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

July 2, 2021

Mr. Robert D. Quinn
Nuclear Material Management, Director
Westinghouse Electric Company
1000 Westinghouse Drive
Cranberry Township, Pennsylvania 16066

SUBJECT: APPLICATION FOR THE WESTINGHOUSE SENTRY™ DRY STORAGE CASK SYSTEM, CERTIFICATE OF COMPLIANCE NO. 1026, AMENDMENT NO. 5 – REQUEST FOR ADDITIONAL INFORMATION (EPID L-2020-LLA-0105)

Dear Mr. Quinn:

By letter dated April 30, 2020 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML20121A196), as supplemented on June 5, 2020, October 2, 2020, November 20, 2020, and January 15, 2021 (ADAMS Accession Nos. ML20164A120, ML20276A295, ML20329A083, ML21019A509, and ML21019A509, respectively), Westinghouse Electric Company LLC submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for the SENTRY™ Dry Storage System, Certificate of Compliance No. 1026, pursuant to the requirements of Part 72 of *Title 10 of the Code of Federal Regulations*.

This request for additional information identifies information needed by the NRC staff in connection with its review of the application. Each question describes information needed by the staff for it to complete its review of the application and to determine whether the applicant has demonstrated compliance with regulatory requirements.

In order to complete our technical review on schedule, your response should be provided by within 90 days from the date of this letter. If you have any questions regarding this matter, I may be contacted at (301) 415-5196.

Sincerely,

Nishka Devaser

Nishka Devaser, Project Manager
Storage and Transportation Licensing Branch
Division of Fuel Management
Office of Nuclear Material Safety
and Safeguards

Docket No. 72-1026
EPID L-2020-LLA-0105

Enclosure:
Request for Additional Information

SUBJECT: APPLICATION FOR THE WESTINGHOUSE SENTRY™ DRY STORAGE CASK SYSTEM, CERTIFICATE OF COMPLIANCE NO. 1026, AMENDMENT NO. 5 – REQUEST FOR ADDITIONAL INFORMATION (EPID L-2020-LLA-0105)

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*** by email ADAMS Accession Number: ML21146A096**

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Request for Additional Information
Docket No. 72-1026
SENTRY™ Dry Storage System
Certificate of Compliance No. 1026
Set 1

By letter dated April 30, 2020 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML20121A196), as supplemented on June 5, 2020, October 2, 2020, November 20, 2020, and January 15, 2021 (ADAMS Accession Nos. ML20164A120, ML20276A295, ML20329A083, ML21019A509, and ML21019A509, respectively), Westinghouse Electric Company LLC (Westinghouse or the applicant) submitted to the U.S. Nuclear Regulatory Commission (NRC) an application for the SENTRY™ Dry Storage System (SENTRY DSS or SENTRY), Certificate of Compliance (CoC) No. 1026, pursuant to the requirements of Part 72 of *Title 10 of the Code of Federal Regulations*.

The requested information is listed by chapter number and title in the applicant's safety analysis report (SAR or application). The NRC staff used NUREG-2215, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility — Final Report," in its review of the application. This request for additional information identifies information needed by the NRC staff (the staff) in connection with its technical review of the SENTRY DSS application.

Chapter 4: Structural Analysis

4-1. Provide a material description, engineering drawings, and any calculations associated with performance of the components under normal, off-normal, and accident conditions related to the following components:

a. Transfer Mating Device. SAR Table 3.1-1 lists this component as important-to-safety (ITS),

b. Canister handling device. SAR Table 3.1-1 lists this component as ITS,

c. Transfer Cask Lifting Yoke. SAR Table 3.1-1 lists this component as safety related components based on criteria from 10 CFR 50/52, and

d. Paddle Extension. SAR Table 3.1-1 lists this component as safety related components based on criteria from 10 CFR 50/52.

This information is needed to comply with 10 CFR 72.146(a) 72.236(b) and 72.236(l).

4-2. In relation to the sensitivity study for tip-over devices for the W21H and W37 canisters, as described in Appendix G of the DDRWM-CN-00543-GEN, "SENTRY W180 Storage Cask Dynamic Calculations," please provide the following:

a. an explanation of the apparent discrepancy in final peak acceleration results for both canisters,

b. benchmarking data to assess the performance of the tip-over devices,

c. additional description of the mesh used for the tip-over devices including aspect ratios, relative size and contacts between the tip-over device and the rest of the model, and

d. a revision of Table G-1 of the calculation package which appears to have a typo that has reversed the line items corresponding to the W21H canister.

As described in Appendix G, the applicant doubled the number of elements on the tip-over device, subjected the finite element model to tip-over loads (including both types of canisters) and recorded peak accelerations in Table G-1 of the same appendix. The applicant concludes *"that the maximum peak acceleration is obtained on the models with least numbers of elements in the meshing. So, to be conservative, the model with the lower mesh density was used to model the [tip over devices]."*

The staff notes that the difference in peak accelerations for both mesh densities for the W21H canister was around 5%; however, the difference in peak accelerations between the models in the case of the W37 canister is around 15%. Based on the results presented in the SAR, it appears that the final peak accelerations for the tip-over devices are dependent on mesh density.

Given that the applicant has not provided benchmarking data about the tip-over devices, there does not appear to be a quantitative basis for how those elements are expected to behave under tip-over conditions.

Further, Based on Figures G-1 trough G-4, the staff cannot accurately determine how these features were modeled.

This information is needed to comply with 10 CFR 72.236(l).

4-3. Provide LS-Dyna ".d3plot" files related to the W37 and W21H canister tip-over analysis.

In Sections 4.6.1.1.3.1 and 4.6.1.1.3.2 of the SAR, the applicant stated that the tip-over devices were not modeled, but only the nodes of the contact points at the beginning of the impact between these tip over devices and the canisters were modeled.

Since outputs of LS-Dyna (.d3plot files) have not been provided, the staff cannot accurately determine that the FE model and corresponding analyses adequately represent the structural behavior under this accident condition. The reaction forces of the tip-over devices used to analyze the canisters need to be reviewed by the staff.

This information is needed to comply with 10 CFR 72.236(c) and (l).

4-4. Provide additional information for the following figures:

a. Figure 4.4-18, "Tip-over - W37 Reaction Forces Distribution Sketch" is low resolution and not legible. This figure provides the lateral reactions into the W37 canister of the anti tip-over devices which are subsequently modeled into the ANSYs finite element model (FEM). Therefore, the staff needs to review this image to understand how these forces are considered. Provide a higher resolution image of Figure 4.4-18.

b. SAR subsection 4.6.1.1.3.2 “W21H Canister Tip-over” refers to a Figure 4.4-42 for a detailed view of the reaction forces of the tip over device acting on the heat dissipating fins (HDF) external nodes. The staff believes this citation is incorrect and cannot find this figure. Revise the citation and/or provide the missing figure.

This information is needed to comply with 10 CFR 72.236 (c) and (l).

4-5. Provide DDRWM-CN-00542-GEN, "SO Finite Element Model and Lifting Analysis", Revision 0, August 2019.

In WDD-CN-00546-GEN, “Structural Design Report for SENTRY W180 Storage Cask,” the applicant cites DDRWM-CN-00542-GEN, "SO Finite Element Model and Lifting Analysis," Revision 0, August 2019. This document is cited when discussing the handling analyses of structures that are ITS, when describing the W180 storage cask FEM, and in other places.

The staff was not able to locate this document in the electronic reading room or as part of the docketed information. The staff needs this report to understand the intricacies of the W180 storage cask FEM and assess its ability to model structural integrity.

This information is needed to comply with 10 CFR and 10 CFR 72.236(l).

