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November 20th, 2020

Subject: **Updated Responses to Request for Supplemental Information 5-1 for the Application for the SENTRY™ Dry Storage Cask System (Certificate of Compliance [CoC] No. 72-1026; Proposed Amendment 5)**

- References
- (1) "REQUEST FOR SUPPLEMENTAL INFORMATION – WESTINGHOUSE APPLICATION FOR THE SENTRY™ DRY STORAGE CASK SYSTEM, CERTIFICATE OF COMPLIANCE NO. 1026 (DOCKET NO. 72-1026, CAC NO. 001028, EPID: L-2020-LLA-015", ADAMS Accession No. ML20223A036.
 - (2) "Responses to Requests for Supplemental Information for the Application for the SENTRY™ Dry Storage Cask System (Certificate of Compliance [CoC] No. 72-1026; Proposed Amendment 5)," ADAMS Accession No. ML20276A295.

Requests for Supplemental Information (RSIs) related to review of the Application for the SENTRY™ Dry Storage System were provided to Westinghouse in Reference 1 and initial responses were provided to the NRC for most of the RSIs in Reference 2.

This letter provides Westinghouse's updated response to the RSI 5-1. The updated response provided in this letter provides additional supplemental information responsive to the Staff's request in this area.

A handwritten signature in blue ink, appearing to read "Robt Quinn", is written in a cursive style.

Robert D. Quinn, Director
Nuclear Material Management

cc: John McKirgan (NRC)
Niskha Devaser (NRC)

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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-6.1.2-2013¹, 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)² states: “[T]he 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).

Updated Westinghouse Response:

Methodology Road Map

Dose rate profiles characterizing the SENTRY™ canisters and casks are established for the base cases described in the SAR. The demonstrations are performed for two base cases. For the W37 canister, the base case uses a source term pertaining to 4.5 w/o fuel with an assembly average discharge burnup of 48 GWD/MTU and a decay time of 5.63 years. For the W21 canister, the base case makes use of a source term pertaining to 4.5 w/o fuel with an average discharge burnup of 54 GWD/MTU and a decay time of 1.5 years. All of this work is performed by Monte Carlo analysis making use of the MCNP code.

In order to obtain cooling times associated with fuel of different enrichment and burnup values, cooling tables are then constructed which are presented in the form of the minimum cooling time required for each enrichment and burnup. By ensuring that fuel is not loaded prior to these cooling times, we demonstrate that the dose rates presented in the MCNP tabulations and figures will not be exceeded.

¹ 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.

² <http://www.oecd-nea.org/tools/abstract/detail/ccc-0543>

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Figure 5-1 diagrams the process used for characterizing the dose rates for all casks and canisters. These analyses are performed and reported based on the base case state points described above. This work is performed by means of the MCNP code.

Figure 5-2 shows the process by which importance values are developed for each of the cask and canister combinations (e.g., W37 canister in a W110 transfer cask, W21 canister in a W180 storage cask, etc...). The importances are generated from adjoint results obtained via the DORT discrete ordinates radiation transport code. In the course of developing these importance values, Westinghouse compared and demonstrated the full consistency of the adjoint results with results obtained by running the same models in forward mode.

Figure 5-3 outlines the process of developing the cooling tables themselves. In this process, several inputs are taken from the results developed by MCNP and DORT, as described above. The importances developed through the DORT code are used to characterize the result at the index location on each cask: the radial outer surface at the canister beltline. Technically, these importances are not themselves adjoint values, but rather responses that represent the dose contribution made by a single particle (neutron or photon) situated in the canister fuel region of interest. As described in the ADJOINT user manual—available in the reading room—these values are adjoint responses provided for each energy group and particle type. An importance or adjoint response is expressed in mrem/hr per particle/sec. These values thus describe the fractional contribution of particles from that energy group to the total dose rate at the index location. The value of such an approach is that it becomes possible to rigorously perform the thousands of shielding calculations required to establish cooling tables of this type.

In addition to the dose-related information discussed here, there is also another key input required to generate the cooling tables. That is the allowable heat in the canister or region of interest. The ADSORB code determines the minimum time required to meet both the specified dose rate limit and this input decay heat value. The code then conservatively constructs cooling tables making use of the longest time of the two values. For the purposes of this RSI, further discussion of this decay heat aspect will be neglected.

