Enclosure 3

Westinghouse's Responses to the NRC's Requests for Supplement Information (RSIs) and Observations (OBSs) (Non-Proprietary)

October 2020

NRC Introduction

By letter dated April 30, 2020 (Agencywide Documents Access and Management System [ADAMS] Accession No. ML20121A196), as supplemented on June 5, 2020 (ADAMS Accession No. ML20164A120), Westinghouse Electric Company LLC (Westinghouse) 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 supplemental information identifies information needed by the NRC staff (the staff) in connection with its acceptance review of the SENTRY™ Dry Storage System application to confirm whether the applicant has submitted a complete application in compliance with regulatory requirements.

Of note, an additional consideration the staff would like to call attention to is the applicant's decision to commit to American Society of Mechanical Engineers (ASME) Section III Division 3 Subsections WD and WC. As the agency has not yet endorsed this section of the code, committing to this code may have impacts on the timeliness of the review for multiple disciplines.

Westinghouse Response:

Although this is not presented as an RSI (or an Observation), Westinghouse would like to respond to this additional consideration. This topic was discussed in the pre-submittal meeting on November 21, 2019, and it was generally uinderstood that the differences between the past practice of using parts of ASME Section III Division 1 for spent fuel storage systems, and the use of the newer ASME Section III Division 3 which is written specifically for spent fuel storage and transportation packages are minimal and summarized below. As we also discussed, Westinghouse would like to point out that the use of division 3 is more prescriptive and includes code cases that were needed to be considered when using division 1. So no impact on the structural, thermal…rules to demonstrate compliance.

The differences between ASME Boiler & Pressure Vessel Code Division 1 (ASME III Division 1) and Division 3 (ASME III Division 3), were addressed by a Technical Paper written by a current Westinghouse employee for the 10th International Conference on Environmental Remediation and Radioactive Waste Management (ICEM). The Technical Paper, "*ICEM05-1417 - U.S. Code Developments for Spent Fuel Storage Containers*," illustrated the differences between ASME III Division 1 and ASME III Division 3. Since the paper was written there have been further developments by ASME which have resulted in a complete code that addresses the construction of dry fuel storage and transportation casks or canisters which ASME mutually calls "containments." This white paper includes some excerpts from Technical Paper ICEM05-1417 and the changes that in the ASME code that have transpired since the paper was written in 2005.

Prior to the development of ASME III Division 3, "*Containment Systems for Transportation and Storage of Spent Nuclear Fuel and High-Level Radioactive Material*," no dedicated code was available for the construction of spent nuclear fuel (SNF) or high-level waste (HLW) storage and transportation containments. Therefore, commercial manufacturers used portions of ASME III Division 1, typically as follows:

- ASME Subsection NB, "*Class 1 Components*" for the containment shell design
- ASME Subsection NC, "*Class 2 Components*" for containment component design

- ASME Subsection NG, "*Core Support Structures*" for the fuel basket design

- ASME Subsection NF, "*Supports*" for simple linear supports in the containment

As a result, the NRC reviews and licenses each containment design separately without regard to any standardized rules other than the required criteria based on 10CFR72.

ASME III Division 1 does not address all the regulatory requirements in 10CFR72 and therefore is not well suited for SNF and HLW containments. ASME III Division 1 rules address features common in power plants – specifically the loads created by pressure and temperature in vessels and piping. The governing criterion of SNF and HLW containments, however, is not pressure and temperature but rather configuration control to prevent criticality and containment of the radioactive materials within the containment to protect the environment, workers, and the public from radioactive releases.

To supplement the regulatory requirements that ASME III Division 1 does not address, the NRC issued interim staff guidance (ISG) as follows:

- ISG-4, "*Cask Closure Weld Inspections*" provides guidance for performing either volumetric or dye penetrant closure weld inspections since code required radiographic inspections cannot be performed after a welded containment is loaded with SNF or HLW.

- ISG-10 "*Alternatives to the ASME Code*" addressed the lack of an industry code or standard for the design and fabrication of dry cask storage systems. The NRC stated that ASME III Division 1 is an acceptable standard for the design and fabrication of dry storage casks but that dry storage casks are not pressure vessels and therefore the ASME code could not be implemented without allowing some alternatives to its requirements. The ISG recommended that applicants propose alternatives to ASME III Division 1 that addressed the regulatory requirements.

- ISG-18, "*The Design and Testing of Lid Welds on Austenitic Stainless-Steel Canisters as the Confinement Boundary for Spent Fuel Storage*" provides guidance for the design of final closure welds on the lid of a welded containment. On a SNF or HLW welded containment, the lid joints cannot be tested to verify there is no potential leak path like a typical ASME III Division 1 vessel because the lid welds occur after the containment is loaded with SNF or HLW.

- ISG-25 "*Pressure and Helium Leakage Testing of the Confinement Boundary of Spent Fuel Dry Storage Systems*" provides guidance for evaluating the helium leakage testing that ASME III Division 1 does not address.

The primary difference between Division 1 and Division 3 is that Division 1 is concerned about pressure and temperature whereas Division 3 is concerned about configuration and containment.

To address these issues, ASME issued code cases for ASME III Division 1 which eventually were incorporated into the development of ASME III Division 3. ASME III Division 3 is written specifically to provide code rules for the construction of safe storage and transportation of SNF and HLW containments. It ensures compliance with 10CFR72, standardization of the rules for future containments, assists the NRC in the review and approval process, and ensures that the essential criteria related to SNF and HLW storage and transportation is adequately addressed.

ASME III Division 3 consists of the following subsections:

- Subsection WA — General Requirements for Division 3
- Subsection WB — Class TC Transportation Containments
- Subsection WC — Class SC Storage Containments
- Subsection WD — Class ISS Internal Support Structures

The following table shows benefits of ASME III, Division 3 over the current use of ASME III, Division 1.

DIVISION 3	DIVISION 1
1. WB/WC is written specifically for transportation and storage containments and includes only options and alternatives applicable to SNF or HLW containments.	1. NB/NC is written for pressure components and includes alternatives applicable to pressure components (e.g., NB/NC-7000, Overpressure Protection) not applicable to containments. • Use of NB/NC needs interpretation in selecting appropriate rules.
2. WC includes rules needed for containment closures. • Includes joint designs and fabrication rules to accommodate the final closure weld following fuel loading. • Provides NDE criteria for closure welds that are made after the containment is loaded	2. NB/NC does not include rules needed for final closure welds of containments: • Pressure vessel rules do not have provisions for NDE methods and final certification that accommodate final end closures that may need to use partial penetration welds and must be performed after the SNF or HLW is loaded.
3. WB/WC includes rules for conducting helium testing of the containment.	3. NB/NC does not include any rules for helium testing of the vessel.
4. WC/WD contains specific rules to ensure invokes WA, General Requirements, which is consistent with how containments are currently being licensed.	4. NB/NC/NG invokes NCA, General Requirements, which includes certification process for design and fabrication of piping and vessels that is not consistent with SNF/HLW containment licensing.
5. WC/WD requires no deviations or exceptions.	5. NB/NC/NG/NF requires requests for deviations that must be approved by the NRC.
6. WD addresses boundary parameters between the basket and the containment shell.	6. NG addresses the boundary parameters of a reactor core which has no correlation to SNF or HLW containment.