4-6. Provide additional information on the six end – drop cases analyzed.

SAR Section 4.6.5.6.1 states that “*There are six cases evaluated for the end drop of a storage cask. All cases are free drops from a height of 24 inches and impact flat on the cask’s bottom end. Four main cases are used to evaluate the maximum g-loads on the canister.”* However, in the aforementioned section the applicant did not provide any additional details or results of the six cases apart from what is in the referenced citation above. Provide the specifics of these cases, associated results, conclusions reached and update the SAR, as appropriate.

This information is needed to comply with 10 CFR 72.236(l).

4-7. Provide the following information on end drop analysis.

SAR Section 4.6.5.6.2 discusses the structural evaluation of the storage cask after an end drop and further refers to Figures 4.6-50 through 4.6-54 for additional results. The staff requests the following:

a. The applicant states that “Cracking or spalling of the cover concrete at the bottom part of the storage cask may result from the end drop, however any loss of the cover concrete on the bottom end of the storage cask does not affect its structural integrity.” As stated in SAR Section 1.2.1.1, one of the primary functions of the reinforced concrete in the storage cask is to maintain biological shielding during design basis loadings. The staff finds that the information provided in this section doesn’t fully address how shielding is maintained after the drop accident.

b. SAR Figures 4.6-50 and 4.6-51 show the resulting concrete effective plastic strains during an end drop event. The staff noted that given that this is a one-half symmetry FEM that is vertically striking the target, the resulting plastic strains should behave in a similar symmetric manner. The aforementioned figures show some areas of possible antisymmetry in the response. These could be caused by a number of reasons including possible contact discrepancies between elements. Justify these apparent discrepancies.

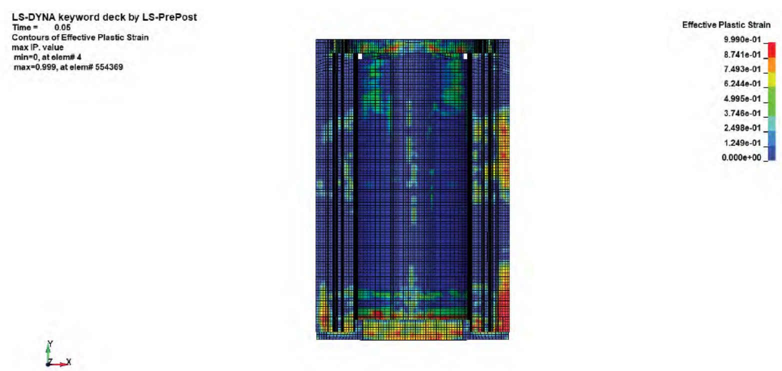

Figure 4.6-50 - SENTRY W180 Storage Cask Concrete Effective Plastic Strain during End Drop Event - Inner View

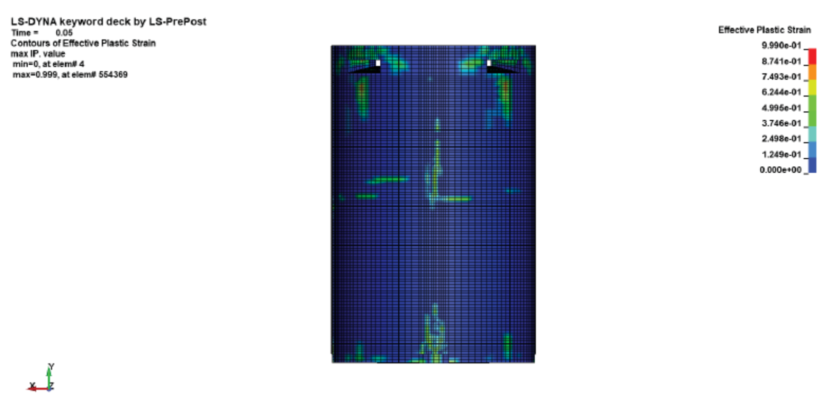

Figure 4.6-51 - SENTRY W180 Storage Cask Concrete Damage during End Drop Event - Outer View

c. SAR Figures 4.6-52 and 4.6-53 shows steel reinforcement axial stresses after the end drop accident. The figures provided are low resolution and not legible. This information will help the staff ascertain how stresses are distributed in the reinforcement sections and that allowable stresses are not surpassed. Provide higher resolution figures.

This information is needed to comply with 10 CFR 72.236(l).

4-8. Provide additional information on end drop shock absorbers validation.

In its response to RSI 2-1 (ADAMS Accession Number ML20276A299), the applicant refers to the W150 storage cask from the FuelSolutions™ storage system (CoC 72-1026), which also relies on end drop shock absorbers to support canisters and to limit deceleration during a postulated end drop event. The applicant states that *"The methodology used for the SENTRY W180 storage cask and contained canisters is essentially the same as that used for the FuelSolutions W150 storage cask and canister."* In addition, in SAR Section 4.6.5.6.1, since the applicant relies on the performance of these end drop shock absorbers for acceptable end drop accident results, the staff needs to ensure that the FEM produces realistic behavior. In order for the staff to reach a safety determination, provide the following:

a. Evaluations as to how the W180 and the W150 storage cask are comparable. This evaluation should address pertinent parameters such as cask weight, cask dimensions, materials and others, as appropriate.

b. LS-DYNA output files (.d3plot) related to the W180 end drop analyses. The staff needs these files to visualize the results of how the methodology used for the W150 cask analysis characterizes the structural behavior of the end drop shock absorbers for the W180 storage cask.

This information is needed to comply with 10 CFR 72.236(l).

4-9. Evaluate stresses on fuel basket plates as a result of tip-over.

SAR Section 4.4.1.2.2 discusses finite element analyses performed for the W37 and W21H canister fuel baskets. The applicant does not mention if the fuel baskets were analyzed for tip-over accidents and/or the staff cannot determine if such an evaluation was performed. Decelerations related to a non-mechanistic tip-over can represent the worst-case loading scenario that a basket can experience. This scenario can transfer maximum impact load after drop to the fuel assemblies which can have adverse effects on overall basket geometry and subsequently, criticality. Analyze induced stresses in the fuel basket panels and welds between panels caused by decelerations after tip-over accident and compare to allowables. This analysis should include multiple orientations of the baskets.

This information is needed to comply with 10 CFR 72.236 (c) and (l).

4-10. Address the apparent differences in symmetry in stress intensity after tip-over.

Figure 4.6-33 shows the stress intensity in the heat dissipation fins after tip-over. The staff notes that this is a one-half symmetry FEM, and therefore the resulting stress intensities should behave in a symmetric manner. In other words, once the canister hits the target, the resulting stresses, most likely, would be higher at the point of impact and dissipate outwards. By only relying on the picture in Figure 4.6-33, there seems to be two possible places where the fins were challenged (see red colored areas in figure). This situation may be characteristic of possible differences in symmetry in the model in the response to tip-over. These could be caused by a number of reasons, including possible contact discrepancies between elements. Provide reasons for the possible differences in symmetry of the stress intensities.

This information is needed to comply with 10 CFR 72.236 (f) and (l).

4-11. Provide Hourglass energy information.

In SAR Section 4.A.6, "Description of Model Assembly," the applicant stated that *"Hourglass mode has been considered in the model by using Belytschko-Binderman (1993) since one integration point has been implemented in this model. Hourglass energy is reasonable when it is less than 10% of total energy."* However, the applicant did not provide any information documenting this behavior. Excessive hour glassing can be an indicator of the FEM possibly exhibiting unrealistic behavior. Provide the following:

a. Total internal energy and total hourglass energy plots for the SENTRY W180 storage cask FEM for the tip-over and drop scenarios.

b. Hourglass energy plots for critical parts and sections of the FEM such as shock absorbing tubes and tip-over devices.

