[7590-01-P]

NUCLEAR REGULATORY COMMISSION

[Docket Nos. 72-58 and 50-263; NRC-2018-0207]

Xcel Energy, Monticello Nuclear Generating Plant; Independent Spent Fuel Storage Installation

AGENCY: Nuclear Regulatory Commission.

ACTION: Exemption; issuance.

SUMMARY: The U.S. Nuclear Regulatory Commission (NRC) is issuing an exemption in response to a request submitted by Xcel Energy on October 18, 2017, from meeting Technical Specification (TS) 1.2.5 of Attachment A of Certificate of Compliance (CoC) No. 1004, Amendment No. 10, which requires that all dry shielded canister (DSC) closure welds, except those subjected to full volumetric inspection, be dye penetrant tested in accordance with the requirements of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel (B&PV) Code Section III, Division 1, Article NB-5000. This exemption applies to five loaded Standardized NUHOMS® 61BTH, Dry Shielded Canisters (DSCs) 11 through 15, at the Monticello Nuclear Generating Plant (MNGP) Independent Spent Fuel Storage Installation (ISFSI).

ADDRESSES: Please refer to Docket ID **NRC-2018-0207** when contacting the NRC about the availability of information regarding this document. You may obtain publicly-available information related to this document using any of the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2018-0207**. Address questions about Docket IDs in Regulations.gov to Jennifer Borges; telephone: 301-287-9127; e-mail: Jennifer.Borges@nrc.gov. For technical questions, contact the individual(s) listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **NRC's Agencywide Documents Access and Management System (ADAMS):**

You may obtain publicly-available documents online in the ADAMS Public Documents collection at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The ADAMS accession number for each document referenced (if it is available in ADAMS) is provided the first time that it is mentioned in this document. In addition, for the convenience of the reader, the ADAMS accession numbers are provided in a table in the "Availability of Documents" section of this document.

- **NRC's PDR:** You may examine and purchase copies of public documents at the

NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

FOR FURTHER INFORMATION CONTACT: Christian Jacobs, Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; telephone: 301-415-6825; e-mail: Christian.Jacobs@nrc.gov.

SUPPLEMENTARY INFORMATION:

I. Background

Northern States Power Company-Minnesota, doing business as Xcel Energy (Xcel Energy, or the applicant) is the holder of Renewed Facility Operating License No. DPR-22, which authorizes operation of the MNGP, Unit No. 1, in Wright County, Minnesota, pursuant to part 50 of title 10 of the *Code of Federal Regulations* (10 CFR), "Domestic Licensing of Production and Utilization Facilities." The license provides, among other things, that the facility is subject to all rules, regulations, and orders of the NRC now or hereafter in effect.

Consistent with 10 CFR part 72, subpart K, "General License for Storage of Spent Fuel at Power Reactor Sites," a general license is issued for the storage of spent fuel in an ISFSI at power reactor sites to persons authorized to possess or operate nuclear power reactors under

10 CFR part 50. The applicant is authorized to operate a nuclear power reactor under 10 CFR part 50, and holds a 10 CFR part 72 general license for storage of spent fuel at the MNGP ISFSI. Under the terms of the general license, the applicant stores spent fuel at its ISFSI using the TN Americas LLC Standardized NUHOMS® dry cask storage system in accordance with CoC No. 1004, Amendments No. 9 and No. 10. As part of the dry storage system, the DSC (of which the closure welds are an integral part) ensures that the dry storage system can meet the functions of criticality safety, confinement boundary, shielding, structural support, and heat transfer.

II. Request/Action

The applicant has requested an exemption from the requirements of 10 CFR 72.212(a)(2), 10 CFR 72.212(b)(3), 10 CFR 72.212(b)(5)(i), 10 CFR 72.212(b)(11), and 10 CFR 72.214 that require compliance with the terms, conditions, and specifications of CoC No. 1004, Amendment No. 10, for the Standardized NUHOMS® Horizontal Modular Storage System, to allow continued storage of DSCs 11–15 in their respective Horizontal Storage Modules (HSMs). This would permit the continued storage of those five DSCs for the service life of the canisters. Specifically, the exemption would relieve the applicant from meeting TS 1.2.5 of Attachment A of CoC No. 1004 (ADAMS Accession No. ML17338A114), which requires that all DSC closure welds, except those subjected to full volumetric inspection, be dye penetrant tested in accordance with the requirements of the ASME B&PV Code Section III, Division 1, Article NB-5000. Technical Specification 1.2.5 further requires that the dye penetrant test (PT) acceptance standards be those described in Subsection NB-5350 of the ASME BP&V Code.