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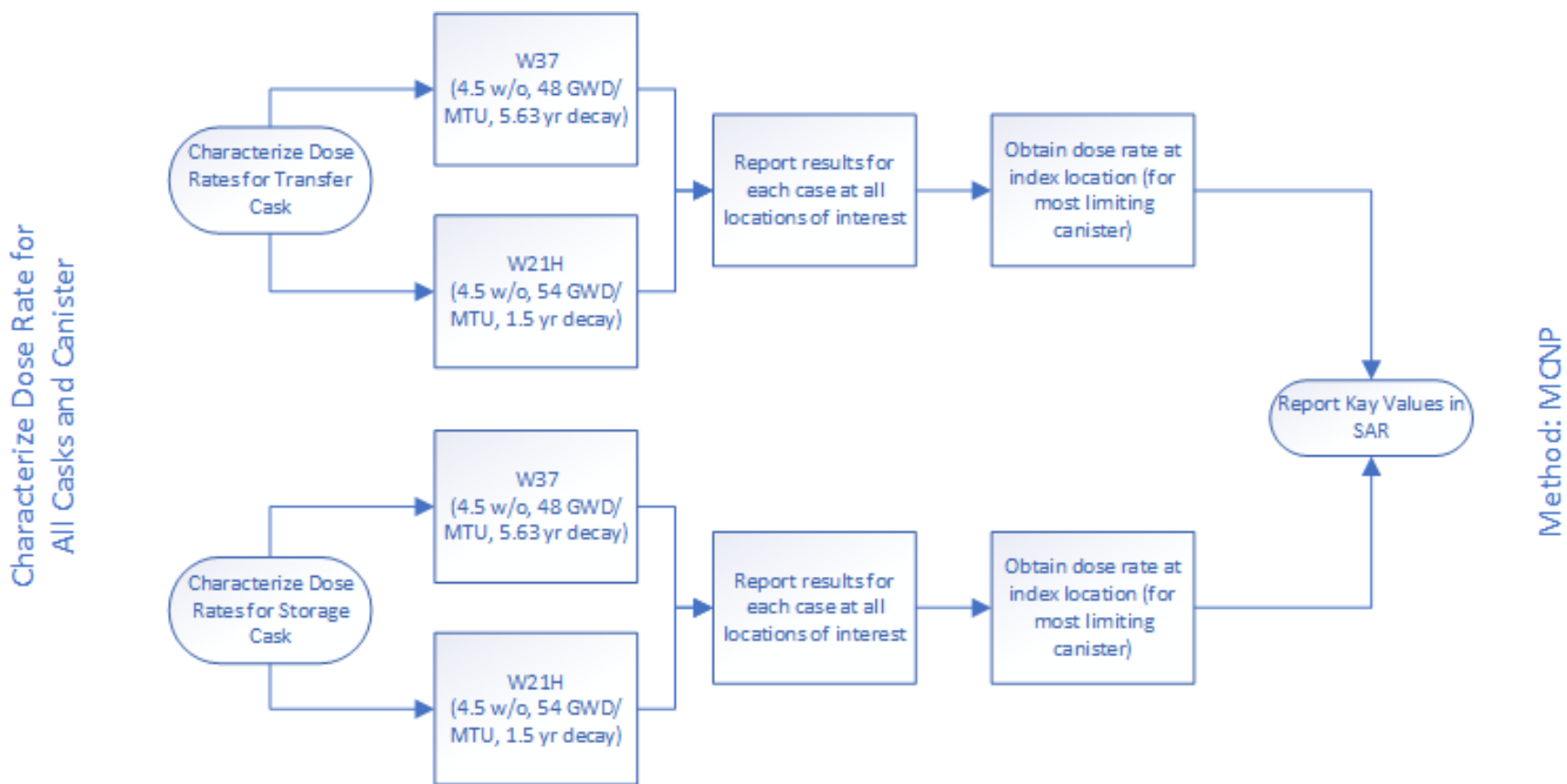


