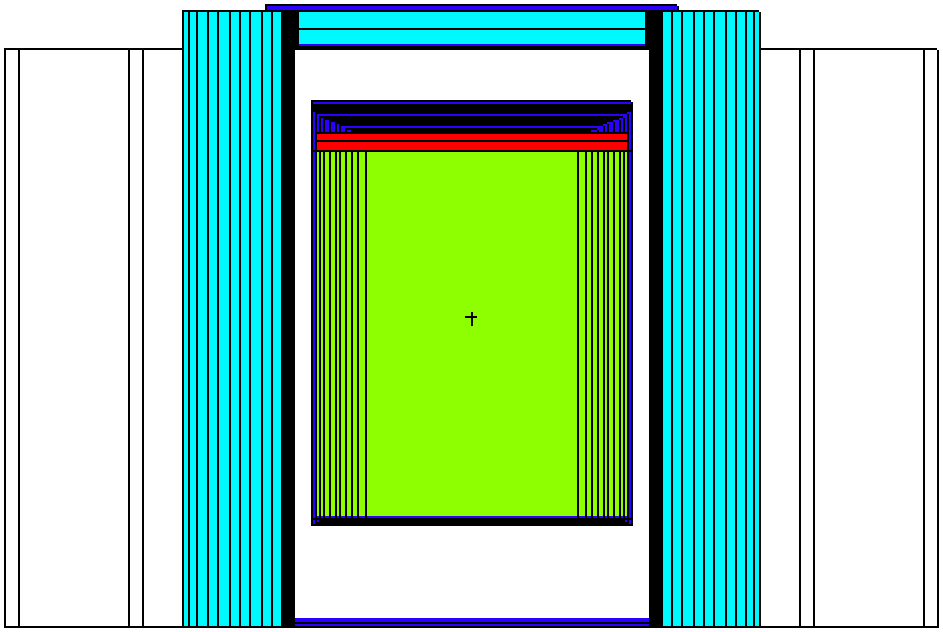

Figure 2. LaCrosse BWR case in NAC-MPC overpack.