Xcel Energy loaded spent nuclear fuel into six 61BTH DSCs starting in September 2013. Subsequent to the loading, it was discovered that certain elements of the PT examinations, which were performed on the DSCs to verify the acceptability of the closure welds, do not comply with the requirements of TS 1.2.5. All six DSCs were affected. Five of the six DSCs (numbers 11–15) had already been loaded in the HSMs when the discrepancies were discovered. DSC 16 remained on the reactor building refueling floor in a transfer cask (TC). On June 8, 2016, NRC granted an exemption (ADAMS Accession No. ML16159A227) from 10 CFR 72.212(a)(2), 10 CFR 72.212(b)(3), 10 CFR 72.212(b)(5)(i), 10 CFR 72.212(b)(11), and 10 CFR 72.214 for DSC 16 only with regard to meeting TS 1.2.5 of Attachment A of CoC No.1004, Amendment No. 10. The exemption granted on June 8, 2016, restored DSC 16 to compliance with 10 CFR part 72 and allowed Northern States Power Company-Minnesota to transfer DSC 16 into an HSM for continued storage at MNGP ISFSI for the service life of the canister.

In a letter dated October 18, 2017 (ADAMS Accession No. ML17296A205) (Exemption Request), as supplemented in responses to NRC requests for additional information dated April 5, 2018 (ADAMS Accession No. ML18100A173) (RAI Response 1) and May 31, 2018 (ADAMS Accession No. ML18151A870) (RAI Response 2), the applicant requested an exemption from the following requirements to allow continued storage of the remaining DSCs 11–15 in their respective HSMs at the MNGP ISFSI:

- 10 CFR 72.212(a)(2), which states that this general license is limited to storage of spent fuel in casks approved under the provisions of part 72;
- 10 CFR 72.212(b)(3), which states that the general licensee must ensure that each cask used by the general licensee conforms to the terms, conditions, and specifications of a CoC or an amended CoC listed in 10 CFR 72.214;
- 10 CFR 72.212(b)(5)(i), which requires that the general licensee perform written evaluations, before use and before applying the changes authorized by an amended CoC to a cask loaded under the initial CoC or an earlier amended CoC, which establish that the cask, once loaded with spent fuel or once the changes authorized by an amended CoC have been applied, will conform to the terms, conditions, and specifications of a CoC or an amended CoC listed in 10 CFR 72.214;
- 10 CFR 72.212(b)(11), which states, in part, that the licensee shall comply with the

terms, conditions, and specifications of the CoC and, for those casks to which the licensee has applied the changes of an amended CoC, the terms, conditions, and specifications of the amended CoC; and

- 10 CFR 72.214, which lists the approved spent fuel storage casks.

III. Discussion

Pursuant to 10 CFR 72.7, the Commission may, upon application by any interested person or upon its own initiative, grant such exemptions from the requirements of the regulations of 10 CFR part 72 as it determines are authorized by law and will not endanger life or property or the common defense and security and are otherwise in the public interest.

Authorized by Law

This exemption would permit the continued storage of DSCs 11–15 at the MNGP ISFSI for the service life of the canisters by relieving the applicant of the requirement to meet the PT requirements of TS 1.2.5 of Attachment A of CoC No. 1004. The provisions in 10 CFR part 72 from which the applicant is requesting exemption require the licensee to comply with the terms, conditions, and specifications of the CoC for the approved cask model it uses. Section 72.7 allows the NRC to grant exemptions from the requirements of 10 CFR part 72. As explained below, the proposed exemption will not endanger life or property, or the common defense and security, and is otherwise in the public interest. Issuance of this exemption is consistent with the Atomic Energy Act of 1954, as amended, and not otherwise inconsistent with NRC's regulations or other applicable laws. Therefore, the exemption is authorized by law.