Figure 5- 1 - Dose Rate Characterization

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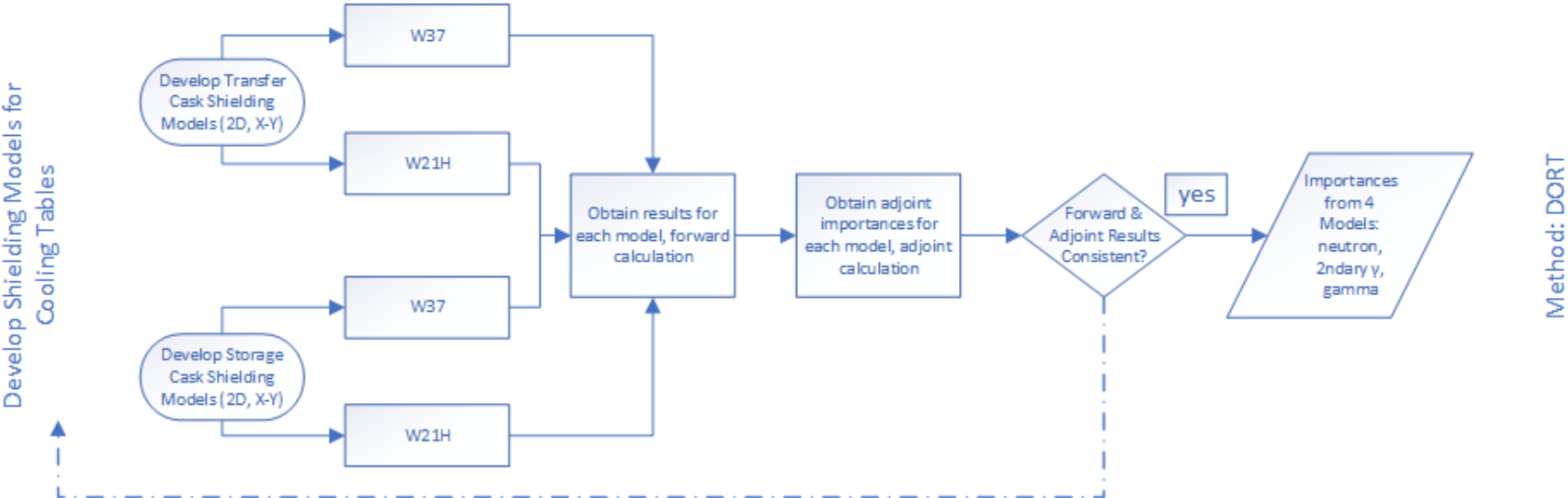


Figure 5- 2 – Adjoint Shielding Models for Cooling Tables

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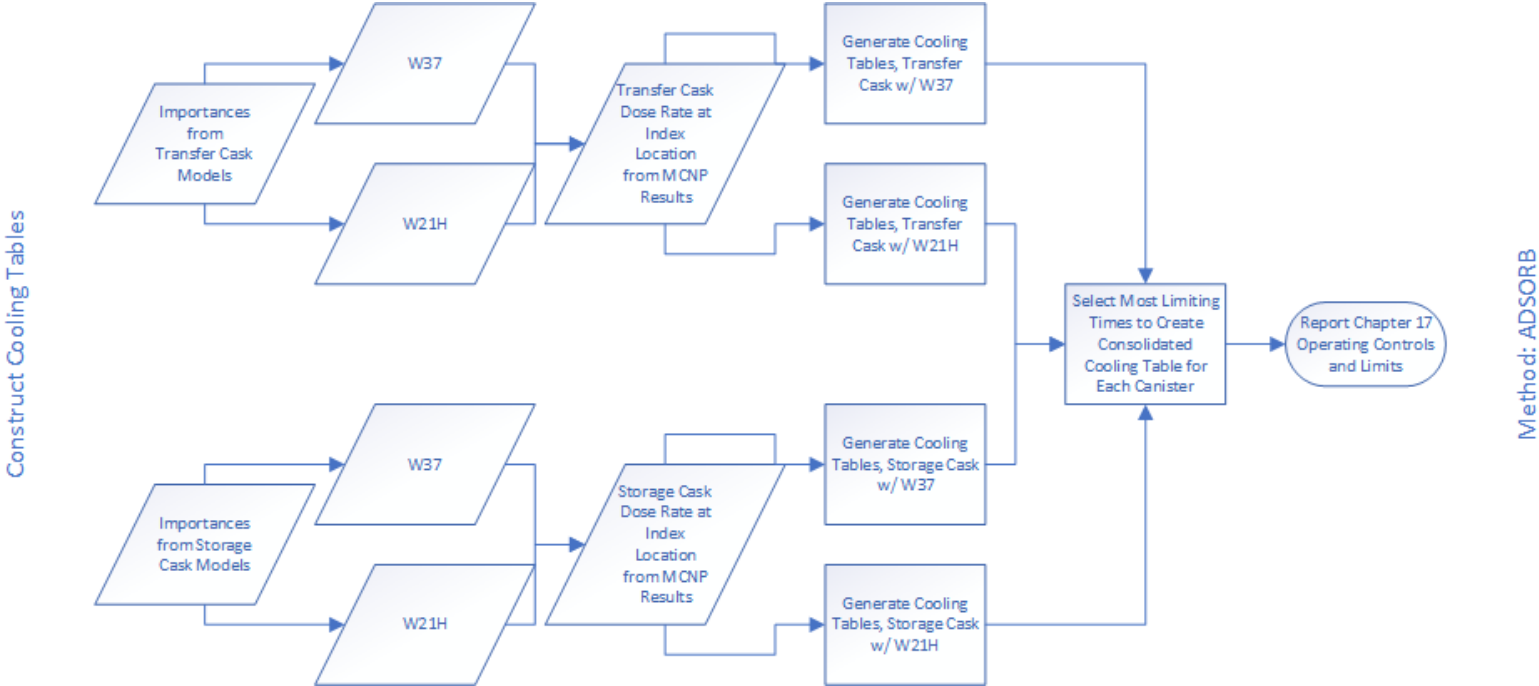