This information is needed to comply with 10 CFR 72.236(l).

4-12. Justify the assumed value of interference due to differential thermal expansion.

In SAR Section 4.5.1.2.3 the applicant states that a "there is a small mechanical interference between the basket and the canister shell in the W37 canister due to differential thermal expansion." The applicant should justify why the assumed interference is conservative and why a larger interference is not expected, with consideration for the nominal gap between the fuel basket outer surface and the canister shell. Note that an interference equal to the nominal gap will exceed the limit for Primary plus Secondary stresses as discussed SAR Section 4.5.1.2.3.

This information is necessary to comply with 10 CFR 72.236(c)(e) and (l).

Chapter 6: Shielding Analysis

6-1. Explain why the maximum burnup in the proposed technical specifications (TS) is not consistent with the maximum burnup in the SAR.

TS Table 2.1-9 indicates that the maximum burnup is 65 GWd/MTU, however, in TS Tables 2.1-11 to 2.1-56, the maximum burnup is listed as 60 GWd/MTU. The applicant needs to verify that the maximum burnup is consistent between the TS and the SAR and ensure that the source term calculations in the SAR support the TS limit.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(a) and 72.236(d).

6-2. Explain in detail how the spent fuel is qualified as authorized contents. If a design basis fuel assembly was used in the shielding analysis, provide specification for the design basis assembly used in the shielding design.

SAR Section 6.1.2 states: "*The approach used to determine fuel assembly acceptance for the SENTRY system is an integrated fuel qualification approach. Primary system safety parameters (such as component dose rates and heat loads), form the basis for fuel acceptance. Discrete analyses are performed to develop parameters for the fuel cooling tables for the W37 and W21H canisters, <u>which provide qualified cooling times for a broad range of fuel assembly classes</u>, with consideration for their burnup and enrichment. These analyses incorporate all the characteristics of each qualified fuel assembly class over the qualification range of initial enrichments and burnups. The cooling time which satisfies the allowable component dose rates and heat loads is determined for each initial enrichment and burnup combination. Thus, any applicable fuel assembly that meets the required cooling time satisfies the safety constraints used to construct the cooling table. This integrated fuel qualification approach has the benefit of reducing the number of 'bounding assembly' assumptions.*"

From these statements, excerpted from the SAR, it is not clear how the fuels were qualified. The applicant needs to clarify if shielding analyses were performed for all fuel designs as specified in the TS for the SENTRY system and provide a step-by-step explanation for this approach. The applicant also needs to provide the discretized intervals for these variables, e.g., the intervals of burnup, enrichment, and cooling time and demonstrate that these selected discretization levels are appropriate.

The staff also notes that in the SAR, as quoted above, the underlined statement seems to indicate a bounding fuel design is used. Otherwise, it is not clear how the analyses for some discrete values of the fuel parameters could provide qualification for a broad range of fuel assembly classes.

If a bounding fuel assembly approach was used, the applicant should provide specification for the selected fuel assembly design with specific parameters, such as the total uranium load per assembly, a given combination of burnup, enrichment, and cooling time that produces the bounding source terms for the shielding calculation. A design basis fuel assembly is an explicitly defined fuel assembly rather than a set of parameters derived from the "safety parameters (such as component dose rates and heat loads), form the basis for fuel acceptance" as stated in the SAR.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(a) and 72.236(d).

6-3. Demonstrate that it is appropriate to use the midplane dose rate for the calculation of direct doses to the public at the controlled area boundary and radiation levels near the cask for radiation protection of the workers inside the controlled area boundary.

SAR Section 6.1.2 states: "*The radiological acceptance criteria used to construct the cooling tables is the calculated dose rate on the storage cask surface over the peak burnup section of the fuel, which occurs near the cask midplane (midheight). This criterion is selected primarily to limit off-site direct doses to the public at the controlled* *area boundary of a typical ISFSI. It also serves to maintain the storage cask and transfer cask surface dose rates, which affect the public and operating personnel As Low As Reasonably Achievable (ALARA) during canister loading, closure, and transfer operations. As discussed in Section 6.1.4, the transfer cask midplane dose rate criterion is also considered in the construction of the cooling tables.*"

However, it is not clear that using the midplane dose rate is appropriate for calculating the annual dose at the controlled area boundary and the dose rates near the storage cask and transfer cask. The staff's experience is that for a typical storage cask design, the top and bottom vent areas of the cask usually have higher dose rates and that the transfer cask may have a different dose rate profile. Also, the staff's understanding is that a significant portion of the radiations at the controlled area boundary is contributed by skyshine and all the neutrons and gammas coming out of the cask may reach the controlled area boundary via skyshine.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

6-4. Explain how decay heat was used to calculate the bounding source terms and demonstrate that this approach will not produce non-conservative results for shielding analyses.

SAR Section 6.1.2 states: "*The thermal acceptance criteria used to construct the cooling tables is the maximum canister heat load which satisfies all the allowable temperature criteria for fuel cladding and SENTRY canisters, storage cask, and transfer cask materials, as described in Chapter 5. The greater of the resulting cooling times derived from the radiological and thermal evaluations is identified in fuel cooling table.*"

However, it is not clear how decay heat was used to derive source terms for the fuel to be stored in the SENTRY system. Based on NUREG/CR-6700 and a recent study published by Oak Ridge National Laboratory[1], there is essentially no correlation between decay heat and radiation source terms (neutron and gamma). Therefore, using a higher decay heat load will not necessarily result in higher source terms for shielding calculation. The shielding calculations for dose rates and site boundary dose should be made based on the allowable fuel that produces the maximum source terms.

The applicant needs to demonstrate that the dose rates for the all loading patterns are bounded by the values used for the shielding calculation for the SENTRY system.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

[1] R. Cumberland et. al., "A Study on the Relationship between Dose Rate and Decay Heat for Spent Nuclear Fuel Casks," Oak Ridge National Laboratory, June 17, 2020. https://doi.org/10.2172/1649326.

6-5. Justify the adequacy of the ORIGEN2.1 computer code for calculations of the source terms for the spent fuel to be stored in the SENTRY system.

The applicant used the ORIGEN2.1 computer code for calculations of the source terms for spent fuel to be stored in the SENTRY DSS. The staff notes that the code was developed in the 1970's (reference 2 of SAR Chapter 6, "ORNL/TM-7175, A User's Manual for the ORIGEN2 Computer Code, Oak Ridge National Laboratory, Tennessee, July 1980"). The staff's understanding is that this computer code is no longer supported by the developer and significant deficiencies of the code and cross section set, which was distributed with the code, have been identified. Specifically, NUREG CR-6484 shows that the source term can be underestimated by as much as 15% using this code. The staff also notes that SAR Section 6.2.2.5 states: "*Since ORIGEN 2.1 does not calculate neutron energy spectra, the neutron energy spectrum of the most significant contributor is assumed to apply to all neutrons generated by the SNF.*"

The staff performed an independent analysis and calculated the source terms for the Westinghouse 17x17 PWR fuel assembly with one of the BECT combinations (60 GWd/MTU, 5% enrichment, 1.8 years of cooling time) as shown in Table 2.1-41 of the TS using the Origen/Arp module of the SCALE 6.1 computer code. The results of the staff's calculation show some significant differences with the source terms presented in the SAR. As such, the ORIGEN2.1 code and cross section data may not be appropriate for the source term calculations for the SENTRY system.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

6-6. Pertinent to the use of the DORT code for shielding calculation:

a. Clarify how the two-dimensional (2-D) importance obtained from the DORT code was used in the three-dimensional (3-D) shielding calculations for the storage and transfer casks,

b. Clarify how flux-to-dose rate conversion factors were used in the dose rate calculations, and

c. Demonstrate that the results are reliable and accurate.