Will Not Endanger Life or Property or the Common Defense and Security

This exemption would relieve the applicant from meeting TS 1.2.5 of Attachment A of CoC No. 1004, which requires PT examinations to be performed on the DSCs to verify the acceptability of the closure welds, and would permit the continued storage of DSCs 11–15 in their respective HSMs at the MNGP ISFSI for the service life of the canisters. As detailed below, NRC staff reviewed the exemption request to determine whether granting of the exemption would cause potential for danger to life, property, or common defense and security.

Review of the Requested Exemption

The NUHOMS® system provides horizontal dry storage of canisterized spent fuel assemblies in an HSM. The cask storage system components for NUHOMS® consist of a reinforced concrete HSM and a DSC vessel with an internal basket assembly that holds the spent fuel assemblies. The HSM is a low-profile, reinforced concrete structure designed to withstand all normal condition loads, as well as abnormal condition loads created by natural phenomena such as earthquakes and tornadoes. It is also designed to withstand design basis accident conditions. The Standardized NUHOMS® Horizontal Modular Storage System has been approved for storage of spent fuel under the conditions of CoC No. 1004. The DSCs under consideration for exemption were loaded under CoC No. 1004, Amendment No. 10.

The NRC has previously approved the Standardized NUHOMS® Horizontal Modular Storage System. The requested exemption does not change the fundamental design, components, contents, or safety features of the storage system. The NRC staff has evaluated the applicable potential safety impacts of granting the exemption to assess the potential for danger to life or property or the common defense and security; the evaluation and resulting conclusions are presented below. The potential impacts identified for this exemption request were in the areas of materials, structural integrity, thermal, shielding, criticality, and confinement capability.

Materials Review for the Requested Exemption: The applicant asserted that there is a reasonable assurance of safety to grant the requested exemption to continue the storage of DSCs 11–15 in their respective HSMs. The applicant's assertion of reasonable assurance of safety is based on the following factors:

- Reasonable assurance of weld integrity;

- Low dose consequences for a DSC in storage; and
- Low risk to the public.

The applicant further stated that there is reasonable assurance of weld integrity based on the existing Quality Assurance (QA) documentation, engineering analysis, and expert evaluations, which demonstrate that the subject DSC welds possess sufficient quality to perform their design functions due to the following:

- Fuel cladding integrity is maintained, as no damaged fuel was loaded and no unexpected dose readings were observed during drying operations.
- The weld design assures that there are no pinhole leaks and there is no credible process for service-induced flaws.
- The material, including the DSC shell, lids and weld filler, met quality requirements and quality welds were ensured by welding process qualification, welder qualification and the use of an automated welding process specifically designed for the application.
- In-process visual inspections of welds performed by the welders, Quality Control (QC) visual examination (VT) inspections of fit-ups and welds, and the vacuum hold, helium pressure and helium leak test all ensured confinement and quality of the welds.
- Strain margins for the DSC welds were demonstrated by structural analysis assuming flaw distributions conservatively derived from the Phased Array Ultrasonic Testing (PAUT) examination of DSC 16.
- Based on the DSCs 11–15 site-specific heat load conditions, additional margin exists to account for any remaining flaw uncertainty.

The NRC materials review for the requested exemption focused on the applicant's assertion of reasonable assurance of weld integrity and each of the supporting assertions of: (1) fuel cladding integrity; (2) weld design; (3) material and welding process; (4) tests performed; (5)

adequate strain margins to accommodate flaws; and (6) additional strain margins in welds. A specific review of each of the supporting statements is provided in the following sections.

Fuel Cladding Integrity: The applicant provided information on the nature of the spent nuclear fuel in DSCs 11–15 to demonstrate that the fuel cladding fission product barrier is intact and any postulated canister weld leak would have an insignificant effect on radioactive release. At the time of loading in 2013, the applicant stated that the combined decay heat load in the limiting DSC did not exceed 10.96 kilowatts. In addition, only one of the 305 loaded fuel assemblies was considered to be high burnup, with a maximum recorded burnup of 45.12 gigawatt days per metric ton of uranium (GWD/MTU) (in DSC 15). The applicant stated that cask loading reports and supporting radiochemistry records indicate that all of the fuel assemblies loaded into DSCs 11–15 met the TS requirements (TS Table 1-1t) for cladding integrity and no damaged fuel was loaded. The applicant stated that the integrity of the fuel was further demonstrated by the fact that no unexpected dose rate readings were observed during the vacuum drying processes of DSCs 11–15.