Figure 5- 3 – Construction of Cooling Tables

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FuelSolutions™ Cask System Methodology and Approval

This method of developing cooling tables has previously been used and approved, for one example in the FuelSolutions™ Storage System, SER Docket No. 72-1026, Certificate of Compliance No. 1026 (Adams Accession No. ML003762509). The following statements from this example pertain to the work and the methodology.

- *[Section 5.4] The shielding analyses are presented in Section 5.4 of WSNF-200. The applicant used DORT and MCNP with the BUGLE-93 cross-section library to determine the dose rates from the storage and transfer casks. The BUGLE-93 cross-section library has 47 neutron groups and 20 photon groups. The applicant uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose conversion factors to calculate dose rates in the shielding analysis.*
- *The applicant performed adjoint calculations using the DORT computer code. The adjoint calculations were used to determine the neutron and gamma importance functions (units of mrem/hr/particle/sec-cm). Multiplying the importance functions by a neutron and gamma source term per unit length yields dose rates on the surface of the cask. Using the importance functions, the applicant determined the minimum cooling time required to meet both the decay heat limit and the TS 5.3.5 maximum dose rate limit of 50 mrem/hr on the side of the concrete cask.*
- *The bulk shielding calculations are also performed with the DORT computer code. WSNF-200 presents calculations for normal condition dose rates for both bounding and typical source terms.*
- *The staff agreed with the applicants conclusion that locations limited by decay heat had lower dose rates than locations limited by dose rate.*
- *Confirmatory shielding calculations for the FuelSolutions™ Storage System were made with the SAS4 module in the SCALE 4.4 system. The staff homogenized each fuel assembly, but explicitly modeled the spacer plates, both inside and outside the fuel region. A comparison between the applicant's results and the staff's confirmatory calculations showed a variation in the results which is expected when two different codes are used for shielding calculations. The surface dose rate calculated by the staff for the bounding source term is 29.4 mrem/hr on the side of the cask. The staff's dose rate for the bounding source term is less than the applicants dose rate of 32.7 mrem/hr.*
- *The applicant's dose rates 1 meter from the cask are in good agreement with the confirmatory calculations. The staff's dose rate 1 meter from the surface of the storage cask is 15.7 mrem/hr. The staff's dose rate is lower than the applicant's value of 17.7 mrem/hr.*

With respect to the above statements, one comment should be made. While the FuelSolutions bulk shielding calculations were performed with the DORT computer code as stated above, Westinghouse has not done that with the SENTRY submittal. Instead, the bulk shielding analyses in the case of the SENTRY cask have all been performed via the MCNP code. Thus the SENTRY application rests far less strongly on the use of the DORT code.

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Examples of Use of Discrete Ordinates for Biological Shielding of Spent Fuel Cask Systems

The seven references presented below show that the use of discrete ordinates codes in general and of DORT or its predecessor, DOT (“DOT3W” at Westinghouse), has been a common shielding tool in the industry. Prior to around the year 2000, it represented the state of the art, and neither more accurate nor more refined tools were available to model complex shielding configurations. As is shown below, its use both for spent fuel casks and for biological doses has been a standard best practice. All listed references are publically available.