SAR Section 6.2.1.1 states: "*Adjoint shielding calculations are performed by running the discrete ordinates code DORT (Reference 3) in 'adjoint mode' to obtain importance functions given flux-to-dose conversion factors. The adjoint shielding calculations are run for each SENTRY canister in the storage and transfer cask to establish sets of importance functions by energy group. These importance functions can be readily folded with source terms (i.e., the generic decay calculations) to arithmetically compute the storage or transfer cask side wall dose rate for a given cooling table state point.*"

SAR Section 6.4.2.2 further states: "*Source terms for the adjoint cases are the flux-to-dose conversion factors. The basis for these factors is the ANSI/ANS-6.1.1-1977 data described in Section 6.5.1.*"

However, it is not clear how the results of a 2-D importance were used in the 3-D code for dose rates calculations. It is not clear either how the flux-to-dose conversion factors were folded into the neutron and gamma importance functions.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

6-7. Describe how the ADSORB code uses decay heat and source terms to create cooling tables that can also assure meeting the dose requirements for the shielding evaluation of SENTRY DSS.

SAR Section 6.2.1.1 states: "*The tool used to automate production of the cooling tables is the code ADSORB (Reference 4), which combines the shielding and thermal acceptance criteria, the adjoint shielding result (which are canister unique importance functions or adjoint fluxes), and the generic decay library (decay heat, neutron and gamma source terms) to execute the steps described above. The output of the ADSORB code is the fuel cooling tables.*"

SAR Section 6.2.1.4 states: "*Generic PWR and BWR decay libraries for ADSORB UO_2 sources are created using ORIGEN-2.1.*"

It is not clear to the staff how the output of the ADSORB code was used in the shielding calculation. A step-by-step explanation on how this approach works may help the staff understand the method.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

6-8. Justify the adequacy of using 0.489 MTU per assembly over 144.0 inches for the shielding source calculation.

SAR Section 6.2.1.2, the applicant states that it assumed that the bounding fuel assembly design has 0.489 MTU/assembly over 144.0 inches. The staff notes that the TS includes other assembly designs as allowable contents that may be longer and/or have more heavy metal load per assembly. As such, it is not clear if the hypothetical assembly with 144.0 inches and 0.489 MTU will produce bounding source and shielding calculation results.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

6-9. Explain how the secondary gamma is considered in the shielding calculations.

The note to SAR Table 6.1-1 indicates that secondary gamma is included in the dose rate calculations. SAR Section 6.4.2 further states: "*Production of secondary photons via neutron capture in system materials is treated explicitly within the group to group transfer matrices as a down-scattering interaction form the individual neutron groups to the appropriate photon group.*" However, it is not clear how the secondary gammas were included in the calculation. Applicant needs to show how the secondary gammas, produced by (n, gamma) reactions, were considered in the shielding calculation.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

6-10. Demonstrate why it is acceptable to exclude the neutron source from a single neutron source assembly (NSA) in the shielding calculation.

SAR Section 6.2.2.5 states: "*The ^{238}Pu-Be and ^{241}Am-Be sources, however, have a significantly longer half-life, 87.4 years and 433 years, respectively. As a result, their source intensity does not decrease significantly before storage in the MPC. Since the ^{238}Pu-Be and ^{241}Am-Be sources may have a source intensity similar to a design-basis fuel assembly when they are moved to dry storage, only a single NSA is permitted for storage in canister. Since storage of a single NSA would not significantly increase the total neutron source in a canister, storage of NSAs is acceptable and detailed dose rate analysis of the neutron source from NSAs is not performed.*"

The conclusion for not needing to perform detailed dose rate analysis for a single NSA may not be valid if it is loaded in the periphery location of the canister.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

6-11. Justify the neutron source peaking factor used for fuel with burnup greater than 46 GWd/MTU.

In SAR Table 6.2-8 and Table 6.2-9, the applicant provided the neutron source peaking factors for the neutron sources at various burnup levels. The staff, however, found that the peaking factors at high burnup (BU ≥ 46 GWd/MTU) may be not conservative. Based on NUREG/CR-6801, the fuel assembly peaking factor for BU ≥ 46 GWd/MTU is 1.114 and the burnup dependent neutron source peaking factor is proportional to the fourth power of burnup. Thus, the neutron source should be $PF = BU^4 = 1.14^4 = 1.54$.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(d).

6-12. Clarify the units for both sides of the formula used to calculate the discharge of Co-60 or revise the equation to assure it is correct.

The applicant used a formula on SAR Section 6.2.1.5 (Document No. WSNF-230, April 2020) to calculate the Co-60 activity level at a given cooling time per fuel assembly. However, it seems that the unit of each term on the right side of the equation does not match the unit of left side of equation.

The staff needs this information to determine if the request meets the regulatory requirements of 10 CFR 72.236(d).

6-13. Provide the basis for the use of a 33,000 MWd/MTU source term for creating a data library for burnup from 15,000 to 40,000 MWd/MTU, and a 50,000 MWD/MTU reactor model for burnup from 40,000 to 60,000 MWd/MTU.

SAR Section 6.2.1.4 states; "*Because the fuel cooling tables include extended burnup fuel, two ORIGEN-2.1 data libraries are used.*" SAR Section 6.2.1.4 also states: "*Standard PWR library (PWR-US) for burnups from 15,000-40,000 MWd/MTU (this ORIGEN-2.1 library is created using a 33,000 MWd/MTU reactor model). Extended PWR library (PWR-UE) for burnups from 40,000-60,000 MWd/MTU (this ORIGEN-2.1 library is created using a 50,000 MWd/MTU reactor model)*."

First, the applicant should provide the basis for why it is acceptable to use a reactor model with 33,000 MWd/MTU to create a library for the burnup range from 15,000 to 40,000 MWd/MTU and use a reactor model with 50 GWd/MTU to create a library for the burnup range from 40,000-60,000 MWd/MTU.

Second, the staff notes that this approach involves extrapolations and using these values may give invalid results in the calculation of cooling time, since the source is not linear with respect to burnup. The staff's understanding is that the code has never been benchmarked to this burnup level. NUREG CR-6484 shows that the source term can be underestimated by as much as 15% using this code.

The staff needs this information to determine if the request meets the regulatory requirements of 10 CFR 72.236(d).

6-14. Justify using a Cobalt (Co-60) source term based on 11 grams to obtain the bounding source term for fuel to be stored in the storage cask.

SAR Section 6.2.2.4 states: "*For the core hardware gamma source for the Monte Carlo calculations of the storage cask, a calculation similar to the cooling table calculations in Section 6.2.1.5 is performed with the difference that a Co-60 source term based on 11 g, instead of 50 g, of cobalt is used and applied to the reference fuel of Section 6.2.2.1. To do this the values of Table 6.2-5 are replaced by the values of Table 6.2-22 and applying the 50 GWd/MTU value "F(B)" adjustment from Table 6.2-6 as a conservative value for the 48GWd/MTU fuel considered in Section 6.2.2.1.*"

SAR Section 6.2.1.5 states: "*This is done by modeling an assembly with 1.0 grams of cobalt present in the assembly fuel zone. The code then outputs the ^{60}Co activity at assembly discharge (i.e., at a cooling time of zero). The ^{60}Co activity level at other cooling times is determined simply by decaying the discharge activity levels using the ^{60}Co half-life of 5.27 years.*"

The applicant needs to explain the base unit of the Co-59 impurity, i.e., whether the one gram is total Co-59 per fuel assembly or per kg of fuel hardware. The quantity of fuel hardware per assembly is essential for calculating the Co-60 source in the fuel region. In addition, the applicant needs to justify it is conservative or bounding using this Cobalt-59 impurity value in the fuel during activation. Furthermore, the values in Table 6.2.-5 are not conservative values since it required shorter irradiation time for fuel with higher enrichments to reach the same burnup.

