DEPARTMENT OF ENERGY
WEST VALLEY DEMONSTRATION PROJECT
REVIEW OF CONTRACT DELIVERABLES FORM

DELIVERABLE NUMBER: 80-DLVR-062613

TITLE: Documented Safety Analysis Update – WVNS-DSA-001, Rev. 17
Technical Safety Requirements – WVDP-146, Rev. 10

DATE: June 26, 2013

REVIEWER: U.S. Nuclear Regulatory Commission

Comment Number	Page No, Section or Paragraph No.	Comments	Response
1	Chapter 1	**Comment 1-C1:** Section 1.5, DOE-STD-1020-2002 does not require any design requirements for tornadoes, tornado driven projectiles/missiles or straight wind driven projectiles/missiles. Also, in the last sentence of the third paragraph of Section 2.4.5.4 it is stated that the "shield plug and lid … provide a cover and seal to protect the canister from the environment and postulated tornado missiles." However, on page 44 of 462 of the Document Safety Analysis (DSA) it is stated that there is no design requirement for projectiles/missiles. The only required wind-related design mitigation is for a straight wind. Consequences from credible natural phenomenon should be addressed. **Basis:** The cask must be analyzed to show that it will not slide, tip over, or drop in its storage condition as a result of a credible natural phenomenon event, including tornado winds and tornado missiles. Confinement casks are generally not vulnerable to damage from overpressure or negative pressure associated with tornadoes or extreme winds. However, they may be vulnerable to secondary effects, such as wind-borne missiles. **Path Forward:** Regulatory Guide 1.76, "Design Basis Tornado for Nuclear Power Plants" and NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition – Design of Structures, Components, Equipment, and Systems" (Section 3.5.1.4)	Natural phenomena hazards (NPH) including straight wind loads and seismic accelerations were evaluated as part of the design of the vertical storage cask (VSC). Citations related to NPH calculations performed by NAC International, calculation number 630087-2010, *MPC-WVDP VSC Structural Evaluation*, with associated basis will be added to the Documented Safety Analysis (DSA). As the comment acknowledges, DOE-STD-1020-2002, *Natural Phenomena Hazards for Design and Evaluation Criteria for Department of Energy Facilities*, does not impose tornado or tornado missile design requirements for Performance Category (PC) 2 facilities. In accordance with DOE-STD-1021-93, *Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components*, Hazard Category (HC) 3 facilities (higher than the HC of the high-level waste storage system [HLWSS]) are PC 2 or less. Consistent with the guidance provided relative to descriptive information in a DSA in DOE-STD-3009-94, *Preparation Guide for U.S. Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*, no additional NPH information is required to be provided for the WVDP vertical storage cask (VSC). Nevertheless, a qualitative assessment of the