The NRC staff reviewed the information provided by the applicant on the characteristics of the spent fuel loaded in DSCs 11–15. The NRC staff also reviewed the loading records for the loading campaign and confirmed that (1) no damaged fuel assemblies were loaded in the DSCs; (2) only one fuel assembly had burnup that marginally exceeded the 45 GWD/MTU criterion for high burnup fuel however, the cladding of the fuel assembly was shown to be intact through cask loading reports and supporting radiochemistry reports; and (3) no unexpected dose readings were observed in the loading campaign. Based on the review of the information from the loading campaign, the NRC staff confirmed that the characteristics of the fuel loaded in the DSCs included in the exemption request were accurately described.

Weld Design: The applicant stated that the updated final safety analysis report (UFSAR) only describes weld failure in terms of a possible pinhole leak in individual weld layers. The applicant further stated that the UFSAR assumes or stipulates that pinholes may exist in individual layers but the UFSAR makes no explicit mention about how a pinhole leak in a weld layer is formed, whether it occurs during the weld formation or by subsequent canister loading operations, fatigue cycles during storage, or accidents. The applicant stated that the existence of pinhole leaks is a non-mechanistic assumption of the UFSAR; and there is no underlying malfunction that causes its formation.

The applicant stated that, once in storage, there is no credible failure mechanism of the DSC top cover plate closure welds that would adversely affect DSC confinement because (1) the top cover plate and weld material are stainless steel and the only welds subject to the outside environment are the outer layer of the outer top cover plate (OTCP) weld and the test port plug (TPP) weld; (2) a reduction in cross section from plastic strain is not applicable to the top cover plate welds because the differential pressure across the top cover plates conditions is minimal (less than one atmosphere); and (3) the mechanism of cyclic loading is not applicable to the top cover plate and closure welds because the extent of fatigue cycling experienced by the canister is below the threshold which the ASME B&PV Code Section III has established.

The NRC staff have previously reviewed the design of the NUHOMS® 61BTH DSC included in the UFSAR. The NRC staff verified that the top cover plate and weld material are stainless steel and the only welds subject to the outside environment are the outer layer of the OTCP weld and the TPP weld. The NRC staff verified that the differential pressure across the top cover plates is minimal and consequently the reduction in cross section from plastic strain is not credible. The NRC staff have reviewed the assessment of fatigue and determined that the DSCs are not subjected to cyclic loading that requires a fatigue analysis. Based on the NRC staff's previous analysis of the DSC weld design, the NRC staff determined that the applicant's assessment of the weld design is accurate and there is no credible mechanism for the propagation of an existing weld flaw to result in a through weld thickness penetration that would result in a leak.

Material and Welding Process: The applicant stated that procurement records such as certified material test reports (CMTRs) demonstrate that the canisters, lids, and weld filler materials met design standards and quality requirements, thereby assuring compatibility between materials and satisfactory material performance characteristics (e.g., material strength).

The applicant stated that the weld closures of DSCs 11–15 were performed under a 10 CFR part 50 Appendix B QA program, such that the canister integrity is assured. The applicant stated that welding materials were procured to quality requirements, welding processes were developed and qualified for the given configuration, and welders were appropriately qualified to the ASME B&PV Code requirements. Finally, the applicant stated that welding parameters were specified in associated procedures and monitored as required.

In addition to the original weld head video review conducted in conjunction with the DSC 16 exemption request, the applicant included another examination of the weld head video and the general area videos taken during the 2013 cask loading campaign. Based on the examination of the videos, the applicant made a correlation between weld techniques and typical weld flaw characteristics such as those identified in the PAUT of the inner top cover plate (ITCP) and OTCP welds from DSC 16. The applicant provided an assessment conducted by Structural Integrity Associates, Inc. (SIA), which concluded that defects would be limited in the through thickness dimension to the thickness of a single bead. The applicant also stated that, even considering the possibility that any given layer of weld may have a leak through that layer, the licensing basis criterion stated in the UFSAR Section 3.3.2.1 assures that the chance of pinholes being in alignment on successive independently-deposited weld layers is not credible.