The staff needs this information to determine if the request meets the regulatory requirements of 10 CFR 72.236(d).

6-15. Demonstrate that converting the gamma source from the 18-group structure to the BUGLE-96 group structure using the conversion factors as defined in SAR Table 6.2-13 will produce the same or conservative shielding calculation results.

SAR Section 6.2.2.3 states: "*The gamma source term of the spent nuclear fuel assembly is composed of a fuel gamma source, fission product and actinide sources, and a light element activation source primarily associated with fuel hardware. Spectra are initially produced in the default 18-group energy spectrum of ORIGEN2. The source is then decayed and rebind into the BUGLE-96 (Reference 8) group gamma structure, via the transfer factors in Table 6.2-13.*"

It is not clear if such a gamma source conversion scheme will produce the same or more conservative results comparing a direct use of the calculated source terms. The staff is particularly interested in the validity of the transformation because the conversion factors seem to include further decay factors as stated in the SAR and these decay factors may affect the transformation.

The staff needs this information to determine if the request meets the regulatory requirements of 10 CFR 72.236(d).

6-16. Clarify if the Co-60 level for the non-fuel hardware (NFHW) is calculated based on zero cooling time or justify why further decay of the Co-60 is appropriate.

SAR Section 6.2.1.5 states: "*The assembly discharge fuel zone cobalt activation levels shown in Table 6.3-4 can be decayed, using a 5.27-year ^{60}Co half-life, to yield fuel zone cobalt activation level as a function of burnup and cooling time.*"

It is not clear to the staff if the Co-60 level is calculated based on zero cooling time. If not, justify why a further decay of Co-60 is appropriate since the decay of Co-60 has already been accounted in the source term calculation for the NFHW since cooling time is already used in calculating source term.

The staff needs this information to determine if the request meets the regulatory requirements of 10 CFR 72.236(d).

6-17. Provide the detailed model for calculation of the maximum dose rate around the storage cask under a tip-over accident.

The applicant stated that it has calculated the dose rate of the storage cask under a tip-over accident condition and states that the maximum dose rate is about nine times of the maximum dose rate around the storage cask. However, it is not clear how the cask is modeled, where the maximum dose rate is identified, and what is the value of the maximum dose rate. The applicant also needs to provide an estimated time for the accident recovery time in order to demonstrate that the maximum exposure under the tip-over accident is within the regulatory limit prescribed in 10 CFR 72.106.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.106.

Chapter 7: Criticality Analysis

7-1. Revise the fuel specifications in the Technical Specifications (TS) Section 2.2 "Functional and Operational Limits Violations" to include the detailed fuel specifications as provided in Table 7.2-1, "Fuel Characteristics and Assembly Class Definitions." Also, confirm that there are no other fuel assembly designs that are not listed in TS Table 2.1-10 or revise this table as necessary.

TS Section 2.2 "Functional and Operational Limits Violations" should include detailed fuel specifications as provided in Table 7.2-1 of the SAR per the regulatory requirements of 10 CFR 72.236(a).

In addition, SAR Section 7.2.1 states: "*The fuel assemblies indicated in Table 7.2-1 are grouped in different configurations and number of fuel rods in the lattice. <u>These fuel assemblies envelope a wide range of fuel assemblies</u>, as the maximum reactivity is directly based on, and can be directly derived from, the three main characteristics affecting reactivity…*" These statements seem to imply that there are other fuel assembly designs that are the intended contents but are not listed in TS Table 2.1-10. In accordance with the regulatory requirements of 10 CFR 72.236(a), specifications must be given for all intended contents even if the criticality safety is bounded by fuel assembly designs as listed in the TS.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124, 72.236(a) and (c).

7-2. Clarify the quantity of fuel debris, including ruptured fuel rods, severed rods, and loose fuel pellets, that is allowed to be loaded into each damaged fuel can (DFC).

The W37 canister allows loading of up to four DFCs that can hold ruptured fuel rods, severed rods, and loose fuel pellets. However, there is no limit on the quantity of the fuel debris including ruptured fuel rods, severed rods, and loose fuel pellets are allowed to be loaded into each DFC. The value of this limit will affect the k_{eff} of the system.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124, 72.236(a) and (c).

7-3. Revise Table 7.2-1 of the SAR to provide the maximum fuel assembly width for each fuel assembly design.

Table 7.2-1 of the SAR provides detailed specifications for the fuel assembly designs that can be stored in the SENTRY system. However, the maximum fuel assembly width is missing in this table. The fuel width is a key parameter for criticality safety analyses. It determines the water gap between the fuel assembly and the fuel basket cell walls and hence the k_{eff} value.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124, 72.236(a) and (c).

7-4. Clarify whether the SENTRY W37 canister allows for a mixed load of damaged fuel and non-fuel hardware (NFHW) in the same canister and provide a safety analysis for the mixed fuel load if so desired.

Both NFHW and damaged fuel (in damaged fuel cans) are allowable contents for this canister design. However, it is not clear whether a canister can have both damaged fuel and NFHW in the intact fuel assemblies. If a mixed load is intended, a criticality safety analysis is necessary to assess the required soluble boron concentration and a minimum burnup for burnup credit if credit is taken for both measures for criticality safety control.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124, 72.236(a) and (c).

7-5. Provide an assessment that demonstrates that it is conservative for the criticality safety analyses to assume the guide tubes as void when the NFHW is inserted in fuel assemblies for all canister types.

The applicant states in SAR Section 7.4.2.6: "*Non-fuel hardware can be stored in SENTRY W37 canister. These devices fill the space of the guide tubes. These components displace the bor[at]ed water therefore, the insertion of these components results in increased of reactivity. An assessment is performed with the guide tubes empty (void), in order to ensure that absorption in the material is neglected in the model and the presence of non-fuel hardware is enveloped in this assessment.*" The same statements are made in SAR Section 7.4.2.7 for the W21H canister, however, when inserted in the guide tubes of the fuel assemblies, these NFHW components displace borated water, i.e., they displace both moderator and boron in the water at the same time. Moderator and boron create competing effects on k_{eff} of a system flooded by borated water because moderator increases reactivity of a light water reactor fuel design and boron reduces reactivity. As such, insertion of these components may result in an increase or decrease in the k_{eff} depending on the concentration of soluble boron. Because the minimum soluble boron concentration requirements for these two canisters are significantly different, i.e., 1900 PPM for W21H and 2600 PPM for the W37 canister, it is not clear if the assumption of the voided guide tubes is true for either or both canister designs.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124 and 72.236(c).

7-6. Demonstrate that neglecting the grid spacers in the fuel assembly is conservative in the criticality safety analysis model.