Ueki affirms, in Reference 1, that Sn methods have been the main means of solving shielding problems of this type and complexity in Japan prior to 2004. More specifically, the paper demonstrates that results from MCNP are found consistently to predict the more conservative DOT [discrete ordinates] results.

The following statements are germane.

- *As predicated, compared with the DOT 3.5 calculations, the total dose-equivalent rates with the actual configurations are reduced to approximately 30% at 1m from the upper side surface and 85% at 1m from the lower side surface, respectively. Accordingly, the employment of detailed models for the Monte Carlo calculations is essential to accomplish more reasonable shielding design of a spent fuel transport cask and an interim storage cask.*
- *Up to now, the two-dimensional discrete ordinates Sn method has been employed for radiation shielding analysis of spent fuel transport casks in Japan, and as an advantage contour maps of dose-equivalent rates around such casks can be described using the two-dimensional discrete ordinates Sn code DOT 3.5 (W. A. Rhoades, et al., 1973, and ccc-276, 1977) with DLC-23/Cask Library (ORNL-RSIC, 1973). However, the DOT 3.5 code has noticeable restrictions for the three-dimensional modeling of the shielding system in a cask, and so the canister containing the fuel basket and the cooling fins are homogenized, the supplemental shields located outside the lower nozzles are simplified and attached outside the fuel basket, and the trunnions are ignored in the DOT 3.5 calculations. As a result, the calculated dose equivalent rates tend to be overestimated considerably as compared with the measured data (Y. Momma, et al., 2000).*
- *In one case, the canister containing the fuel basket with 12 spent fuel assemblies was modeled three-dimensionally with the MCNP 4B code, and in the other case, two-dimensional homogenized model was used as employed in the Sn code DOT 3.5. By adopting the actual configuration with the Monte Carlo code, the gamma ray and neutron dose-equivalent rates were reduced to 67 % and 80 % respectively, on the TN-12A cask surface, as compared with the two-dimensional model of the Sn code. (K. Ueki, et al., 2000).*

In addition to their common use, Ueki indicates that discrete ordinates methods have led to conservative results. It should be noted that the use of MCNP described in this reference has been driven by a wish to obtain less conservative results.

Reference 2 also points to the ubiquity of 2-D Sn code calculations for dose rate calculations characterizing spent fuel casks. Oka makes the following statements.

- *Deterministic codes such as ANISN and DOT3.5 are usually used in shielding design for fission reactors. Recently, DORT and TORT codes in a DOORS system are often employed to*

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complete calculation results using the other codes. Deterministic method has an advantage than Monte Carlo because of no statistical error in the calculated values.

- *MCNP4A, DORT and DOT-3.5 were employed in both calculations.*
- *The results of comparisons between calculated values with MCNP4A and DORT and measured data indicate a good agreement for neutron dose rates and total neutron fluxes.*
- *Through the large-scale calculations mainly by two-dimensional Sn code DOT3.5 and its comparison with the measurement, the accuracy of FBR shielding analysis system was confirmed.*
- *All the measurement data were compared with analysis values which were mainly obtained by two-dimensional transport code DOT3.5 and DORT, and by the three-dimensional transport code TORT.*
- *Calculations of induced radioactivity and gamma-ray dose rate in the work area performed with DOT3.5, ORIGEN-79 and QAD codes.*
- *Under “Spent Fuel Transport Cask and [Shipping] Vessel”: The discrete ordinate Sn code DOT3.5 with group constants DLC-23 was employed to analyze the shielding problems of the NFT casks. Monte Carlo code MCNP4B was also used to double-check the shielding safety of the casks by the competent authority. Both the DOT3.5 and the MCNP4B calculations took into account the axial-direction burn-up distribution of the spent fuels for the neutron and gamma-ray source generation in the spent fuels and the detail configuration around the trunnion.*

These statements underscore the fact that discrete ordinates, DOT3, and DORT have been both ubiquitous and authoritative in what was then the world’s second largest nuclear power producer.

The Momma paper, Reference 3, makes similar statements, including the following.