7. WD addresses the configuration of basket components to maintain criticality control.	7. NG does not address criticality control.
8. WD contains specific rules for fuel basket accident loading. • Provides rules for buckling of the basket plates from a drop accident. • Provides rules for bolting connections made in the basket. • Provides rules for linear basket components.	8. NG is specifically for a reactor vessel core and does not contain specific rules for SNF or HLW basket. • NG does not contain rules for buckling since the reactor core is not subject to drop accidents. • NG does not contain all the rules for bolting of SNF or HLW basket components. • Parts of NF are used to address linear basket supports.
9. Division 3 rules address Certification applicable to SNF and HLW containment regulatory requirements.	9. Division 1 rules address Certification applicable to piping and vessel components.
10. Division 3 rules do not require stamping.	10. Division 1 rules require stamping of components.
CONCLUSION	
Use of Division 3 provides a complete, repeatable, approvable set of rules that support regulatory review since they address requirements in 10CFR72.	Use of Division 1 requires: • Interpretations by the manufacturer to apply the code provisions correctly for many applications not related to piping or vessels. • Use of the NRC interim staff guidance or alternative methods to ensure compliance with 10CFR72 requirements. • Interpretations by the NRC since it does not address provisions in 10CFR72

RSI 2-1

Provide additional information to support the benchmarking effort used to simulate the W180 storage cask with respect to end drop and tipover scenarios.

The applicant used analytical methods to calculate the response of the W180 and benchmarked them against information in NUREG/CR-6608. Appendix 4B of the FSAR, “End and Side Drop Analysis Methodology Evaluation,” states that the analytical models were benchmarked against the steel billet drop test results presented in NUREG/CR-6608. Based on this benchmarking, the applicant concluded that their simulation of the W180 will accurately predict the response of the W180 and therefore it is acceptable for use in predicting the impact loads resulting from the postulated storage cask end drop and tip over scenarios. Sections 4.1.5.1, 4.6.5.6.1, and 4.6.5.7.1 of the FSAR discuss the end drop and tipover evaluations performed. In these sections, the applicant discusses design features credited for the mitigation of postulated end drop and tipover events.

The staff notes that physical test data used to validate or benchmark an analytical model for the W180 should be similar to ensure that the analytical model accurately predicts the behaviors of interest for the package (see Appendix 4A of NUREG-2215 for additional details). For instance, a package with similar design features credited to mitigate the effects of postulated end drop and tipover events as the W180 storage cask should be used for benchmarking. The applicability of the steel billet used in the drop tests presented in NUREG/CR-6608 as a benchmark for the W180 storage cask should be further substantiated. In addition, the W180 also has a construction that is composed of several shells and lids, which is unlike the solid cylinder in NUREG/CR-6608. Therefore, the W180 would be expected to have a different response behavior than the steel billet described in NUREG/CR-6608 in an end drop or tip-over scenario.

To allow the NRC staff to make its safety determination with respect to end drop and tipover evaluations, the following information is requested:

a. Additional benchmarking information. Analytical models used to represent the W180 storage system must be of good “quality,” which includes being based on a physical set of data from a system that is similar to the W180. In particular, physical testing of a given system should be representative of the W180. Any model used for simulation of a storage system needs to accurately capture items important to safety such as bolt behavior, plate vs shell behavior, welds that are three dimensional in behavior rather than just two dimensional, key design features as configured within the W180, plastic strain data at the confinement boundary etc. The solid cylinder in NUREG/CR-6608 does not capture all of the features and physical behaviors of the W180 storage system that are relied upon to ensure safety with respect to criticality, shielding and dose during end drop and tipover scenarios.

b. Input and output files of the analytical models used to simulate the storage system. Input and output files describe complex nonlinear behavior found in an end drop or tip-over analysis whose accuracy cannot be determined sufficiently solely by text descriptions or screen shots in an FSAR. A determination of reasonable assurance cannot be made without the input/output files used in these analyses.

This information is required to comply with 10 CFR 72.122(a)(b) and 72.236 (b).

Westinghouse Response:

The W180 storage cask dynamic analyses (including end drop and tipover) are described in Calculation Note DDRWM-CN-00543-GEN rev. 0 “SENTRY™ W180 Storage Cask Dynamic Calculations”. This Calculation Note includes the LS-DYNA computer code verification under Westinghouse’s QA Program by comparing numerical results for series of dynamic simulations with the solutions.

According to NUREG-2215, Section 4.5.4.2, the review validation of the analytical model has followed acceptance criteria consistent with NUREG/CR-6608 “Summary and Evaluation of Low-Velocity Impact Tests of Solid Steel Billet onto Concrete Pads”. The methodology described in Chapter 7 of NUREG/CR-6608 was applied and is summarized in Appendix 4A and 4B of the SAR.

As discussed in SAR, Section 4.6.5.6.1, the canister is supported by [ ](a,c) (internal impact limiters), which are designed to crush and limit the canister g-load during the postulated end drop event. [ ](a,c) are selected for this application ([ ](a,c) is a well-known phenomenon with predictable response). [ ](a,c) impact limiters have already been used for package cushioning in many applications, including shipping casks[1] for nuclear application. The FuelSolutions™ storage system (CoC 72-1026) [ ](a,c) during a postulated end drop event. The analytical model used to predict the W150 cask loads was also validated using the LLNL billet test measurements, in accordance with NUREG/CR-6608. 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.

An additional benchmarking has been performed by hand calculation. [

](a,c)

Shock-absorbing devices ([ ](a,c)) for tipover event have also been used previously in other application accepted by NRC. In this application, the benchmarking is focused in comparing peak accelerations in every case of billet drop analyzed by LLNL in NUREG/CR-6608.

As described in Calculation Note DDRWM-CN-00543-GEN, a great effort has been made in sensitivity analyses regarding the mesh and the elements (shell or solid) of the [ ](a,c) used as tipover devices, and the results are satisfactory. A further benchmarking is not considered necessary for a non-credible event (non-mechanistic tipover according to NUREG-2215).

Input files are shared in order to help the understanding of the analytical models used to simulate the storage system.

[1] Shappert, L. B., A Guide for the Design, Fabrication, and Operation of Shipping Casks for Nuclear Applications, Oak Ridge National Laboratory, ORNL-NSIC-68, February 1970.

Documents In Reading Room Folder for RSI 2-1

1. WDD-CN-00546-GEN, Revision 1.
2. Energy Absorption of Pipes.pdf
3. Various input files in descriptive folders.

RSI 2-2

Provide stability calculations that support the W180 storage cask when subjected to accident conditions.

Table 4.4-4 of the SENTRY [DSS] FSAR lists the analytical approach used to evaluate the W180 Storage Cask. It mentions stability evaluations for various scenarios such as earthquake, flooding, tornado and explosion and refers to appropriate sections in the FSAR for additional information. However, the information provided in the FSAR does not have sufficient detail for the staff to reach a safety finding. The applicant is requested to provide the data/calculations labeled as “Stability Evaluation” for the accident conditions identified in Table 4.4-4.

This information is necessary to allow compliance with 10 CFR 72.122 and 72.103.

Westinghouse Response:

Stability calculations for the SENTRY DSS are performed in Westinghouse calculation note WDD-CN-00546-GEN, Revision 1. This document will be made available for review by the NRC.

<u>Documents In Reading Room Folder for RSI 2-2</u>

1. WDD-CN-00546-GEN, Revision 1.

OBS 3-1

Provide the SSCs ITS temperatures and canister pressures during accident transfer conditions, especially a condition that challenges the shielding material.

There was limited information about canister pressures and SSCs ITS temperatures during accident transfer conditions that challenge concrete and the lead and epoxy resin of the W110 transfer cask, which are used for shielding. For example, it is possible fire could impact the transfer cask, considering that SAR Sections 3.2.2.3.6 and 5.4.1.2.7 stated there is enough fuel within the vehicle that could result in a fire.

This information is needed to determine compliance with 10 CFR 72.236(b).

Westinghouse Response:

The SENTRY W110 transfer cask used for canister transfer operations is designed and licensed to operate inside of a facility licensed under a 10 CFR 50/52 license. It is not designed to operate outside of such a facility building therefore, this component remains under 10 CFR 50/52 regulations. As the accidents and off-normal conditions to be considered are consistent with the bases of the associated 10 CFR 50/52 license, no accident conditions are necessary to be considered in the design basis for the W110 transfer cask. Only one off-normal event is assessed: “active cooling system failure,” as discussed in Chapter 16 of the SAR.