In its Observation 6-2 (ADAMS Accession No. ML20345A121), the staff requested the applicant to demonstrate that it is conservative with respect to the criticality safety evaluation to neglect the grid spacers, even when the required soluble boron concentration is at 2600 PPM level for the W37 canister. In its response to the Observation (ADAMS Accession No. ML20276A295), the applicant states: "*The grids* *and sleeves have not been considered in the model. This modeling assumption is consistent with the generic analyses applicable to PWR UO_2 fuel contained in document WCAP-17483 'Westinghouse Methodology for Spent Fuel Pool and New Fuel Rack Criticality Safety Analysis'. In this calculation, multiple criticality calculations are documented with 0 ppm and 2500 ppm of soluble boron in the pool and with multiple burnup values with and without grids.*"

The staff reviewed this response and WCAP-17483. The staff notes that the analysis performed in WCAP-17483 covers the soluble boron up to 2500 PPM. The applicant needs to provide additional information demonstrating that the study can be extrapolated to the W37 canister which requires a minimum soluble boron concentration to be 2600 PPM to assure criticality safety. The boron concentration level is of particular concern to the staff because the calculated k_{eff} is currently close to the acceptance limit.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124 and 72.236(c).

7-7. Clarify what assumption was used for the reflector in the critical safety analysis models.

SAR Section 7.3.1, the applicant states: "*Specular reflection conditions have been considered, with a minimum gap of 30 cm of water from the outermost part of the cask and the limit of the model.*" The staff is seeking clarification on what the "minimum gap of 30 cm of water from the outermost part of the cask and the limit of the model" is.

The applicant needs to clarify the meaning of "the outermost part of the cask" and the meaning of "the limit of the model." The staff's understanding is that a model typically refers to the representation of a physical system in a mathematical equation or computer simulation. The applicant also needs to clarify whether fresh water or borated water was used as the reflector and explain why such an assumption is appropriate. These assumptions will affect the calculated k_{eff} of the system.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124 and 72.236(c).

7-8. Confirm that the center location of the W21 canister is not allowed to be loaded and revise all the loading patterns for the W21H canister in the TS to show this restriction.

The applicant provides the loading configuration for the W21H canister in Figure 7.2-4. The center location is marked with an X. The TS also include multiple loading patterns for the W21 canister. However, the legends for these loading patterns are inconsistent and none of them indicate that the center location of the basket is not allowed to be loaded. The loading configurations for the center location may affect the k_{eff} values.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124, 72.236(a) and (c).

7-9. Revise the code benchmarking analyses for the SCALE/KENO VI computer code with additional critical experiments or perform a normality test on the distribution of selected critical experiments to demonstrate that these experiments follow a normal distribution and make adjustments on the results as necessary if the distribution is not a Gaussian (normal) distribution.

The applicant selected 13 critical experiments from the International Criticality Safety Benchmark Evaluation Project (ICSBEP) Handbook and 12 critical experiments from NUREG/CR-6361. The staff notes that experiments No. 12 and No. 13 include plutonium and should be removed because they are not applicable to this system and will skew the benchmarking results of the W21H canister criticality safety analyses that assume unirradiated fuel in the canister. In addition, based on NUREG/CR-6361, the minimum number of samples is 25 for assuming a normal distribution or a normality test has to be performed and the results may have to be adjusted if the normality test does not pass. After removal of the two inapplicable experiments, the sample size becomes 23 and a normality test on the distribution of the experiments may become necessary if no new critical experiments are added. Code benchmarking is important because it determines the applicability of the computer code and cross section library for this application and the potential bias and bias uncertainty.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124 and 72.236(c).

7-10. Provide additional information on how the TSUNAMI assessment performed demonstrates similarities between the GBC-32 and W37 canister.

Concerning the applicability of the recommendations of ISG-8, Rev. 3 for burnup analyses for PWR fuel, the staff requested in Observation 6-1 (ADAMS Accession No. ML20345A121) that the applicant demonstrate the W37 canister is sufficiently similar to the GBC-32 cask based on recommendations made for the biases and bias uncertainties of the depletion code and criticality safety analysis code. In its response to the Observation 6-1 (ADAMS Accession No. ML20276A295), the applicant states:

"*Westinghouse has performed an assessment of the similarity between the GBC-32 system and the W37 canister. Performed with the TSUNAMI code, the Ck index between the designs is greater than 0.9, indicating that the two systems are similar to demonstrate the application of the guidance of ISG-8 is appropriate.*"

Though the applicant provided an overall C_k value for the similarity between the W37 canister and GBC-32 cask, there is no information on what parameters were used in deriving this C_k value. In accordance with the recommendations of ISG-8, Rev. 3: "This demonstration should consist of a comparison of system materials and geometry, including neutron absorber material and dimensions, assembly spacing, and reflector materials and dimensions, etc. This demonstration should also include a comparison of neutronic characteristics such as hydrogen-to-fissile atom ratios (H/X), energy of average neutron lethargy causing fission (EALF), neutron spectra, and neutron reaction rates. Applicability of the validation methodology to systems with characteristics that deviate substantially from those for the GBC-32 should be justified."

The applicant needs to provide additional information on how the TSUNAMI assessment it performed demonstrates similarities between these two cask designs and considers the above-mentioned parameters associated with neutronic characteristics.

In addition, the staff does not understand how the burnup credit analysis is performed for the canisters that contain both intact fuel and damaged fuel and burnup credit is taken only for the intact fuel. The applicant needs to provide additional information demonstrating the similarities of the GBC-32 to that of the W37 with burned intact fuel and unburned damaged fuel.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124 and 72.236(c).

7-11. Demonstrate that the credited minor actinide and fission product worth is no greater than 0.1 in k_{eff}.

One of the prerequisites of using the recommendations of ISG-8, Rev. 3 is to demonstrate that the credited minor actinide and fission product worth is no greater than 0.1 in k_{eff}. However, the applicant has not provided such a demonstration.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124 and 72.236(c).

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7-12. Revise and correct the following typographic errors in the SAR as necessary.

Section 7.4.2.2 SENTRY **W21H** Canister Configuration of the SAR states: "The SENTRY W37 canister will be flooded with 1900 ppm of borated water during the loading and unloading of the fuel assemblies in the pool, which is at atmospheric pressure." There seems to be a typographic error since this section of the SAR is related to the W37 canister only.

Section 7.4.2.7 "Non-Fuel Hardware inside SENTRY W21H canister" of the SAR states: "Non-fuel hardware can be stored in SENTRY **W37** canister." This seems to be a typographic error since this section of the SAR is related to the W21H canister only. If this is a typographic error, correct this.

The staff needs this information to determine if the SENTRY spent fuel DSS design meets the regulatory requirements of 10 CFR 72.124, 72.236(a) and (c).

Chapter 8: Materials Analysis

8-1. Provide drawings for the transfer cask neutron shielding shell.

The SAR does not contain drawings for the important-to-safety transfer cask component listed above. The staff needs this information to support its review of materials design, fabrication, and examination criteria.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(b).

8-2. Resolve the apparent discrepancy between the lead alloy specified for the transfer cask gamma shield and the cited mechanical properties.

The mechanical properties of the transfer cask lead gamma shield do not appear to be applicable to the specified alloy. SAR Section 4.1.4 states that the lead gamma shield is constructed of chemical copper lead in accordance with ASTM B29, however, the strength properties in SAR Table 4C-8[2] appear to be associated the reported values for high purity lead. The staff notes that the ultimate tensile strength value for chemical copper lead in the cited reference is about 15 percent lower than that of the high purity lead at 100°F.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(b).

8-3. Clarify if the strength of the bolting that attaches the fuel basket peripheral guides is relied on in any mechanical loading scenarios and, if so, justify the absence of a materials standard to provide minimum mechanical property requirements. In addition, demonstrate that the bolting is not subject to potential overload due to differential thermal expansion between the bolt and guide materials.

SAR Section 4.2.1.2.3 states that the peripherical basket components provide boundary support to the basket and are in the load path in some loading cases. It is unclear if the attachment bolts have a role in these loading cases. If so, define the material procurement controls (e.g., material standard) that ensures that the bolts are capable of performing their structural function.