- *The measured dose equivalent rates were 1/2 - 1/20 lower than the values estimated by the shielding design method which took the radiation source strength of the spent fuels loaded into the casks into consideration, confirming that the shielding design method used for the casks was conservative enough.*
- *The two-dimensional discrete ordinates Sn code DOT3.5 and the DLC-23 library were used for the dose equivalent rate calculations and the ORIGEN code was used for the source conditions estimations in the SAR (Safety Analysis Report) calculations **for all the NFT casks** [bolding added]. **The calculation methods are widely used for various cask shielding designs.***

This third reference clarifies the extent to which discrete ordinates and the immediate DORT predecessor have been used in spent fuel biological dose applications.

NUREG/CR-6802, Reference 4, makes the following statements, observations, and recommendations.

- *The report compared several major shielding codes in these analyses including SAS4/MORSE, MCNP, DORT, and MARMER, which is a point kernel code. The conclusion was that all these*

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codes gave results that were in general agreement with each other. MCNP4A, DORT and DOT-3.5 were employed in both calculations. The results of comparisons between calculated values with MCNP4A and DORT and measured data indicate a good agreement for neutron dose rates and total neutron fluxes.

- *The selection of flux-to-dose conversion factors is an important part of the overall shielding analysis. The accepted values for use in dose rate studies for cask shielding qualification are the ANSI/ANS 6.1.1-1977 values or their equivalent. The use of the latest version of this standard is not recommended since it predicts dose rate that are, in some instances, substantially lower than those of the 1977 standard and the NRC has not adopted the approach embodied in the 1991 updated standard.*
- *The use of discrete-ordinates methods is typically limited to the cavity and shield portions of the geometry. This limitation is due to both inefficiencies and inaccuracies in extending the solution through several meters of void or air. The recommended approach for these calculations is like that in XSDOSE (1-D) and FALSTF (2-D and 3-D), where the leakage from the shield is processed along with a last flight collision estimator to produce fast and accurate estimates of the flux and dose rates for external detectors.*
- *The typical approach for a cask shielding submittal is the calculation of dose rates from either a single limiting condition (specific burnup/enrichment/cooling time) or a series of limiting conditions that produces equivalent (or less than) dose results to that of the single bounding condition. Each of the specific conditions in the series has a maximum burnup, minimum enrichment, and minimum cooling time.*

Westinghouse has followed the recommendations above, making use of the 1977 dose conversion factors and avoiding the use of untreated discrete ordinates methods when ray effects might arise or when substantial distances of low density materials such as air are being modeled. The earlier FuelSolutions submittal took the recommendation of using the FALSTF last collision approach, and the current SENTRY submittal makes use of dose rates only at locations near the cask surface.

It is of interest to note that the final statement from NUREG/CR-6802 points to just the type of approach Westinghouse has used with the cooling tables constructed for the FuelSolutions and SENTRY dry storage system applications. Considering a large series of enrichment and burnup conditions as described can only be done efficiently with an adjoint approach. For a number of reasons, such an approach is impractical with the MCNP code but can be achieved readily through the use of the DORT adjoint capability.

The fifth document, Reference 5, states succinctly that all included work has been performed with the DORT code. It also makes the following statement.

- *The adequacy of the computer codes used to obtain the shielding and criticality results are discussed in refs. 1-5.*

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These citations include References 6 and 7, which lend substantial additional credence to the use of discrete ordinates methods for problems of just this type.

References

1. "Radiation Shielding Analysis of a Spent Fuel Transport Cask with an Actual Configuration Model Using the Monte Carlo Method – Comparison with the Discrete Ordinates Sn Method," Kohtaro Ueki, Advanced Reactor Technology Co., Ltd., et. al., 14th International Symposium on the Packaging and Transportation of Radioactive Materials (PATRAM 2004), Berlin, Germany, September 20-24, 2004.
2. "Radiation Shielding for Fission Reactors," Yoshiaki Oka, The University of Tokyo, Journal of Nuclear Science and Technology, Supplement 1, p. 1-10 (March 2000).
3. "Radiation Measurements and Shielding Calculations for Spent Fuel Casks," Y Momma, Tokyo Electric Power Co., et. al., Journal of Nuclear Science and Technology, Supplement 1, p. 342-346, March 2000.
4. NUREG/CR-6802, "Recommendations for Shielding Evaluations for Transport and Storage Packages," B. L. Broadhead, Oak Ridge National Laboratory, Published by U.S. Nuclear Regulatory Commission Office of Nuclear Material Safety and Safeguards, May 2003.
5. ORNL/TM-12395, "Shielding and Criticality Analyses of Phase I Reference Truck and Rail Cask Designs for Spent Nuclear Fuel," B. L. Broadhead, et. al., June 1996.
6. C. V. Parks et al., "Assessment of Shielding Analysis Methods, Codes, and Data for Spent Fuel Transport/Storage Applications," ORNL/CSD/TM-246, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., July 1988.
7. B. L. Broadhead, M. C. Brady, and C. V. Parks, "Benchmark Shielding Calculations for the NEACRP Working Group on Shielding Assessment of Transportation Packages," ORNL/CSD/TM-272, Martin Marietta Energy Systems, Inc., Oak Ridge Natl. Lab., November 1990.