As stated above, the NRC staff have previously reviewed the design of the NUHOMS® 61BTH DSC included in the UFSAR. The NRC staff reviewed the materials used in the construction of DSCs 11–15 and the NRC staff confirmed that the materials used met the specifications called out in the NUHOMS® 61BTH DSC design. The NRC staff reviewed the

CMTRs and confirmed that the materials met specified compositional and mechanical property requirements.

The NRC staff reviewed, “TRIVIS Inc. Welding Procedure Specification (WPS) SS-8-M-TN, Revision 10,” (Enclosure 2 to RAI Response 1) which was used for the machine welding of the ITCP and the OTCP as well as, “TRIVIS Inc. WPS SS-8-A-TN, Revision 8,” (RAI Response 1 Enclosure 3) used for manual welding of the ITCP and the OTCP. The NRC staff compared WPS SS-8-M-TN, Revision 10 and WPS SS-8-A-TN, Revision 8 to the essential variables required for the gas tungsten arc welding (GTAW) in ASME Section IX Part QW Welding, Article II Welding Procedure Qualifications, Table QW-256 and Article IV Welding Data, Subsection QW-400 Variables. The NRC staff determined that the WPS SS-8-M-TN, Revision 10 and WPS SS-8-A-TN, Revision 8 are acceptable because all of the essential variables identified in ASME Section IX for GTAW WPSs were included and the range of permissible values were specified.

The NRC staff reviewed, “TRIVIS, Inc. Procedure Qualification Record (PQR) PQR-1, Revision 2” (Enclosure 4 to RAI Response 1). The NRC staff compared the testing documented in PQR-1, Revision 2 against ASME Section IX Part QW Welding, Article I Welding General Requirements. The NRC staff determined that PQR-1 Revision 2 was acceptable because all the testing necessary to qualify WPS SS-8-M-TN, Revision 10 and WPS SS-8-A-TN, Revision 8 were performed with satisfactory results and documented in PQR-1, Revision 2.

As documented in NUREG-1536, Revision 1, Section 8.9.1 (ADAMS Accession No. ML101040620) the NRC previously determined that for a multipass lid-to-shell weld of an austenitic stainless steel canister designed and fabricated in accordance with the ASME B&PV Code Section III Subsection NB (Class 1 components), no flaws of significant size will exist such that the flaws could impair the structural strength or confinement capability of the weld. For a spent nuclear fuel canister, such a flaw would be the result of improper fabrication or welding technique, as service-induced flaws under normal and off-normal conditions of storage are not credible.

The NRC staff notes that per the guidance in NUREG-1536, Revision 1, Section 8.4.7.4, the large structural lid-to-shell weld designs fabricated from austenitic materials may be tested using non-destructive examination methods such as a volumetric ultrasonic test (UT) or a multi-pass PT. If a multiple-pass PT examination is utilized in lieu of UT inspection, a stress reduction factor of 0.8 for weld strength is imposed. In the absence of valid PT examinations of the closure welds for DSCs 11−15, the applicant asserted that the helium leak rate tests performed on all DSCs and the PAUT results for DSC 16, which show that weld defects are limited to the height of one weld bead, support the claim that DSCs 11−15 do not have flaws that would impair the structural strength or confinement capability.

The NRC staff reviewed the information provided by the applicant including the DSC lid-to-shell closure weld design for the ITCP and the OTCP, the manual and machine GTAW WPSs, the helium leak testing results for DSCs 11−15 and the PAUT results for DSC 16. The NRC staff concluded that the design of the DSC closure weld and the GTAW WPSs used to weld the ITCP and the OTCP are unlikely to result in weld flaws that could impair the structural strength or confinement capability of the weld. The NRC staff concluded that the helium leak testing results for DSCs 11−15 confirmed that there were no flaws that impaired the confinement capability of the DSC 11−15 ITCP welds. The NRC staff concluded that the PAUT results for DSC 16 is sufficient to show that the GTAW of the ITCP and OTCP welds do not result in defects that would impair structural strength or confinement capability of the DSC closure welds.

Tests Performed: The applicant stated that a number of independent tests were conducted on the DSC 11–15 welds which verify that adequate welds were performed on DSCs 11–15. The applicant stated that these tests include:

- In-process visual examination and QC visual examinations to demonstrate that weld processes were followed and a weld meeting visual examination criteria was developed; and

- Helium leakage tests to verify the confinement integrity function and, to some extent, the structural integrity function of the DSC welds.