In addition, the Observation 7-3 requested information to demonstrate that stresses due to differential thermal expansion cannot cause overload of all ITS bolting. The response did not address the bolting that attaches the peripheral guides.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(b) and (c).

8-4. Provide additional justification for why the storage cask lid bolts do not require acceptance testing to verify that procured material has adequate fracture toughness.

SAR Section 8.3.2.2 states that the ASTM A193 Grade B7 lid bolts have adequate fracture toughness based on the material's low ductile-to-brittle transition temperature and the fact that the bolts are not loaded in shear during a tip over event. As a result, there are no toughness testing requirements in the SAR for the lid bolts.

The staff notes that the purpose of the fracture toughness testing is to ensure that the product received from the supplier was appropriately manufactured such that the expected toughness is achieved. The bolting alloy is not one that is generally considered to be exempt from fracture toughness testing (e.g., the exemption criteria in

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[2] References Tietz, T. E., "Determination of the Mechanical Properties of a High Purity Lead and a 0.058 % Copper-Lead Alloy," WADC Technical Report 57-695, ASTIA Document No. 151165, Stanford Research Institute, Menlo Park, CA, April, 1958.

Division 3 of the ASME Code), and the cited design approach in the SAR[3] specifically notes that ductile bolt behavior is assumed and bolting material must meet ASME testing criteria. Finally, SAR Section 4.6.5.7.2 states that, in a tipover event, the lid bolts must be able to resist the impact of the canister onto the lid to prevent the canister from sliding out of the storage cask.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(b) and (d).

8-5. Provide a technical justification for the allowable temperature for the aluminum basket components.

SAR Table 5.2-1 provides an allowable temperature of 1100°F for the aluminum basket components under normal, off-normal, and accident conditions. Provide the additional detail to support the limit, as follows:

- A technical basis for the criterion provided in Note 2 of Table 5.2-1, which was used to calculate the limit. It is unclear how this criterion was chosen to ensure that the aluminum components will be capable of fulfilling their safety function.

- A justification that the cited limits are consistent with the technical literature on the melting range of aluminum alloys. Aluminum melts over a range of temperatures; alloy 1100 melts between 1190°F and 1215 °F, and alloy 6063 melts between 1140°F and 1210°F[4].

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(b) and (c).

8-6. Demonstrate that the structural and shielding analyses adequately account for the allowable off-normal and accident temperatures for concrete that exceed the maximum surface temperature limit in ACI 349-13.

ACI 349-13 states that concrete *surface* temperatures shall not exceed 350°F for accidents and other short-term events. This standard provides for higher temperatures if testing is performed to evaluate the reduction of strength and if the reduction is applied to design allowable stresses. SAR Table 5.2-2 applies the 350°F criteria to the *average* temperature across the entire concrete cross section, rather than the surface. This results in allowing significant portions of the concrete cross section to exceed the ACI threshold.

The response to Observation 7-1 (ADAMS Accession No. ML20276A295) provided information on the expected strength and weight loss of concrete at elevated temperatures; however, the staff needs the following additional information:

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[3] NUREG/CR-6007, "Stress analysis of closure bolts for shipping casks," Lawrence Livermore National Laboratory, Livermore, CA, January 1993.
[4] Kaufman, J.G. (2016). Fire Resistance of Aluminum and Aluminum Alloys & Measuring the Effects of Fire Exposure on the Properties of Aluminum Alloys. ASM International.

Structural performance

Describe how the structural analysis accounts for a reduced concrete strength for all portions of the concrete cross section that exceed 350°F in accidents and other short-term events. ACI 349-13 Section E.4.3 requires that the strength reduction be applied to the design of concrete structures that exceeds the allowable surface temperature. The response to Observation 7-3 provided data that showed a strength loss approaching 20% at the 650°F maximum allowable local temperature defined in the SAR, while the staff notes that other sources report potentially greater strength reductions (e.g., 35% reduction at 572°F for 2 days)[5]. The response suggested that concrete hardening during curing over 10 years would offset any strength loss; however, that is not applicable to concrete that has not yet undergone that hardening.

Shielding Performance

Describe how the shielding analysis accounts for potential reduced neutron absorption for all portions of the concrete cross section that exceeds 350°F in accidents and other short-term events. The response to Observation 7-3 stated that the weight loss of concrete at elevated temperature is minimal; however, a quantitative basis was not provided to describe how that weight loss due to dehydration reactions may affect shielding performance. Peterson (1960)[6] measured significant increases in neutron flux through concrete as water was lost during elevated temperature exposure.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(b) and (d).

8-7. Provide the basis for selecting the Zircaloy-4 cladding in the W17xW17 OFA fuel type as the most limiting assembly for the structural analysis.

SAR Sections 4.6.2 and 5.2.2 describe the mechanical properties used in the structural analysis of the fuel cladding. The SAR states that, consistent with prior amendments of the FuelSolutions storage system, the Zircaloy-4 W17x17 OFA fuel type has the lowest buckling capacity. Therefore, this fuel type was chosen to provide the most conservative structural analysis.

The SAR does not describe the basis for why fuel assemblies that use the M5 cladding alloy introduced in the new amendment were also determined to be bounded by the Zircaloy-4 W17x17 OFA fuel. Provide the basis for why the previously identified limiting fuel assembly type with Zircaloy-4 cladding is bounding, including the consideration of the mechanical properties of M5.

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[5] M.K. Kassir et al., "Thermal Degradation of Concrete in the Temperature Range from Ambient to 315°C (600°F)," Brookhaven National Laboratory Report BNL 52384, October 1996.
[6] E.G. Peterson, "Shielding Properties of Ordinary Concrete as a Function of Temperature," Hanford Atomic Products Operation Report HW-65572 for the Atomic Energy Commission, August 1960.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(b) and (c).

8-8. Clarify and provide a technical justification for the fuel cladding oxide thickness used in the structural analyses of the fuel assemblies.

SAR Section 3.2.1 states that, for assemblies exceeding a burnup of 45 GWd/MTU, the fuel cladding oxide layer thickness is limited to 70 μm for the purposes of the structural analyses. The SAR states that general licensees have the responsibility of ensuring that oxide thickness is not exceeded.

The staff does not consider the 70 μm oxide layer thickness for high burnup fuel cladding to be consistent with available data (prior 72-1026 amendments considered a 100 μm thick oxide layer to be reasonably limiting per PNL-4835[7]), and thus it is not clear what practical controls could be put into place to allow a general licensee to verify that the limit is not exceeded.

As a result, the staff requests either (1) a technical justification that demonstrates that the identified criterion is a reasonable oxide layer threshold for all allowable high burnup contents or (2) revise TS Tables 2.1-6 through 2.1-9 to state that the allowable cladding condition is limited to a 70 μm oxide layer thickness for high burnup fuel and provide additional CoC requirements that describe how licensees can practically verify that the loaded fuel meets the criteria, considering the methods that would be used and what sampling approach would provide a reasonable estimate of oxide variability (e.g., rod-to-rod variability, variability along the axial direction of the rods, variability associated with increased oxide thickness in areas of fretting).

In addition, the staff requests clarifying information with respect to which analyses the 70 μm oxide layer limit applies. In the prior amendments of the FuelSolutions CoC (e.g., W21 Canister Final SAR, Rev. 5), the 70 μm limit appears to have been used only to evaluate the case of buckling of higher burnup fuel, while the remaining fuel performance analyses assumed a 100 μm thick oxide layer. The SENTRY SAR does not appear to specifically address this. Revise the SAR to clarify the oxide layer thicknesses assumed for all the fuel structural analyses.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(b) and (c).