		describe tornado winds and missiles. The guidance in the aforementioned documents for tornado missile protection should be consulted	impacts to the MPC-WVDP system due to tornado borne missiles using NAC International analyses performed for the NAC MPC-Yankee system will be provided.
2		**Comment 1-C2:** The environmental characterization is not sufficient. Recent finding on the potential chloride-induced stress corrosion cracking (SCC, or related other corrosion) needs to be addressed in Man-made External Accident Initiators (Section 1.6). Chlorides may be deposited on the canister surface from de-icing salts on the road. **Basis:** Some dry cask storage system (DCSS) designs utilize austenitic stainless steel canisters surrounded by concrete shielding structures to store spent nuclear fuel (SNF) at Independent Spent Fuel Storage Installations (ISFSIs). ISFSIs that are located where chlorides can be deposited on the canisters may have the potential for initiating chloride-induced stress corrosion cracking (SCC). Susceptibility to chloride-induced SCC depends on the environmental conditions at the canister surface, including the following key parameters: temperature; relative humidity (RH); areal density, composition and aqueous concentration of deposited salts; and the stress state of the canister, particularly in the weld and the weld heat affected zone. Recently national and international literature was reviewed and the studies are continuing internationally, additionally including chloride-induced crevice corrosion and microbially-influenced corrosion (NRC, 2012; SERCO, 2010). The West Valley Nuclear Services (WVNS) canisters may have similar susceptible environments to corrosion with chlorides present from de-icing salts on the road (Barber, et al., 2001) and potential microbes, at expected lower temperature compared to DCSS due to lower heat loading and seasonal environmental temperature variations. Lower temperature, below ~80 °C (176 °F) is needed to form aqueous environments by salt deliquescence on the canister surface. If any corrosion were to penetrate through the canister wall, potential consequence of radionuclide release may occur especially under off-normal and accident conditions. The high- level waste (HLW) glass may be hydrated with environmental moisture (NRC, 2008) or swollen with radiation (Donald, et al., 1997). These may in turn cause radionuclide release by increasing radionuclide release fraction, especially caused by any impact under off-normal and accident conditions.	As indicated on Figure 2.4-36, the inner shell of the concrete VSC is made of ASTM A36 carbon steel. The West Valley Demonstration Project (WVDP) VSCs have no direct pathway (i.e., ventilation ports) for chloride salts to contact the multi-purpose canister (MPC) or the vitrification canister within the MPC (see Figure 2.4-37). Due to the multiple physical barriers present in the DCSS (i.e., concrete shield, carbon steel liner, stainless steel vitrification canister), the potential for chloride-induced corrosion is minimal. The consequences of unspecified corrosion resulting in MPC and cask failure were qualitatively evaluated in Table 3.3-1, ID No. HLWSS-10. The unmitigated consequence level is identified as negligible for both the co-located worker and the public. Additionally, previous safety analysis reports (WVNS-SAR-003) prepared during vitrification operations demonstrated that unmitigated consequences from catastrophic melter failure and loss of greater than one canister volume of glass resulted in consequences that are within the DOE evaluation guidelines (see NRC Accession Number 9505120248).

		Path Forward: The report needs to address potential chloride-induced corrosion and its consequence in the storage of HLW glass. **References:** S.H. Barber, N.W. Sachs, and R.N. Taylor, "*Long-Term Monitoring of Two Large Process Vessels*," Materials Performance, pp. 60-63, December 2001. I.W. Donald, B.L. Metcalfe, and R.N. Taylor, "*Review of the Immobilization of High Level Radioactive Wastes Using Ceramics and Glasses*," J. of Materials Science, Vol. 32, pp. 5851-5887, 1997. U.S. Nuclear Regulatory Commission (NRC), "Identification and Prioritization of the Technical Information Needs Affecting Potential Regulation of Extended Storage and Transportation of Spent Nuclear Fuel," NRC ADAMS ML120580143, 2012. NRC, "Dissolution Kinetics of Commercial Spent Nuclear Fuels in the Potential Yucca Mountain Repository Environment," NRC ADAMS ML083120074, NUREG-1914, 2008. SERCO, "Review of Environmental Conditions for Storage of ILW Radioactive Waste Containers," Report to NDA RWMD, SERCO/TASIE.2098/P3443, Issue 04, U.K. 2010.	
3	Chapter 2	**Comment 2-C1:** In Section 2.4.5.3 on page101 (4th paragraph from the top), revise the statement, "the closure lid is also provided with 3 lifting points for the purpose of remote closure lid removal and installation," and associated description by recognizing that a 3-sling lifting amounts to a non-redundant configuration. **Basis**: A three-point lifting is statically determinate and loss of any one lifting point will result in uncontrolled move of the load. Thus, each lifting leg should be sized for critical lift criteria with enhanced safety factors of 6 on yield strength and 10 on ultimate per ANSI N14.6 or NUREG-	ANSI N14.6 applies to containers weighing 10,000 pounds or more, and for those features of the attachment members of the container that affect the function and safety of the lift of the container. In this case, the lid lifting points are only for use to install or remove the lid, which weighs 4,400 pounds. Therefore, DOE-STD-1090-2011, *Hoisting and Rigging*, is being used for lid installation.