The canister transfer from the building to the ISFSI is performed with the W180 storage cask and a heavy haul trailer. Fire and other accident and off-normal events have been assessed for the W180 storage cask in it’s loaded configuration and the temperatures and internal pressure for those associated ITS SSCs are provided in the appropriate SAR Chapters for those parameters (Chapter 4 for structural, Chapter 5 for thermal and pressure conditions) and summarized in Chapter 16 of the SAR.

OBS 3-2

Clarify in SAR Chapter 3 and Chapter 5 that the 80°F ambient temperature specified in the Chapter 5 thermal analyses is one modeling temperature used to demonstrate SENTRY dry storage system (DSS) performance at a specified normal condition, rather than a representation of a bounding normal condition ambient temperature to be used by licensees.

SAR Section 5.3.1 stated "… a long-term annual average design temperature of 80°F is selected for the SENTRY DSS normal storage, which bounds all site locations in the contiguous United States" and SAR Section 3.2.2.1.1 indicated that a licensee is to confirm that a site's annual average ambient temperature is bounded by 80°F. However, it is noted that the thermal analysis performed by a licensee is to demonstrate safe DSS operation at all normal conditions, including high seasonal (e.g., summer) ambient temperatures, which may be greater than 80°F.

This information is needed to determine compliance with 10 CFR 72.236(f).

Westinghouse Response:

The normal ambient temperature corresponds to a long-term annual average[2] design temperature as the ambient temperature which the cask is subjected to throughout its lifetime. It is used to determine long term temperatures of the fuel cladding, concrete, and other ITS components and to evaluate long term fuel cladding and concrete integrity over component lifetime. The summer or winter temperature periods are covered by the off-normal three day averaged ambient temperature of -40ºF and 104ºF, which are expected to bound most of the US sites.

[2] As per climate.gov, the annual average Normal Daily **Mean** Temperature and the annual average Normal Daily **Maximum** Temperature (https://www1.ncdc.noaa.gov/pub/data/ccd-data/nrmavg.txt) is in most cases below 80ºF except for the last 11 weather stations, where there are no NPPs. For the maximum temperature, approximately 7% are above 80ºF

OBS 3-3

Demonstrate with supporting analyses and discussion that the thermal models discussed in the thermal chapter accurately model the thermal-related phenomena for steady-state storage and transient operations.

SAR Chapter 5 described both three-dimensional (3D) and two-dimensional (2D) FLUENT models that were used in the SAR thermal analyses and which are the basis for the reported Important-to-Safety (ITS) component temperatures. The thermal analyses considered the W37 canister and W21H canister in the W180 storage cask, the W37 canister and W21H canister in the W110 transfer cask, and various thermal transient analyses (e.g., vacuum drying).

a. It is noted in the thermal analysis that the PCT results for canister W37 (45 kW) and canister W21H (65 kW) were at similar temperatures. The chapter does not demonstrate how the unique thermal aspects associated with the W21H canister, with much greater power density (kW/m3), would have resulted in a similar PCT as the lower power density canister, nor does it provide input, convergence, energy balances, and solution values to confirm performance of the two canister models.

b. There was limited information and validation associated with the “equivalence” between the (transient) two-dimensional and three-dimensional thermal models, recognizing that a steady-state comparison between 2D and 3D models would not explicitly demonstrate appropriate consideration of thermal mass effects. Using appropriate models are needed to ensure bounding temperatures on important to safety components during steady-state and transient operations.

c. There was limited information on the methodology for modeling fin performance, in the 3D and 2D representations including fin effectiveness and fin efficiency found in ta detailed sub-model. It is noted that details of the turbulence model and its parameters would impact results, especially considering the flow field around the fins is within an enclosed annular gap; these details were not provided. Likewise, there was no validation of the “non-thermal equilibrium porous media” to model the 2D fins, which is an important heat transfer component.

Considering the extent of the above-mentioned issues that need to be addressed, the applicant can provide the input and output files of the 2D and 3D FLUENT thermal models presented in SAR Chapter 5 to aid the review.

This information is needed to determine compliance with 10 CFR 72.234(a) and 72.236(f).

Westinghouse Response:

Response to this observation will be provided separately.

OBS 4-1

In SAR Sections 9.1.1 Design Criteria and 9.1.2.1 Confinement Vessel, reference is made to “Sections 3.1.1 and 3.4.2.” These sections do not seem to exist in the SAR.

Westinghouse Response:

Westinghouse acknowledges that these are incorrect cross-references. SAR Section 9.1.1 and Section 9.1.2.1 should reference Table 3.0-1 and Section 3.3.2.1, rather than non-existent Sections 3.1.1 and 3.4.2. This will be corrected in the next revision of the SAR. See markups on the following page.

9.1 CONFINEMENT DESIGN CHARACTERISTICS

9.1.1 Design Criteria

The design criteria of the confinement boundaries of the SENTRY W37 and W21H canisters are detailed in ~~Sections 3.1.1 and 3.4.2~~ of this SAR. Replace with: "Table 3.0-1 and Section 3.3.2.1

The design and fabrication code of the containment is ASME Code, Section III, Subsection WC (Reference 4) and ASME Section II, Part D (Reference 5) for material selection. The SENTRY canisters are leak-tight designed in accordance with ANSI N14.5 and NUREG-2215 is used for closure sealing requirements.

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April 2020

9.1.2.1 Confinement Vessel

The SENTRY canister cylindrical shell, the bottom end closure plate, the top end closure lid and closure ring, and the vent and drain ports with their associated covers comprise the confinement vessel. This constitutes a totally seal-welded vessel for the storage of SNF. They are designed, fabricated, and tested in accordance with the applicable requirements of ASME Section III, Subsection WC using ASME Section II, Part D austenitic stainless steel, as discussed in ~~Sections 3.1.1 and 3.4.2~~ of this SAR. The SENTRY canisters require no bolted closures or me Replace with: "Table 3.0-1 and Section 3.3.2.1

Section 3.1 of this SAR lists the design pressure and temperature for normal, off-normal and accident conditions for the SENTRY canister confinement. Additional information is provided in Section 1.2.

The canister cylindrical shell, the bottom end closure plate, and the closure lid as well as the shell seams and the shell-to-bottom closure plate welds are helium leak tested on the unloaded canisters.

The SENTRY canister cylindrical shell circumferential and longitudinal seam butt welds and bottom closure plate to shell weld are full penetration, radiographically[1] and liquid penetrant examined welds. These welds are hydrostatically pressure tested to 125% of canister design pressure, in accordance with the applicable requirements of ASME Section III, Subsection WC.

RSI 5-1

Provide code benchmarking analyses for the DORT code for shielding analyses or references that demonstrate the code has been adequately benchmarked for this application.

The applicant used the DORT code to calculate the adjoint function that is used in the shielding calculation. In accordance with the ANSI/ANS standard ANSI/ANS-6.1.2-2013[3], the analyses performed for numerical benchmarking shall be documented in sufficient detail to allow an experienced shielding analyst to duplicate the results. In order to assure the validity of the method of evaluation for shielding design, it is imperative to have the code benchmarked for the specific application. However, the applicant provides little information regarding how the code is benchmarked for this application. In addition, in the description of the DORT code, the Nuclear Energy Agency (NEA)[4] states: "[T]the Boltzmann transport equation is solved, using either the method of discrete ordinates or diffusion theory approximation. In the discrete ordinate's method, the primary mode of operation, balance equations are solved for the flow of particles moving in a set of discrete directions in each cell of a space mesh and in each group of a multigroup energy structure." It does not provide code benchmarking or references that demonstrate the code has been adequately benchmarked for this application. As such, the staff cannot determine if the code is appropriate for this application.

The staff needs this information to proceed with its review of the SENTRY dry storage system design to determine if the design meets the regulatory requirements of 10 CFR 72.236(d).