Original 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 individual plant ISFSI site boundary dose calculations.

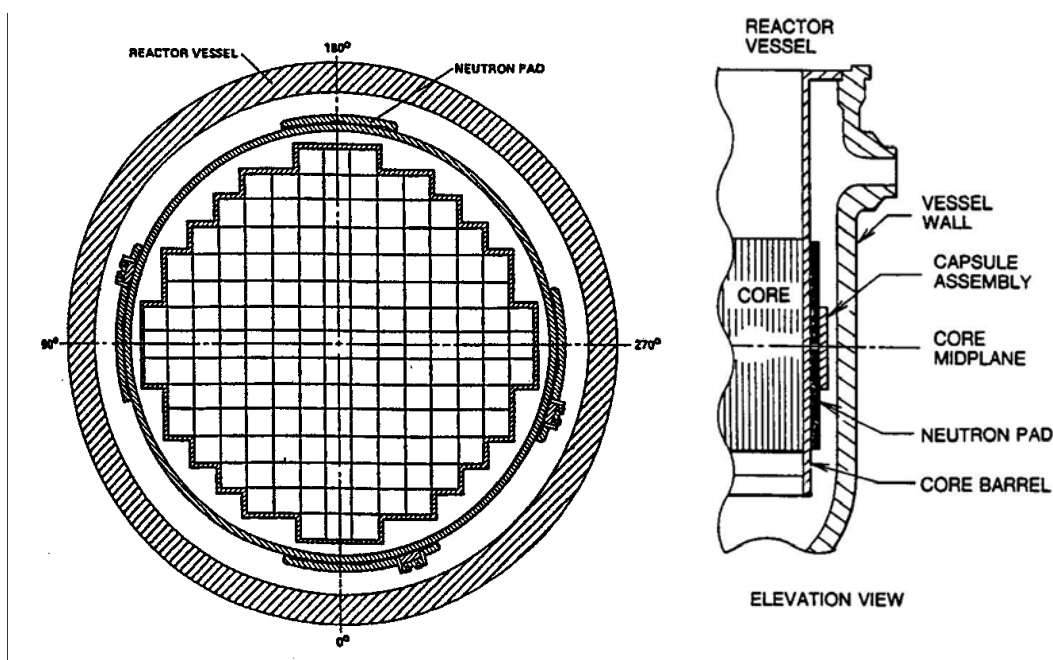
The techniques and methodology for the cask shielding work are the same as those used for the analysis of reactor vessel fluence in accordance with US NRC Regulatory Guide 1.190. In both cases,

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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.

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.



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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 simulator³ 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 experiment⁴.
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™ 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 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 work was performed with a P₅ expansion order and an S₁₆ order of angular quadrature. This exceeds the minimum required by WCAP-17459-P, which calls for models with at least a P₃ Legendre expansion and at least an S₈ order of angular quadrature. Those requirements are in compliance with the requirements of Regulatory Guide 1.190. The approach taken in the SENTRY work is also in

³ I. Remec and F. B. K. Kam, "Pool Critical Assembly Pressure Vessel Facility Benchmark," NUREG/CR-6454, July 1997.

⁴ I. Remec and F. B. K. Kam, "H. B. Robinson Pressure Vessel Benchmark," NUREG/CR-6453 (ORNL/TM-13204), February 1998.

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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 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 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-104329⁵.

Documents In Reading Room Folder for RSI 5-1

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

⁵ Broadhead, B.L., et al. Evaluation of Shielding Analysis Methods in Spent Fuel Cask Environments, EPRI TR-104329, May 1995.