		0612.	
4	Chapter 2	**Comment 2-C2:** This comment references Figure 2.4-33, Drawing 630087, on page 195. Revise, as appropriate, the drawing by providing notes on inspection and maintenance schedules for item 8, "Parker O-Ring," and item 12, "Outer Gasket," to ensure the vertical storage cask (VSC) designed service life of 50 years. **Basis:** There is no indication in the Bill of Materials that the subject O-rings and gasket are qualified for the VSC design service life of 50 years.	The VSC design service life of 50 years is intended to be applied to the structural integrity of the VSC as indicated in the *HLW Canister Relocation and Storage System* design criteria (WVNS-DC-074). The design criteria do not quantify the confinement characteristics of the VSC. The EPDM gasket and O-ring do not provide any structural integrity function; however, they do provide a confinement boundary. EPDM and Parker o-rings (Spec. E0740-75) were chosen based on their compression characteristics over the temperature range of interest, springback adequacy, ability to withstand a water environment, radiation resistance, and suitability for the joints required to withstand impact loads. The seals have a temperature limit of 120°C (248°F) under normal operations and 127°C (260°F) under off-normal and accident conditions.
5	Chapter 2	**Comment 2-C3:** There is no Vertical Storage Cask (VSC) confinement design features information provided in the WVDP Documented Safety Analysis (DSA). **Basis:** To clarify the confinement design features (leak tight, non-leak tight, or maximum allowable leakage rate) of the VSC. **Path Forward:** Add a statement to clarify whether the VSC is leak tight or not. If not leak tight, the maximum leakage rate, per 10 CFR Part 72 and ANSI N14.5, should be provided.	The design criteria document (WVNS-DC-074) does not specify a maximum allowable leakage rate for the VSC. The HLW canisters do have a specified leakage rate of less than 10^{-7} atm-cc/sec helium (DSA section 2.4.1.1.1.2) that has been demonstrated in conformance with DOE/EM-0093, *Waste Acceptance Product Specifications (WAPS) for Vitrified High-Level Waste Forms,* and DOE/RW-0351, *Waste Acceptance Requirements Document*.
6	Chapter 2	**Comment 2-C3:** Section 2.4.5 High Level Waste Storage System (page 99) only describes the principal components of the VSC (including a basket to accommodate five WVDP high-level waste (HLW) canisters, a stainless steel HLW overpack, and a concrete and steel VSC) with no information of its confinement boundary and components provided. **Basis:** To clarify the confinement boundary of the VSC and its confinement components which should be helium-leak tested. **Path Forward:** Provide the confinement boundary of the VSC and identify its confinement components which should be helium-leak tested.	See response to NRC Comment 2-C3 (5).