Westinghouse Response:

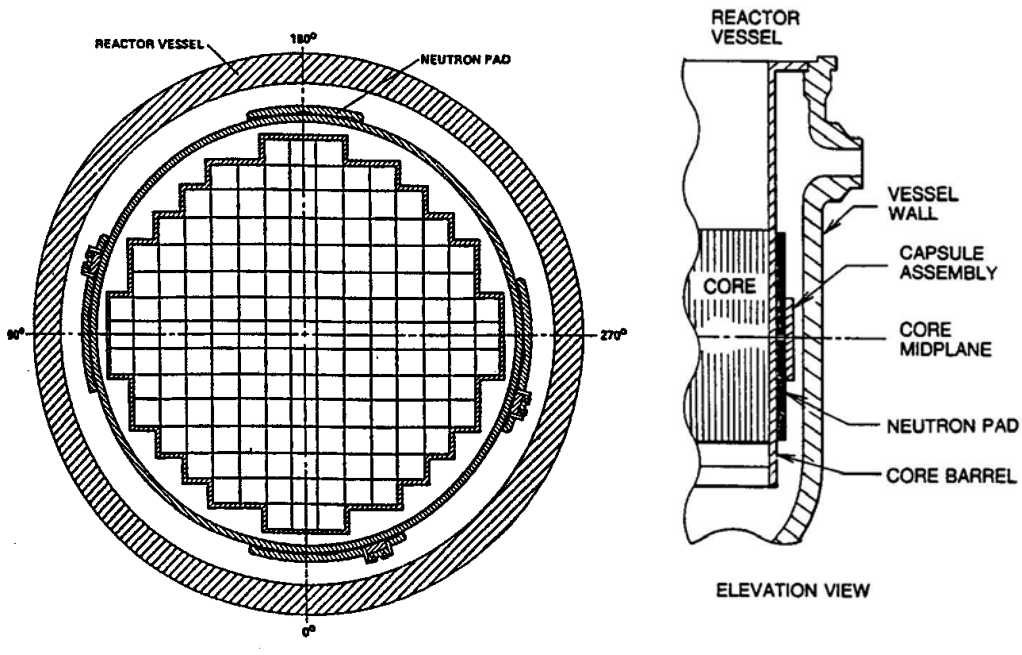
The DORT code and the DOORS package, and its forebear, the DOT code, have been used in this type of shielding application for many years. Such work has been performed by Oak Ridge National Laboratory and INL for the Department of Energy. It has also been performed by Westinghouse for previous shielding analyses supporting cask shielding projects ranging back to the FuelSolutions™ (then "Wesflex") cask system. Previously, the Westinghouse MC-10 cask was also designed making use of discrete ordinates codes, though making use of lesser order 1-Dimensional models. More recently, Westinghouse has made use of the DORT code and the same methodology to perform the individual plant ISFSI site boundary dose calculations.

The techniques and methodology for the cask shielding work are the same as those used for the analysis of reactor vessel fluence in accordance with US NRC Regulatory Guide 1.190. In both cases, the methods developed and verified by the Westinghouse radiation analysis group have been utilized, including cross-section libraries, source processing methods, and interpretation of results. This work is performed in support of ASTM E185, "Standard Practice for Design of Surveillance Programs for Light-Water Moderated Nuclear Power Reactor Vessels" as a means of demonstrating compliance with 10 CFR 50, Appendix H, "Reactor Vessel Material Surveillance Program Requirements." WCAP-17459-P, Rev 0, "Westinghouse Neutron Fluence Methodology – Description and Experience," provides further background associated with the methods.

[3] ANSI/ANS standard ANSI/ANS-6.1.2-2013, "Group-Averaged Neutron and Gamma-Ray Cross Sections for Radiation Protection and Shielding Calculations for Nuclear Power Plants," American Nuclear Society, 2013.
[4] htpp://www.oecd-nea.org/tools/abstract/detail/ccc-0543

The methods employed make use of the S_n transport calculations performed with DORT-2D techniques. In the case of reactor internals and reactor vessel analysis, these calculations compared to measured results extensively determined through the use of neutron dosimetry housed either in materials surveillance capsules or in ex-vessel neutron dosimetry programs. In accordance with Regulatory Guide 1.190, calculated vessel fluence calculations performed in this fashion must be shown to agree with measured values, including bias and uncertainties, to within 20%. This is an extremely high level of precision for calculations involving a fall-off in flux values of 5 to 7 orders of magnitude. The calculations and measurements obtained and used in the Westinghouse methodology routinely exceed this criterion substantially. Response functions evaluated include fast neutron fluence and flux, spectrum, thermal neutron fluence and flux, displacements per atom (dpa) and dpa per second.

As shown in the following figures, neutron dosimetry used to establish measurements is positioned in surveillance capsules positioned so that lead factors meet the ASTM E185 recommendation of falling between 1 and 3 and thus realistically representing materials and fluence conditions at the clad interface of the reactor vessel. The first figure shows a plan view of a 3-loop plant with neutron pad reactor internals; the figure shows the surveillance capsule locations. The second figure shows the capsule locations in elevation view.



The transport methodology described in Section 2.1 of WCAP-17459-P is identical to that described in the NRC approved versions of WCAP-16083-NP-A, “Benchmarking Testing of the FERRET Code for Least Squares Evaluation of Reactor Dosimetry,” and WCAP-14040, Revision 3, “Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves.” The first of these two reports includes the US NRC Safety Evaluation Report (SER) that approves the Westinghouse methodologies for transport calculations and dosimetry evaluations for application to LWR analysis and is available to the general public.

1. Validation of the transport methodology is based on the guidance provided in US Regulatory Guide 1.190. In particular, the validation consists of the following four stages:

2. Comparisons of calculations with benchmark measurements from the Pool Critical Assembly (PCA) pressure vessel simulator[5] at the Oak Ridge National Laboratory (ORNL).

3. Comparisons of calculations with surveillance capsule and reactor cavity measurements from the H. B. Robinson power reactor benchmark experiment[6].

4. An analytical sensitivity study addressing the uncertainty components resulting from important input parameters applicable to the plant specific transport calculations used in the exposure assessments.

5. Comparisons of calculations with operating plant measurement.

Transport and shielding benchmarks performed for this analytical approach include the Pool Critical Assembly and the H. B. Robinson Benchmark Experiment. Experimental and analytical results for both benchmarks, as well as adjusted and unadjusted comparisons are provided in WCAP-17459-P.

More details on these benchmarks and their uncertainties are provided in Section 2.2 of WCAP-17459-P. the section also provides further discussion on the methodology and the discussions and conclusions with the USNRC staff and its contractors. The information provided in the final issue of Regulatory Guide 1.190 indicates that either Monte Carlo or discrete ordinates approaches, if executed properly, can produce satisfactory results for problems with geometries of this sort. Vendor experience has also shown this to be the case.

Descriptions of the shielding work performed via the DORT code are provided in Sections 5.3, 5.4, and 5.5 of the base FuelSolutions FSAR. The work documented in that FSAR made use of essentially the same methodology used in the present SENTRY DSS work. In keeping with current practices in the Radiation Analysis group, a higher order of scattering representation (the P-order Legendre polynomial) was used for the SENTRY DSS work. In addition, a higher angular quadrature order was employed, improving the code's angular and directional representation of neutron and gamma ray transport. The SENTRY DSS work was performed with a P_5 expansion order and an S_{16} order of angular quadrature. This exceeds the minimum required by WCAP-17459-P, which calls for models with at least a P_3 Legendre expansion and at least an S_8 order of angular quadrature. Those requirements are in compliance with the requirements of Regulatory Guide 1.190. The approach taken in the SENTRY DSS work is also in keeping with additional guidance in WCAP-17459-P, which states that, "Westinghouse has found that this [P_3 and S_8] level of approximation is adequate for most applications, but for some analyses applicable to the reactor cavity external to the pressure vessel the use of higher order approximations (P_5 and S_{16}) provides improved results."