The applicant provided an extent of condition assessment as Appendix D of Enclosure 1 of the Exemption Request. The applicant stated that the extent of condition assessment was focused on:

- Compliance with welding administrative requirements;
- Technical specification required testing of welds; and
- Weld depth measurements for outer top cover plate welds.

The NRC staff reviewed the information provided in the application and confirmed that the applicant provided documentation that the welding administrative requirements were met, as follows: (1) welding procedures were available at the job site for welding operators to follow; (2) weld surface preparations were completed such that the weld surface was dry and free of oil, grease, weld spatter, rust, slag, sand, discontinuities, or other extraneous material; (3) weld crown height for the ITCP and vent/siphon port were verified; and (4) welds for the ITCP, OTCP and the vent and siphon ports were all verified.

The NRC staff reviewed the information provided in the application and confirmed that the applicant provided documentation for the TS required tests performed on DSCs 11–15. The NRC staff verified that the application included documentation showing that (1) hydrogen monitoring was properly performed while welding in accordance with TS 1.1.11; (2) pressure testing of the DSC shell to ITCP weld was conducted in accordance with TS 1.1.12.4; (3) two cycles of vacuum drying and verification were conducted at a vacuum less than 2.8 torr and were maintained for times longer than 30 minutes in accordance with TS 1.2.2; (4) the DSCs were backfilled with helium and to a pressure of 17.2 ± 1.0 psi for a time of at least 30 minutes in accordance with TS 1.2.3a; and (5) helium backfilling, pressure verification and leak testing were conducted in accordance with American National Standards Institute (ANSI) N14.5-1997 and leak rates less than 1.0×10^{-7} ref cubic centimeters/sec were documented for DSCs 11–15 in accordance with TS 1.2.4a.

The NRC staff confirmed that the weld depth measurements for the OTCP were conducted at four locations around the weld circumference. The NRC staff confirmed that the weld depth (dimension of the weld throat) measurements met the minimum requirements of 0.5 inches for the OTCP weld for DSCs 11–15.

Based on the review of the information provided by the applicant, the NRC staff determined that the required tests were performed on the ITCP and OTCP welds including in-process visual inspections of welds performed by the welders, VT of fit-ups and welds and the vacuum hold, as well as helium pressure and helium leak testing. The NRC staff determined that the applicant completed an adequate extent of condition assessment which showed that the welding of the ITCP and OTCP were conducted in accordance with welding administrative requirements, the required testing of welds were in compliance with technical specifications, and weld depth measurements for the OTCP met design requirements for the 61BTH DSC.

Adequate Strain Margins to Accommodate Flaws (Exemption Request Enclosures 2 through 5):

The applicant stated that strain margins for DSCs 11–15 were demonstrated by structural analysis using theoretically-bounding full-circumferential flaws and a structural analysis assuming flaw distributions conservatively derived from the PAUT examination of DSC 16. The applicant supported the analysis using:

- a review of weld head video for all available DSCs, general area video for all available DSCs, and welding records;
- the allowable flaw size evaluation in the ITCP closure weld for DSC 16; and
- the ITCP and OTCP closure weld flaw evaluation for a 61BTH DSC based on the DSC 16 PAUT results.

Based on the review of the videos, welding records and the PAUT examination of DSC 16, the applicant determined that the indications found on DSC 16 are representative of those that may be found on DSCs 11–15. Consequently, the applicant determined that the same bounding analyses performed for DSC 16 should provide for similar conservative results for the closure welds for DSCs 11–15. The applicant stated that for the OTCP, the original design basis calculations determined critical flaw sizes. The applicant stated that these design basis analyses determined for a 360° circumferential flaw, an allowable flaw depth of 0.19 inch and 0.29 inch could exist for surface connected and sub-surface flaws respectively. Finally, the applicant stated that the flaw sizes determined by these calculations bound any of the indications found on DSC 16 by PAUT of the OTCP weld.