7	Chapter 2	**Comment 2-C4:** Section 2.4.5.3, the HLW Overpack (page 100) delineates that the VSC is designed to accept an MPC-WVDP 5-cell basket assembly that is sized to accommodate five HLW canisters and to incorporate a welded single closure lid design. There is no description of the single closure weld for its confinement quality. **Basis:** To clarify that the welded closure lid provides a cover and seal capable of protecting the canisters from the environment. **Path Forward:** Describe, as appropriate, how the welded seal joining the single closure lid to the MPC-WVDP HLW overpack is performed, examined, and tested at shop to assure its confinement effectiveness.	See response to NRC Comment 2-C3 (5).
8	Chapter 2	**Comment 2-C5:** Section 2.4.5.3, the HLW Overpack (page 101) notes that all HLW overpack vessel shop welds are liquid penetrant (PT) examined after loading in accordance with Section VIII, Division 2 visual acceptance standards. **Basis:** To assure reliability of the PT examination on the hot closure lid surface and the quality of the closure weld. **Path Forward:** Provide the maximum temperature of the closure lid surface on which the liquid penetrant test (PT) was performed.	The closure lid will not be welded in the shop, but rather remotely welded in the WVDP Equipment Decontamination Room (EDR) or the Load-In/Load-Out (LI/LO). As indicated in section 2.5.3.2, *MPC-WVDP Loading Preparations*, following completion of the root pass, the lid root pass weld will be inspected using remote (by camera) visual (VT) inspection methods in accordance with ASME Code, Section III, Subsection NF VT criteria. After the root pass weld examination has been successfully completed, closure lid welding will continue until the final weld layer is installed and the final weld surface visual examination is completed in accordance with the ASME Code VT acceptance criteria.
9	Chapter 2	**Comment 2-C6:** There is no description of the vertical storage cask thermal design features, material temperature limits, thermal loads and environmental conditions, and analytical methods, models, or calculations. **Basis:** The heat transfer characteristics of the storage system, any material temperature limits, and all necessary inputs to perform a realistic or conservative thermal evaluation of the vertical storage cask are required to confirm the thermal	Thermal evaluation was performed in NAC International calculation numbers 630087-2010, 630087-2015, and 630087-3000. The thermal stress evaluation of the VSC and internal components was performed for the extreme minimum environmental temperature (-40°F), average ambient (normal) temperature (75°F), and extreme maximum environmental temperature (110°F) conditions and found to meet the structural requirements of ANSI

		performance of the storage system. **Path Forward:** In order to complete a review, the DSA would need to provide the design features, material temperature limits (if any), thermal loads, and material characterization that are used to perform the thermal evaluation of the vertical storage cask. Chapter 6 of NUREG-1567 "Standard Review Plan for Spent Fuel Dry Storage Facilities" and Chapter 4 of NUREG-1536 "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" provide guidance on how the staff reviews these systems.	57.9, ASME Boiler and Pressure Vessel Code Section III, ACI-349-06, and DOE-STD-1090-2011.
10	Chapter 2	**Comment 2-C7:** There is no thermal analysis of the vertical storage cask. **Basis:** This information is necessary to make a safety determination based on a thermal evaluation that demonstrates that predicted material temperatures remain below acceptable limits with adequate margin. **Path Forward:** In order to complete a review, the DSA would need to provide a thermal evaluation and predicted material temperatures to demonstrate that the temperatures remain below allowable limits. Chapter 6 of NUREG-1567 "Standard Review Plan for Spent Fuel Dry Storage Facilities" and Chapter 4 of NUREG-1536 "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility" provide guidance on how the staff reviews these systems.	See response to NRC Comment 2-C3 (5).
11	Chapter 2	**Comment 2-C8:** The temperature of the canister surface and the HLW glass needs to be assessed. The degradation of the canister and HLW glass is likely to be sensitive to the thermal loading in the storage system. **Basis:** Comment 1-C2 describes how the temperature could affect the integrity of the canister and HLW glass. Canister corrosion may occur at temperatures below ~ 80 °C (176 °F) with sufficient RH (e.g., aqueous corrosion). The hydration of HLW glass would occur at temperatures below 230 °C (446 °F). Radiation effects may be annealed at higher temperatures. **Path Forward:** Thermal analyses need to be included to assess the temperature of canister and HLW glass.	See response to NRC Comment 2-C3 (5).