As with the work described in WCAP-17459-P and also the FuelSolutions work, the SENTRY DSS discrete ordinates calculations were performed as multi-group calculations making use of the BUGLE-96 cross-section library. The BUGLE-96 library provides a 67 group coupled neutron-gamma ray cross-section data set produced specifically for light water-fueled reactor applications. The generation of the multi-group cross-sections used in these analyses is derived following rigorous procedures that are described in both American Nuclear Society (ANS) standards as well as in US NRC Regulatory Guides. The general methodology for multi-group cross-section generation is described in ANS 6.2 "Neutron and Gamma-Ray Cross Sections for Nuclear Radiation Protection Calculations for Nuclear Power

[5] I. Remec and F. B. K. Kam, "Pool Critical Assembly Pressure Vessel Facility Benchmark," NUREG/CR-6454, July 1997.
[6] I. Remec and F. B. K. Kam, "H. B. Robinson Pressure Vessel Benchmark," NUREG/CR-6453 (ORNL/TM-13204), February 1998.

Plants" and guidance for application to Light Water Reactor (LWR) applications is provided in Regulatory Guide 1.190 "Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence". The procedures outlined in these documents were followed by Oak Ridge National Laboratory (ORNL) in the development of the BUGLE-96 multigroup libraries.

Section 2.1 of WCAP-17459-P provides further description of the transport methodology.

An additional reference of note for shielding analysis of spent fuel casks is EPRI-TR-104329[7].

Documents In Reading Room Folder for RSI 5-1

1. WCAP-17459-P, Rev. 0, "Westinghouse Neutron Fluence Methodology – Description and Experience."
2. WCAP-16083-NP-A, Rev. 1, "Benchmarking Testing of the FERRET Code for Least Squares Evaluation of Reactor Dosimetry," and WCAP-14040, Revision 3, "Methodology Used to Develop Cold Overpressure Mitigation System Setpoints and RCS Heatup and Cooldown Limit Curves."
3. WCAP-14040-A, Rev. 4, "Methodology Used to Develop Cold Overpressure Mitigating System Setpoints and RCS Heatup and Cooldown Limit Curves."
4. I. Remec and F. B. K. Kam, "Pool Critical Assembly Pressure Vessel Facility Benchmark," NUREG/CR-6454, July 1997.
5. I. Remec and F. B. K. Kam, "H. B. Robinson Pressure Vessel Benchmark," NUREG/CR-6453 (ORNL/TM-13204), February 1998.

[7] Broadhead, B.L., et al. Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments, EPRI TR-104329, May 1995.

OBS 5-1

Explain how the adjoint function calculated by the two-dimensional transport theory-based DORT code is used to calculate the dose rates around the cask.

The applicant used the DORT code to calculate the adjoint function that is used in the shielding calculation. However, there is no detailed information on how the discrete-ordinate adjoint function is used in the MCNP model to calculate the dose rates of the cask.

The staff needs this information to proceed with its review of the SENTRY dry storage system design to determine if the design meets the regulatory requirements of 10 CFR 72.236(d).

Westinghouse Response:

Section 6.3.3 of the SENTRY DSS SAR provides an overview of how the DORT code is used in the development of adjoint importances for use in generating spent fuel cooling tables pertaining to the beltline of the W180 storage cask housing a W37 or a W21H canister. The section further describes how the process is also used to develop similar adjoint importances for developing fuel cooling tables for the W110 transfer cask.

Section 6.5.3 of the SENTRY DSS SAR provides a description of the DORT adjoint methodology and explains how adjoint results can be converted into the Importances required for execution of the Westinghouse ADSORB computer code.

Thermal and radiological source term data for the SENTRY DSS have been established through the use of a previously-approved data set characterizing fuel over a wide range of burnup, enrichment, and decay time values. This data is used to construct cooling tables which provide loading rules in the form of a required cooling time required to load the assembly in a particular zone of a SENTRY canister. As a result of this approach, no further plant-specific calculations or demonstrations are required by the utility.

The dataset covers initial fuel enrichments from 1.5 through 5 w/o (5.5 w/o values also available) and burnups ranging from 15 to 60 GWD/MTU. The thermal and radiological source term dataset covers decay times from 1 year through 80 years.

Westinghouse makes use of adjoint methods to develop shielding results which are de-coupled from the source term associated with specific assemblies. As the industry's leading performer of Reactor Vessel Surveillance Programs, Westinghouse makes routine use of a combined forward/adjoint approach to radiation transport. This work is done in accordance with Regulatory Guide 1.190, as documented in WCAP-17459-P.

The adjoint shielding methodology allows for the explicit calculation of dose rates over a very large range of source term values with essentially no additional computation. As a result, Westinghouse can make use of its ADSORB computer code to develop results for the full set of enrichment, burnup, and decay times described above. Each of these approximately 2000 source term sets is then used to establish a dose rate and a total heat for each cask geometry. Each value is then compared to an established dose rate and thermal limit. Finally, the time at which both limits are first met is established, and this becomes the cooling table value for the enrichment and burnup pair. These values have been

compiled and presented in Chapter 17 of the SENTRY DSS SAR. Loading fuel in accordance with the fuel cooling table leads to dose rates that do not exceed those presented for the base cases.

The use of this methodology allows Westinghouse to establish the shortest time for which fuel can be placed into a particular canister zone. The methodology automatically produces and retains the dose and thermal totals for this case along with an indication of whether the solution was limited by the thermal or the dose rate limit. The details of this approach are presented in Section 6.2 of the SENTRY DSS SAR. Resulting cooling tables, background, and notes on their use are presented in Section 17.2 of the SAR.

Using Westinghouse's ADSORB cooling table methodology limits dose rates for all acceptable cask loading patterns, establishing that dose rates will not exceed those that were determined for the W37 and W21H canister base cases determined and documented through the MCNP shielding runs documented in Section 6.3 of the SENTRY DSS SAR. It is not required that a utility demonstrate that particular assemblies being selected for loading are bounded by SAR calculations and no adjustment of dose rate values is needed. Meeting the required cooling time listed for the enrichment and burnup is sufficient to show that dose rates will not exceed those provided in the SAR.

Several documents related to these descriptions have been placed in the reading room. These provide a more detailed understanding of how the ADSORB computer code was developed and has been used in the development of the cooling tables provided included in the Operating Controls and Limits presented in Chapter 17 of the SENTRY DSS SAR.

<u>Documents In Reading Room Folder for OBS 5-1</u>

1. ADSORB Code Development Document, DAR Number CMPC.1703.006, Rev 2—Included in Westinghouse Calculation Document CN-REA-13-21, Rev 0.
2. ADSORB Generic Source Document, DAR Number CMPC.1703.002, Rev. 1.
3. ADSORB User Manual, Included in Westinghouse Correspondence Document LTR-REA-13-28.
4. WCAP-17459-P, Rev. 0, "Westinghouse Neutron Fluence Methodology – Description and Experience."

OBS 5-2

Demonstrate that the 1980 version of the ORIGEN 2.1 computer code is adequate for calculating the source terms of fuel assemblies with burnup exceeding 45 GWd/MTU.

The applicant used the 1980 version of the ORIGEN 2.1 computer code to calculate the source terms. However, the staff's understanding is that the code is not capable for accurately calculating the source terms of fuel assemblies with burnup exceeding 45 GWd/MTU because the algorithm and cross section library the code uses are known to have deficiencies.

The staff needs this information to proceed with its review of the SENTRY dry storage system design to determine if the design meets the regulatory requirements of 10 CFR 72.236(d).

Westinghouse Response:

In support of the work submitted in the base FuelSolutions SAR, several items were developed. These included the ADSORB code and code assets described above. They also included an extensive library of radiological and thermal source term data. These were established over a wide range of burnup, enrichment, and decay time values. The dataset covers initial fuel enrichments from 1.5 through 5 w/o (5.5 w/o values also available) and burnups ranging from 15 to 60 GWD/MTU. The thermal and radiological source term dataset covers decay times from 1 year through 80 years.