8-9. Provide the qualification data that demonstrates that the neutron absorber material is durable in the service environments and is capable of performing its criticality control function.

SAR Section 8.10.1 states the qualification approach of the proprietary metal matrix composite (MMC) neutron absorber is consistent with that described in NUREG-2215. However, no data was provided to demonstrate that the chosen material is capable of

[7] A. Johnson Jr. and R. Gilbert, "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gas," September 1983, Pacific Northwest Laboratory, PNL Document No. PNL-4835.

meeting the qualification acceptance criteria. As a result, the staff requests the following material property information for the proposed absorber material:

- Measured strength and ductility,
- Measured thermal properties (i.e., those used in the thermal analyses), including potential effects of anisotropy of the clad MMCs,
- Porosity data, including both total and interconnected porosity,
- Test data that demonstrates resistance to blistering in the canister drying process, and
- Data on the boron-10 distribution (including uniformity) that justifies the attenuation properties used in the criticality analyses.

In addition, provide the details of the test methods and sampling approaches used to generate each of the above properties. The staff notes that publicly available documents contain some information on the proposed MMC material from the same manufacturer; however, the available data does not address all of the requested information above and, in some cases, the specific alloys used to fabricate the proposed MMC may not be identical to those discussed in the literature.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.124(b).

8-10. Clarify the required maintenance of the W180 storage cask and W110 transfer cask.

SAR Section 12.2.1 and Table 12.2-1 summarize the maintenance activities of the W180 storage cask. In some cases, the SAR description is vague or inconsistent. In order to allow the staff to evaluate the adequacy of these activities, provide the following clarifying information:

- Clarify how many casks are included in the annual inspection discussed in SAR Section 12.2.1 item No. 4,
- Clarify the coverage of the annual inspection, such as whether the inspection is capable of assessing potential corrosion or coating performance of the lid and lid bolts (and, if not, why such an inspection is not considered necessary),
- Clarify what specific “damage” is being inspected for in the 5-year inspection of one of the interior surfaces (SAR Section 12.2.1, item No. 4), and
- Revise SAR Table 12.2-1 to reflect all the activities included in SAR Section 12.2.1 (e.g., the table does not mention the 5-year inspection of one of the interior surfaces, the inspections required if the cask is reused, or the inspection of coating damage).

SAR Section 12.2.2 and Table 12.2-2 summarize the maintenance activities of the W110 storage cask. In some cases, the SAR description appears to be inconsistent. Provide the following clarifying information:

- Revise SAR Table 12.2-2 to reflect all the activities included in SAR Section 12.2.2 (e.g., the table does not mention the inspection of the rupture discs or the annual inspection of all accessible materials and welds).

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.234(b) and 72.236(g).

8-11. Demonstrate that the W21H canister design with the heat dissipation fins is capable of allowing inspection for degradation that could challenge the confinement boundary.

SAR Section 8.16.3.3 states that ISFSIs located in areas subject to atmospheric conditions which may degrade the storage cask or canister should be evaluated by the licensee on a site-specific basis to determine the appropriate frequency for maintenance and inspections. The CoC 72-1026 renewal application (ADAMS Accession No. ML20315A017) also considers the potential need for canister inspections by proposing an aging management program to inspect accessible welds and weld heat affected zones.

Given that a potential need to inspect the W21H canisters has been identified in the SAR and the CoC 72-1026 renewal application, it is unclear to the staff how the design allows for confinement boundary inspections. The heat dissipation fins are expected to obscure portions of the canister welds and heat affected zones, and the fin spacing would not appear to practically allow access for the volumetric examinations proposed in the renewal aging management program.

Provide details of inspection methods and delivery systems that are capable of accessing all the welds (fin-to-shell welds and shell seams). Describe how visual inspections could be reasonably qualified with adequate distance, angle, and lighting and how potential follow-up volumetric inspections could be qualified to effectively survey the fillet weld profile of the fin attachments and shell seam welds and heat affected zones between the fins (i.e., the proposed inspections in the proposed aging management program in the SENTRY renewal application). In addition, describe how the width of the weld heat affected zones and locations of the fin attachments are considered in the response.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.236(g).

Chapter 17: Operating Controls and Limits Analysis

17-1. Provide additional information related to LCO 3.1.5.

TS Section 3.1, Canister Integrity, LCO 3.1.5, Canister Transfer Time Limit to Storage Cask, states a completion time limit based on the Active Cooling System (ACS) Program. The staff noted that neither the SAR nor TS specify an actual completion time limit. This information is needed to assure that the ACS required actions of the associated conditions shall be met within a specified time limit.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.122(h).

17-2. Provide additional information related to the ACS.

TS 5.3.6, Active Cooling System Program, establishes the need for the ACS to be operable during LOADING, TRANSFER OR UNLOADING OPERATIONS, as well as, longer ACS restoring times in case of loss of the ACS, both depending on the total heat load per CANISTER, for both W37 and W21H CANISTERS. The staff notes SAR Table 5.3.1 specifies time limits, required actions, and completion times for a transfer cask containing a canister loaded with the design basis heat loaded fuel and the basis to recover the system should it fail. Additionally, SAR Section 1.2.1.5.3 and Section 5.4.2.1 describes the ACS as having critical operational features of equipment redundant chiller and pump volumetric flow rate of 31.7gpm at 68 °F water temperature without describing the systems performance requirements nor establishing a safety designation/classification (e.g., important-to-safety) of the system. Lastly, the staff notes there are no conditions, time limits, required action and completion time should the ACS not be restored to operable within the required time limits. The staff requests:

a. relocating Table 5.3.1 within the TS LCO Section (vs Administrative),

b. describing/classifying the ACS as important-to-safety and specify the ACS performance requirements, and

c. establishing conditions, time limits, required actions and completion times should the ACS not be restored to operable within the required time limits.

This information is needed to assure that the ACS required actions of the associated conditions shall be met within a specified time limit.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.122(h).

17-3. Provide additional information related to TS Section 5.3.9, Storage Cask Monitoring Program.

TS Section 5.3.9 establishes administrative controls and procedures to assure that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions in the unlikely event of a full blockage of all STORAGE CASK inlet and outlet vent screens during STORAGE OPERATIONS. The staff noted Acceptable means of monitoring the STORAGE CASK include periodic visual inspection of all STORAGE CASK inlet and outlet vent screens. However, the applicant did not define “periodic” interval. The staff requests the applicant to specify the frequency associated with visual inspection of all STORAGE CASK inlet and outlet vent screens.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.122(h).

17-4. Provide additional information related to FLAMMABLE AND EXPLOSIVE REACTIONS.

SAR Section 8.14, FLAMMABLE AND EXPLOSIVE REACTIONS describes the need to monitor the gas in the canister cavity and purging (when necessary) prior to and welding the closure lid to the canister shell in order to eliminate the potential for a hydrogen gas burn event and assure the safety of the public and plant personnel. The staff notes Hydrogen generation is monitored and controlled prior to and during welding operations in accordance with the operations instructions in SAR Chapter 11. In addition to SAR Chapter 11, the operations instructions should be described in the TS’s under Administrative Controls.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.122(b).

17-5. Provide additional information related to TS Section 5.3.3.

TS Section 5.3.3 provides a means for processing changes to the Bases of Technical Specifications. The staff notes that TS Section 5.3.3(d), states, changes that do not meet the criteria of 5.3.3.2 above shall be reviewed and approved by the NRC prior to implementation. Staff notes that there is no TS criteria 5.3.3.2.

The staff needs this information to determine if the SENTRY DSS design meets the regulatory requirements of 10 CFR 72.24 and 72.48.