12	Chapter 2	**Comment 2-C9:** In Section 2.5.3.1.2, page 116 of 462 of the DSA, high-level mixed waste, spent nuclear fuel co-mingled with highly dispersible particulate debris (classified as RCRA hazardous waste) is proposed to be stored on the storage pad. There is not an adequate description that permits discernment of whether storage of this high-level mixed waste is appropriate for storage in the cask system. **Basis:** 10 CFR Part 72 provides no regulatory provisions for storage of high-level mixed waste. **Path Forward:** Clarify the regulatory basis for storage of high-level mixed waste.	The WVDP HLW (vitrified waste form) is not a mixed waste. The RCRA hazardous waste as described in the cited section will be segregated and will not be stored in the HLWSS.
13	Chapter 2	**Comment: 2-C10:** Section 2.2.2.3, page 67 of 462 of the DSA, states that "no special considerations are required to protect against general site flooding." NRC's review evaluates site characteristics to determine if natural phenomena such as floods have been properly identified, quantified, and included in the ISFSI design bases. The effects of natural phenomena (e.g., floods) are considered to be accident events. Specific guidance for how the NRC conducts this review is presented in Chapters 2 and 15 of NUREG-1567,"Standard Review Plan for Spent Fuel Dry Storage Facilities." Additional guidance for flood protection is given in Regulatory Guides 1.59, "Design Basis Floods for Nuclear Power Plants," and 1.102, "Flood Protection for Nuclear Power Plants." **Basis:** Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform their intended design functions. The design bases for these structures, systems, and components must reflect: appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. **Path Forward:** The guidance in the aforementioned documents should be consulted.	The design basis for flooding for the WVDP can be found in section 1.4.2.1.2, *Floods*. NPH have been evaluated in accordance with 10 CFR 830, Subpart B, *Safety Basis Requirements*, DOE-STD-3009-94, *Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*, DOE-STD-1021-93, *Natural Phenomena Hazards Performance Categorization Guidelines for Structures, Systems, and Components*, and DOE-STD-1020-2002, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*.

14	Chapter 3	**Comment 3-C1:** For the hazard of waste container over-pressurization listed in Table 3.3-1, the preventive engineered control is required under OUT-9 and RH-6, but not required under HLWSS-8. **Basis**: To assure whether the preventive engineered control is needed in HLWSS-8 to prevent the potential risk of deflagration (at unloading) due to hydrogen generation in canisters or overpacks during long-term storage. **Path Forward**: Explain in the DSA why the preventive engineered control is not required under HLWSS-8. If required, add the control under HLWSS-8 of Table 3.3-1.	No preventive or mitigative engineered controls are required for any sequence identified in Table 3.3-1. All unmitigated risk is within the Risk Evaluation Guidelines presented in Table 3.3-3. The Risk Evaluation Guidelines are consistent with those presented in DOE-STD-3009-94, *Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*, Appendix A, *Evaluation Guideline*.
15	Chapter 3	**Comment 3-C2:** For the hazard of loss of active ventilation listed in Table 3.3-1, the preventive engineered control is required under MP-15 to prevent loss of airborne contamination confinement, but not required under MPA-14 and OUT-18. **Basis**: To assure whether the preventive engineered control is needed under MPA-14 and OUT-18 to further prevent the negative end scenarios such as loss of all exhaust blowers servicing CSRF or loss of container integrity. **Path Forward**: Explain in the DSA why the preventive engineered control is not required under MPA-14 and OUT-18. If required, add the control under MPA-14 and OUT-18 of Table 3.3-1.	See response to Comment 3-C1 (14).
16	Chapter 3	**Comment 3-C3**: Justifications with details for risk assessment results (Table 3.3-4) need to be provided. **Basis:** The DSA considered both (i) design basis approach using codes and standards and (ii) risk approach considering probability (frequency) and dose consequences. A large number of supporting documents are quoted for the summary made in this report. Therefore, it is difficult to understand the extent in using the two approaches, and bases for determining probabilities and consequences. **Path Forward:** The DSA needs a summary of the bases of Table 3.3-4, including the consideration of the design basis approach.	Chapter 3 has been prepared using the graded approach described in DOE-STD-3009-94, *Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*: *Analytical effort can be limited to a simple, resource efficient hazard analysis geared to facility needs, unless events are noted that are of sufficient complexity to require more detailed, quantitative evaluations to understand the basis for safety assurance. Implicit in this methodology is the statement of DOE-STD-1027 that the largely qualitative*