FuelSolutions Design Analysis Report CMPC.1703.002, Rev. 1 (provided in the Reading Room) provided a comparison and description of the two key ORIGEN2.1 burnup libraries used in the production of the PWR data. These came up during discussion with reviewers at the time and were included to allow for an understanding of the significance of changes in the selection of burnup libraries. The ORIGEN2.1 PWR libraries used were pwrus.lib (or simply PWRUS) and pwrue.lib (or PWRUE). The first of these libraries was used for the cases running from a burnup of 15 up through 40 GWD/MTU. The second library was used for the remaining cases—those characterizing burnups in excess of 40 GWD/MTU (i.e., burnups from 42 through 60 GWD/MTU).

An investigation was performed of the effects of this library assumption, and these can be seen graphically in Figures 1 through 3 (a similar comparison was made for BWR fuel in Figures 4 through 6). It is possible to observe a step change between the two datasets in the case of the neutron source. Two inferences can be drawn from this discontinuity: 1) The discontinuity is not large, though it is noticeable; and 2) Where the higher burnup library (PWRUE) has been selected, the values are somewhat higher. Since the PWRUS was an appropriate library for modeling the lower burnup fuel, it was concluded that the relatively low burnup values of the upper set (approximately 42 through 48) are being somewhat overstated or at least accurately stated.

The PWRUE library was developed to represent 3 extended cycles and considered a 4.2 w/o initial enrichment and operation to a burnup of 50 GWD/MTU. Given the tools in existence at the time, this library was the most accurate one available for describing burnups up to this level and immediately beyond. Substantially beyond these burnup levels—perhaps at burnups of 56, 58, and 60 GWD/MTU, it can be presumed that the neutron source data would exhibit a similar trend to that seen in Figure 3 for the lower burnup data. In addition, it should be noted that an initial enrichment of only 4.2 w/o for these highest burnup values is quite unrealistic.

This neutron source discontinuity is not observed in the gamma source, and this is taken as substantial evidence that the gamma source strengths are largely unaffected by the selection of burnup library. As discussed in DAR CMPC.1703.002 (provided for review in the Reading Room), it seems clear that the substantially lesser discontinuity observed in the decay heat values reflects the neutron component of the decay heat.

Additionally, Westinghouse has had occasions to performed multiple comparisons between the various versions of the ORIGEN computer code (particularly ORIGEN-S and ORIGEN-ARP) and its libraries. Westinghouse is involved in the extensive development of ORIGEN-S and ORIGEN-ARP libraries for use in evaluating decay heat and radiation shielding issues. These are frequently cross-compared and have been compared with ORIGEN2 results. The values are generally found to be quite consistent. This is particularly the case with integral parameters such as total source strength or decay heat.

Westinghouse has recently been performing analyses and comparisons in support of several potential burnup limit extensions—the first an incremental burnup limit extension and subsequently an additional, potentially more substantial increase. In the course of this work, cross-section libraries have been generated to enable evaluations of radiological and thermal impacts. This work includes a parametric review of fuel parameters ranging from fuel and clad temperatures, boron concentrations, moderator temperatures and densities, and burnable poison configurations. A current working conclusion is that over a wide range of physical changes evaluated, change in decay heat is negligible, except in cases which consider variations in specific power. Again, these conclusions lend credence to the observation that differences in code libraries are not of strong importance for decay heat values.

The work cited above also reviewed the impact of the fuel parameters on a number of key nuclides. These nuclides were selected for various reasons, including their significance as direct-dose importance at intermediate and long times and also for decay heat and neutron dose contributions. Through the evaluation of small variations in temperature, specific power, physical dimensions, and poison characteristics, these nuclides were not found not to be significantly impacted by differences in cross-section libraries. Only specific power changes led to significant changes in dose contributors. Moderator density was associated with some differences in actinide values as well as the poison samarium, but these are not of importance to shielding sources or decay heats.

An additional recent Westinghouse calculation, CN-REA-20-25, makes extensive comparisons between ORIGEN-ARP and ORIGEN2 resoults. As stated in this calculation, “Differences are to be expected between the ORIGEN2 and ORIGEN-ARP calculations. The calculation methods are fundamentally the same – both solve a mass balance problem for nuclide production and loss via matrix exponetial solution. But the reaction cross-sections and fission yields will differ between the codes.” This further supports the evaluation of code version differences primarily through considerations of library changes.

<u>Documents In Reading Room Folder for OBS 5-2</u>

1. ADSORB Code Development Document, DAR Number CMPC.1703.006, Rev 2—Included in Westinghouse Calculation Document CN-REA-13-21, Rev 0.
2. ADSORB Generic Source Document, DAR Number CMPC.1703.002, Rev. 1.
3. ADSORB User Manual, Included in Westinghouse Correspondence Document LTR-REA-13-28.

OBS 5-3

Provide the referenced document for the ADSORB computer code.

The applicant developed a computer code, ADSORB, that is used to determine the allowable contents. The applicant referenced a document titled “CMPC.1703.006, Development and Validation of ADSORB Computer Code, Revision 2.” The applicant states that this computer code is used to determine the allowable contents based on combining the shielding and thermal acceptance criteria, the adjoint shielding and the generic decay library and the output of the ADSORB code is in the fuel cooling tables. The applicant, however, did not provide this document. The staff cannot assess the validity of the computer code without the referenced document. An annotated sample input will be helpful for the staff to understand the code since this code is not in the Radiation Safety Information Computational Center (RSICC) collection and hence the staff does not have access to this code.

The staff needs this information to proceed with its review of the SENTRY dry storage system design to determine if the design meets the regulatory requirements of 10 CFR 72.236(d).

Westinghouse Response:

During the work documented in the original FuelSolutions SAR, work was performed to allow for the automated generation of extensive fuel cooling tables. This work resulted in the development of the Westinghouse ADSORB computer code and its associated source term libraries. The code has subsequently been migrated to the current Westinghouse Linux computing environment but is otherwise unchanged.

A number of documents have now been uploaded to the SENTRY DSS NRC reading room (list below). These document the development and function of the ADSORB code, including all functional requirements as well as theoretical and methodological considerations relating to the code. They also present the basis and the techniques associated with the modeling of a wide range of fuel cycles in order to obtain source term data suitable for characterizing the thermal and shielding source strength of fuel spanning burnups from 15 GWD to 60 GWD, initial fuel enrichments ranging from 1.5 to 5.5 w/o, and decay times ranging from 1 year to 80 years after discharge. Documentation has been included in the reading room showing the present configuration status of ADSORB on Westinghouse computing platforms. Finally, an annotated ADSORB input file has been uploaded to the Reading Room for staff understanding of the code.

<u>Documents In Reading Room Folder for OBS 5-3</u>

1. ADSORB Code Development Document, DAR Number CMPC.1703.006, Rev 2—Included in Westinghouse Calculation Document CN-REA-13-21, Rev 0.
2. ADSORB Generic Source Document, DAR Number CMPC.1703.002, Rev. 1.
3. ADSORB User Manual, Included in Westinghouse Correspondence Document LTR-REA-13-28.
4. ADSORB code migration document, Software Change Specification and Validation for ADSORB Version 1.2.1., LMD-SST-13-002 Rev 0.
5. Commented ADSORB input file.

OBS 6-1

Demonstrate how the GBC-32 Canister is sufficiently similar/applicable to the SENTRY W37 Canister for burnup credit analyses.

The applicant took burnup credit in its criticality safety analyses for the SENTRY W37 Canister and states that it used the guidance provided in NUREG/CR-7108 and NUREG/CR-7109. However, the staff notes that the guidance in ISG-8, Rev. 3 explicitly states that the users of the guidance must demonstrate that the neutronic characteristics of the W37 cask are sufficiently similar to that of the GBC-32 cask in order to be able to use the recommendations provided in Revision 3 of Interim Staff Guidance No. 8 (ISG-8). Specifically, ISG-8, Rev. 3 states: “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.”

Also, the staff notes that the applicant took burnup credit only for some of the fuel assemblies as shown is Figure 7.2-2 of the SAR. In addressing the similarity requirement of recommendation of the ISG-8, Rev. 3, the applicant needs to consider this unique feature of the W37 canister design.

The staff needs this information to proceed with its review of the SENTRY dry storage system design to determine if the design meets the regulatory requirements of 10 CFR 72.124 and 72.236(c).

Westinghouse Response:

Westinghouse has performed an assesment of the similarity between the GBC-32 system and the W37 canister. Performed with the TSUNAMI code, the C_k 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.

OBS 6-2

Explain how the grid spacers are treated in the SCALE models for criticality safety analysis and why it is acceptable.

SAR Table 3-1 indicates that the design basis minimum soluble boron concentration is 2600 PPM. The staff's experience is that the grid spacers and non-fuel hardware (if loaded), should be explicitly represented in the criticality safety analysis models when the required soluble boron concentration is at an elevated level because the displacement of the highly borated water could cause the reactivity to increase.

The staff needs this information to proceed with its review of the SENTRY dry storage system design to determine if the design meets the regulatory requirements of 10 CFR 72.124 and 72.236(c).

Westinghouse Response:

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. This document has been provided for review in the Reading Room.

The assessments in this WCAP concluded that "the results of the criticality calculations with and without grids considered are conservative or statistically indistinguishable from one another. There was no statistically significant case examined where the reactivity with grids was greater than the reactivity without grids."

For these reasons, it can be seen that neglecting the grids and sleeves is an appropriate and conservative approach.

<u>Documents In Reading Room Folder for OBS 6-2</u>

1. WCAP-17483, Rev. 0, "Westinghouse Methodology for Spent Fuel Pool and New Fuel Rack Criticality Safety Analysis."

OBS 7-1

In SAR Section 8.6.1, clarify the penetrant testing acceptance criteria for indications in the canister closure lid-to-shell weld.

SAR Section 8.6.1 describes a calculation of the “allowable weld flaw depth” for the closure lid-to-shell welds, which is based on ASME Boiler and Pressure Vessel (BPV) Code Section XI Nonmandatory Appendix C, “Evaluations of Flaws in Piping.”

It is unclear to the staff whether that calculated flaw depth will be used to define the minimum intervals for surface examinations for the multi-pass closure weld (per NUREG-2215, Section 8.5.3.2.1), or if it may also be used to define the acceptance criteria for the liquid penetrant examination of the closure welds. The staff notes that SAR Table 12.1-3 states that the examinations of welds are to be performed in accordance with ASME BPV Code Section III, Division 3, Subsections WC and WD. Paragraph WC-5352 states that any crack or linear indication is not acceptable when examined with the liquid penetrant method.

This information is needed to evaluate compliance with 10 CFR 72.236(d), (e), and (j).

Westinghouse Response:

The canister lid to shell weld penetrant testing is performed in accordance with ASME B&PV Code, Section III, Division 3, Subsections WC and WD. The acceptance criteria is defined, as mentioned, in Paragraph WC-5352 (any crack or linear indication is not acceptable).

For the W37 canister, as indicated in drawing WDD-DW-00116-GEN (refer to Section 1.5.3 of the SAR), the weld is 7/8 inches thick and note 9 states that “Weld shall consist of a minimum of three welding pass. PT shall be done at minimum on root, mid and final pass…”.

The same is indicated for the W21H canister in drawing WDD-DW-00132-GEN (refer to Section 1.5.5 of the SAR with the same weld thickness and same note indicated).

Thus, the maximum allowable flaw depth has been calculated in order to ensure that if only the root, middle and final pass are PT tested, this flaw would be detected (considering that no crack or linear indication are permitted).

The maximum acceptable flaw depth is 0.66 inches (as identified in Section 8.6.1 of the SAR). This demonstrates that with a minimum of three equidistant PT (root, middle and final pass) on a 7/8 thick weld, this flaw depth would be detected. This appraoch is consistent with the guidance of NUREG-2215 Section 8.5.3.2.1.

OBS 7-2

Provide the technical basis for the maximum allowable concrete overpack temperatures for off-normal and accident conditions that appear to exceed the cited American Concrete Institute (ACI) standard.

Section E-4 of ACI 349-13, "Code Requirements for Nuclear Safety-Related Concrete Structures," states that, for accidents or other short-term periods, the temperature of the concrete surface shall not exceed 350°F. A local area from a steam or water jet due to pipe failure is allowed to reach 650°F. ACI 349-13 allows higher temperatures than these if tests are provided that evaluate the potential loss of strength.

SAR Table 5.2-2 references the ACI 349-13 limits, but the SAR appears to apply these limits in manner that is not consistent with the code. Whereas ACI 349-13 applies the 350°F limit to the concrete surface, the SAR applies it to the bulk average concrete temperature. Also, ACI 349-13 applies the 650°F limit to a local area (e.g., location of a steam leak); it is not clear how the SAR defines a "local" area allowed to reach 650°F.

As a result, the staff requests the following:

1. A technical basis that demonstrates that a bulk average concrete temperature exposure of 350°F (where the surface may reach as high as 558°F per SAR Figure 5.4-14) will not decrease strength and lead to moisture loss to an extent that could prevent the overpack from fulfilling its structural and shielding functions.

2. Clarification of what defines the extent a "local" area (i.e., area and depth affected), and a demonstration that the exposure of the concrete overpack to conditions where a local area reaches 650°F will not prevent the overpack from fulfilling its structural and shielding functions.

This information is needed to determine compliance with the requirements of 10 CFR 72.236(b) and (d).

Westinghouse Response:

1. Paragraph E.4.3 of ACI 349-13 states that tests are provided to evaluate the reduction in strength and this reduction is applied to design allowable if temperatures are higher than those established in E.4.1 for Normal Conditions (long term) and in E.4.2 for Off-Normal or Accident Conditions (short term). Long-term limits are accomplished by considering the alternative limit indicated in NUREG-2215 (Section 8.5.2, point 2), 300°F in general or local areas. During short term events temperature limits are accomplished by limiting the general temperature to 350°F and the local temperatures to 650°F (from ACI 349-13 E.4.2). The approach used to establish concrete temperature limits during short-term events has been already used in other applications accepted by the NRC (docket number 72-1031, where it is stated that 350°F limit applies to concrete bulk temperature).

   According to conclusions of NUREG/CR-6900 "The Effect of Elevated Temperature on Concrete Materials and Structures – A Literature Review", although an initial small loss of strength is shown for concrete in the temperature range of 20°C to 200°C (68°F to 392°F), a regain of strength is often observed between 120°C and 300°C (248°F to 572°F) and beyond 350°C (662°F) there can

be a rapid decrease in strength. So, small strength loss is shown below the short-term limit of 650°F.

In addition, structural calculations have been performed taking into account a conservative compressive strength of concrete at 28-days. According to NUREG-2215, Section 8.5.8.2, the strength and elasticity of concrete increase as it ages, showing typical increases up to 67 percent in 10 years relative to the recorded 28-day strength.