			level of effort in hazard analysis is appropriate and sufficient for accident analysis of Hazard Category 3 facilities. DOE-STD-5506-2007, *Preparation of Safety Basis Documents for Transuranic (TRU) Waste Facilities*, describes the approaches used in preparation of Chapter 3. It should also be noted that the HLWSS has been categorized as a below HC 3 facility, using DOE-STD-1027-92, *Hazard Categorization and Accident Analysis Techniques for Compliance with DOE Order 5480.23, Nuclear Safety Analysis Reports.* As a below HC 3 facility, a DSA is not required. Therefore, the level of description and hazard analysis has been prepared commensurate with the potential for significant airborne releases. In all cases the probability of failure is based upon the factor of safety afforded by the design bases, derivative design bases, or the initiator frequency. All consequences are based upon the described failure mode and guidance provided in DOE-HDBK-3010-94, *Airborne Release Fractions/Rates and Respirable Fractions for Non-Reactor Nuclear Facilities.*
17	Chapter 6	**Comment 6-C1:** There is no complete description of the contents that will be loaded into the multi-purpose canister (MPC) storage canisters. **Basis:** Reviewers need to understand the material that will be loaded into the MPC to be able to make a finding on its criticality safety. **Path Forward:** Provide a description of the contents that will be loaded into the MPC including amount of fissile material, nuclides and other materials present that may act as a moderator as well as the geometry of all materials.	Chapter 6 was developed using DOE-STD-3009-94, *Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*: *The purpose of this chapter is to provide information that will support the development of a safety basis in compliance with the provisions of 10 CFR 830.204(b) (6) regarding the definition of a criticality safety program. If this information is available in a site-wide criticality safety program description, and it complies with the Rule requirements, then it can be included by reference and summarized in this chapter.* WVNS-NCSE-002: *Criticality Safety Evaluation for the Handling and Storage of Fissile Bearing Debris in*

			the Head End Cells was developed for the spent nuclear fuel (SNF) debris that will be canistered and loaded into the MPC. NRC has reviewed the complete vitrification process (see Accession Number 9505120248), including the terminal waste form, and concluded that "Based on the analyses reviewed, the vitrification process is considered to be reasonably safe with respect to criticality safety for both normal and abnormal conditions"
18	Chapter 6	**Comment 6-C2:** There is no criticality analysis of the contents within the MPC. **Basis:** The reviewer makes criticality safety findings based on calculations that demonstrate that storage packages are subcritical. **Path Forward:** Perform a calculation demonstrating that the MPC contents are subcritical. Guidance for how the staff reviews this type of calculation is in Chapter 8 of NRC's standard review plan, NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities" (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1567/sr1567.pdf).	The vitrified HLW contains < 450 g fissile mass distributed throughout the borosilicate glass matrix with a minimum weight of about 1,800 kg. This concentration is less than the fissile material exemption limit from NUREG/CR-5342, *Assessment and Recommendations for Fissile-Material Packaging Exemptions and General Licenses within 10 CFR Part 71*. The NRC has evaluated the criticality safety of the vitrification process (see Accession Number 9505120248) and concluded that "Based on the analyses reviewed, the vitrification process is considered to be reasonably safe with respect to criticality safety for both normal and abnormal conditions". The SNF debris from the Head End Cells has already been evaluated by the NRC and found to not pose an undue risk of inadvertent criticality (Accession Number ML012840528). The storage configuration in the MPC is bounded by the nuclear criticality safety evaluations performed for the decommissioning of the Head End Cells. Section 2.5.3 reports that the SNF debris contains less than 180 FGE Pu-239 which is below the minimum critical mass. Calculations for criticality requirements are provided in NAC International calculation number 630087-6001, *Criticality Analysis for HLW and SNF in the WV-MPC System*.

19	Chapter 7	**Comment 7-C1:** There is no complete description of the source term that will be loaded into the MPC storage canisters. **Basis:** A description of the source term to understand the material that will be loaded into the MPC is required to be able to make a finding that the MPC meets shielding and radiation protection regulations. **Path Forward:** Provide a description of the neutron and gamma source terms that will be loaded into the MPC.	Table 9.4-3, *Typical WVDP High Level Waste Canister Characteristics (2014)*, contains the radiological inventory information. Complete radionuclide inventories are contained in WVNS-CAL-396, *Estimation of Radioactivity in WVDP High Level Waste Canisters.* Calculations for shielding and radiation protection requirements are provided in NAC International calculation number 630087-5001, *West Valley MPC VSC Shielding and Source Term Evaluations.*
20	Chapter 7	**Comment 7-C2:** There is no shielding analysis of the contents within the transfer cask or storage MPC. **Basis:** The calculated site boundary dose and surface dose rates on the transfer cask and storage overpack are used to determine if the system and its contents meet radiological safety regulations. **Path Forward:** Perform a shielding analysis calculating transfer cask and storage overpack dose and dose rates. Guidance for how the staff reviews this type of calculation is in Chapter 7 of NRC's standard review plan, NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities" (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1567/sr1567.pdf).	The storage cask is being used as the transfer cask. See response to Comment 7-C1 (19).
21	Chapter 7	**Comment 7-C3:** There is no radiation protection evaluation. **Basis:** This information is needed to determine if the system and its contents meet radiation protection regulations. **Path Forward:** Provide information related to radiation protection such as operational procedures for loading and estimate of occupational doses during transfer cask operations. Guidance for how the staff reviews this evaluation is in Chapter 11 of NRC's standard review plan, NUREG-1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities" (http://www.nrc.gov/reading-rm/doc-collections/nuregs/staff/sr1567/sr1567.pdf).	The radiation protection program at the WVDP is in compliance with 10 CFR 835, *Occupational Radiation Protection*. The DSA was prepared in accordance with 10 CFR 830, Subpart B, *Safety Basis Requirements*, DOE-STD-3009-94, *Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses*. Guidance provided in DOE-STD-3009-04 indicates: *The purpose of this DSA chapter is to provide information that will satisfy the requirements of 10 CFR 830. This chapter is not intended to be the vehicle for review and approval of the radiation protection program. It is intended to describe the essential*

			characteristics of the program as it relates to facility safety.
22	WVDP-146	**Comment TSR-C1:** The technical safety requirements document has none of the procedures contained in standard technical specifications. **Basis:** Technical specifications for dry cask storage systems are intended to be a clear and consistent set of procedures that identify: 1) approved contents; 2) limiting conditions for operation and applicability; 3) surveillance requirement and applicability (e.g., fuel integrity; cask integrity, and cask criticality control program); 4) design features (e.g., design features significant to safety; codes and standards; structural performance; and cask handling/canister transfer facility); and 5) administrative controls. These details in the dry cask technical specifications will assure the overall safety goals for dry cask storage are met, including maintaining subcriticality, controlling radiation dose to the workers and the public, and maintaining the confinement barriers. NUREG-1745 "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance," provides guidance on the format and level of detail expected in technical specifications. **Path Forward:** Provide information related to technical specifications. Guidance for how the staff reviews technical specifications is documented in NUREG-1745 "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance".	The Technical Safety Requirements (TSRs) were prepared in accordance with 10 CFR 830, Subpart B, *Safety Basis Requirements*, DOE-STD-3009-94, *Preparation Guide for U.S Department of Energy Nonreactor Nuclear Facility Documented Safety Analyses,* DOE-STD-5506-2007, *Preparation of Safety Basis Documents for Transuranic (TRU) Waste Facilities*, and DOE-STD-1186-2004, *Specific Administrative Controls*. The TSRs were developed using the graded approach guidance of DOE-STD-3009-94: *For Hazard Category 3 facilities, TSRs may consist solely of an inventory limit to maintain the Hazard Category 3 classification and provide appropriate commitments to safety programs in the administrative controls section of TSRs.* The HLWSS is a below HC 3 facility and does not require TSRs as derived from Chapter 5, *Derivation of Technical Safety Requirements*, in the DSA.