2. This approach considers the definition of local area in ACI 349-13, “region of structure exposed to elevated temperature that may affect concrete performance and stress distribution in the immediate area, without changing the overall component or member behavior”. NUREG/CR-7031 “A Compilation of Elevated Temperature Concrete Material Property Data and Information for Use in Assessments of Nuclear Power Plant Reinforced Concrete Structures” includes the following figure:

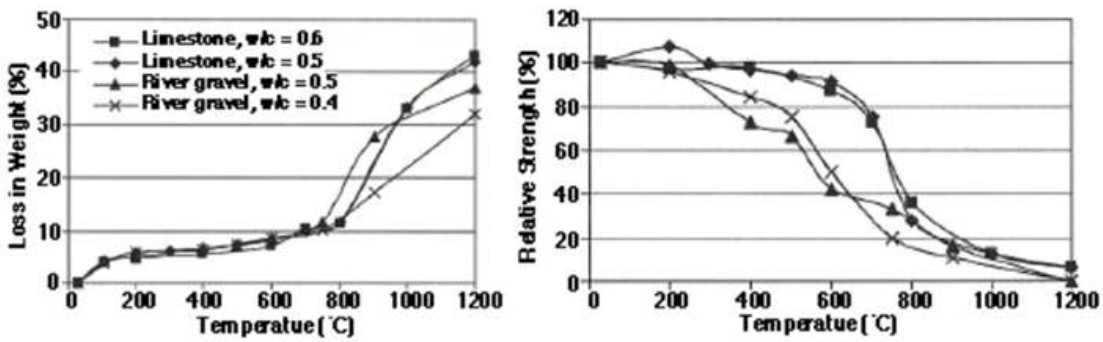


Figure 2.1 Weight change and residual compressive strength of siliceous gravel and limestone concretes as a function of temperature.

Source: O. Arioz, “Effects of Elevated temperatures on Properties of Concrete,” *Fire Safety Journal* **42**, pp. 516-522, 2007.

The figures above show how weight and strength do not substantially change in ordinary concretes exposed to temperatures varying between the long term limit (176 °C) and the short term limit considered for local areas (650 °F, i.e. 343 °C). Due to those minimum variations, shielding and structural behavior is minimally affected in the zones reaching these local temperatures that, on the other hand, are less than half the thickness of the concrete shell. The other half of the thickness is under the short-term bulk temperature limit and, therefore, it can be concluded that this exposure will not prevent the overpack from fulfilling its structural and shielding functions and the definition of local area is properly applied.

Finally, according to conclusions of NUREG/CR-6900, Portland cement concretes are the most suitable materials for structural applications involving service temperatures in the range of ambient to 300°C (572°F) or 400°C (752°F), without many cycles of large magnitude. For these applications, heat-resistant aggregates (basalt, limestone, etc.) should be used. It is also stated that for limited periods, the Portland cement concretes could probably tolerate temperatures to 600°C (1112°F). Heat-resistant aggregates with low coefficient of thermal expansion have been considered according to SAR Chapter 8 and NUREG-2215, Section 8.5.8.2.

OBS 7-3

Demonstrate that stresses caused by differential thermal expansion cannot lead to overloading of ITS bolting.

SAR Section 8.7 states that no bolts are used in the confinement boundary for the SENTRY dry cask storage system. It does not address other ITS bolting (e.g., transfer cask and storage cask lid bolts) that may experience loads due to different coefficients of thermal expansion of bolting materials and the materials being joined.

This information is needed to determine compliance with the requirements of 10 CFR 72.236(b) and (d).

Westinghouse Response:

There are two ITS bolts in the W110 transfer cask: items 7 and 9 as can be seen in drawing number WDD-DW-00124-GEN (refer to Section 1.5 of the SAR).

The W110 transfer cask is designed in accordance with the ASME Code, Section III, Division 1, Subsection NF (Class 3 component). The structural analysis of both bolts is discussed in Chapter 4 of the SAR, with descreptions of theall the external loads the bolts are supporting. According to NF-3121.11, evaluation of thermal stresses is not required by this Subsection, so no further analysis is needed to demonstrate compliance with ASME Code, Section III, Division 1, Subsection NF.

Regarding the W180 storage cask lid bolts, the effect caused by differential thermal expansion is absorbed by the gap between the bolts and lid holes. [

]$^{(a,c)}$

Considering a conservative scenario where the top of the cask temperature increases to 300°F while the lid remains at ambient temperature. With a coefficient of thermal expansion of α=6.9E-6 in/in°F for carbon steel SA-516 Gr.70, [

]$^{(a,c)}$. With this margin, the differential thermal expansion is not expected to have any effect on the W180 storage cask bolts.

OBS 8-1

Provide the time required for loading, welding, and transferring the canisters to storage pad and appropriate justification for these time estimates (e.g., procedures).

The applicant provided some general descriptions for loading, welding, and transferring the canisters to storage pad. However, not enough information is available for the staff to assess the adequacy of the radiation protection plan for workers as provided in SAR Chapter 10.

The staff needs this information to proceed with its review to determine if the SENTRY dry storage system design meets the regulatory requirement of 10 CFR 72.236(d).

Westinghouse Response:

As stated in section 11.1 of the SENTRY DSS SAR: "Generic procedures are provided herein to describe how the associated operations are to be performed. These procedures are not intended to be all-inclusive, but rather are provided as a generic guide **for the preparation of more detailed site-specific procedures by a licensee**…The generic operating procedures for the SENTRY DSS provided herein have been developed to assure that operations required for canister loading, unloading, and transfer are performed safely with minimal personnel exposure, and maximum operational efficiency. **In preparing site-specific procedures, the licensee** shall have the discretion to develop acceptable alternate means to accomplish the same operational objective provided they conform to the safety evaluation documented in this SAR." (emphasis added)

Based on the explicit directions for the licensees (users) to develop detailed procedures, based in the generic procedures provided in the SAR, Westinghouse believes that the SAR as written satisfies the request to provide instructions for the users to develop detailed procedures. Further, this approach is consistent with what was provided in FuelSolutions SAR (CoC 72-1026) and follows NUREG 2215 guidance for a general CoC application (or amendment application). The operational steps in Chapter 10 (Section 10.2) of SENTRY DSS SAR correspond to the generic procedures provided in Chapter 11 and the associated times are provided in Chapter 10 for each operational step, along with the associated dose rates for the operational steps.

The basis for estimates for the time required for the various operational steps described in Chapter 10 is reviews by Westinghouse experts with operational experience in the performance of similar tasks during fuel loading operations.

OBS 9-1

Revise SAR 15.1, QUALITY ASSURANCE PROGRAM FOR SENTRY DSS, Reference 3.0, to reflect currently approved WEC QAP, Revision 8.

SAR 15.1 QUALITY ASSURANCE PROGRAM FOR SENTRY DSS, states that the requirements of Subpart G "Quality Assurance" of 10 CFR 72 are met by the Westinghouse Quality Management System (QMS), Reference 3. Reference 3 commits to Westinghouse Electric Company QMS, Revision 7, dated August 2013. The NRC staff has subsequently reviewed Westinghouse's QMS, Revision 8.0 and has concluded that there is reasonable assurance that Westinghouse's QMS will continue to meet the requirements of Appendix B to 10 CFR Part 50 and 10 CFR Part 71 Subpart H. Subsequently, the NRC staff found Westinghouse's proposed changes in the QMS, Revision 8.0, to be acceptable (ADAMS Accession No. ML20132A321).

Westinghouse Response:

Westinghouse agrees that the next revision of the SAR will include updated text in Chapter 15 to reference the most recent revision of the Westinghouse Electric Company QMS (Revision 8.0). See markup provided in the following page:

15.2 REFERENCES

1. Title 10, U.S. Code of Federal Regulations, Part 72 (10 CFR 72), *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor-Related Greater than Class C Waste,* 2019.

2. NUREG-2215, *Standard Review Plan for Spent Fuel Dry Storage Systems and Facilities*, U.S. Nuclear Regulatory Commission, November 2017.

3. Westinghouse Electric Company Quality Management System (QMS), ~~Revision 7, August 2013.~~ Replace with: Revision 8, May 2020.

4. ML14336A487, *Final Safety Evaluation for Westinghouse Electric Company (Westinghouse) Topical Report Quality Management System, Revision 7 (TAC No. MF2833)*, U.S. Nuclear Regulatory Commission, December 2014.

5. Title 10, U.S. Code of Federal Regulations, Part 50 (10 CFR 50) Appendix B, *Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants*, 2007.

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