For the ITCP weld of DSC 16, the applicant provided a calculation, AREVA Calculation 11042-0204, Revision 3, “Allowable Flaw Size Evaluation in the Inner Top Cover Plate Closure Weld for DSC #16” (Exemption Request Enclosure 4) that documents the critical flaw size based on the maximum radial stresses in the welds due to design loads. The applicant’s analysis calculated the critical flaw size for a weld size of 0.25 inch per the PAUT results for DSC 16, which showed that the distance between the weld root and crown at the canister wall for the DSC 16 ITCP lid weld ranged from 0.25 inch to 0.4 inch. The applicant determined that the critical flaw depth was 0.15 inch, which would exceed the typical weld layer thickness. The applicant noted that the measured weld size for the ITCP weld on DSC 16 was significantly larger than the design thickness of 3/16 inch (i.e., 0.188”). The applicant stated that all analyses for DSCs 11–15 were conducted using the design thickness of the weld. The applicant provided an analysis of the allowable flaw size for the DSC ITCP and OTCP using the weld design thickness which used the flaw sizes from the PAUT examination of DSC-16 (Exemption Request Enclosure 5, AREVA Calculation 11042-0205, Revision 3, “61BTH ITCP and OTCP Closure Weld Flaw Evaluation”).

The applicant stated that, as part of the original extent of condition review, weld head videos were reviewed by SIA in 2014. For DSCs 13 and 16, the review included video recordings of the ITCP root and cover weld layers and the OTCP tack, root, intermediate and cover weld layers. For DSCs 12, 14 and 15, the review included video recordings of the OTCP tack, root, intermediate and cover weld layers. The applicant stated that no weld head video was available for DSC 11. The DSC 16 outer closure weld was concluded to be the most vulnerable to potential defects because a greater frequency of irregular surface conditions was generated during welding.

The applicant stated that SIA performed further reviews of available weld head videos along with general area videos, welding records, and PAUT results for DSC 16 to identify any correlations between the welding processes used during the 2013 loading campaign and the flaws identified by the PAUT. The applicant stated that, by correlating indications to the particular welding methods used on all six canisters (including DSCs 11–15), a reasonable case was made that the types of indications found on DSC 16 are representative of those that may be found on DSCs 11–15.

For the OTCP, the applicant stated SIA concluded that the defects located within the weld deposit of DSC 16 are believed to be inter-bead lack of fusion formed at the interface between adjacent weld bead surfaces. The applicant stated that when the defects are present in the DSC OTCP closure weld, they would be found at the interfaces between weld beads. The applicant included a schematic showing the DSC OTCP weld bead placement and the position of the lack-of-fusion flaws, which were characterized as parallel and offset. The applicant stated that the possible locations where lack of fusion between the sides of adjacent weld beads could form in the DSC OTCP closure weld would result in defects that are not aligned and which would not extend beyond the thickness of one weld pass layer.

For the ITCP, the applicant stated SIA concluded that the locations of the flaws in DSC 16 indicate that they were related to sidewall lack of fusion. SIA also noted that the weld joint geometry, welding system, and welding setup for the ITCP of DSCs 11–15 had potential for forming defects on the sidewall like those identified in DSC 16. The applicant stated that, from the review, SIA concluded the other five canister ITCP closure welds were welded in a similar manner, using similar welding procedures, equipment, welding process, filler material, and welding operators and thus, it is reasonable to assume the other canister ITCP welds will have similar intermittent defects. In addition, the applicant stated that the vertical weld wall of the weld groove is inherent to a single bevel design, and because there is limited room to tilt the tungsten electrode towards the side wall (DSC shell), any lack-of-fusion defects that might form would likely be located on the vertical sidewall. The applicant concluded that the assumptions made for the ITCP closure weld bounding analysis in DSC 16 were considered reasonable for all ITCP canister closure welds.

The NRC staff reviewed the applicant's summary of the weld head video and general area videos. The NRC staff also reviewed the applicant's supporting analyses including:

- AREVA Calculation 11042-0204, Revision 3, "Allowable Flaw Size Evaluation in the Inner Top Cover Plate Closure Weld for DSC #16" (Exemption Request Enclosure 4);
- AREVA Calculation 11042-0205, Revision 3, "61BTH ITCP and OTCP Closure Weld Flaw Evaluation" (Exemption Request Enclosure 5);
- Structural Integrity Associates, Inc. Report 700388.401, Revision 1, "Evaluation of the Welds on DSC 11–15" (Exemption Request Enclosure 3);
- Structural Integrity Associates Inc. Report 1301415.403, Revision 2, "Assessment of Monticello Spent Fuel Canister Closure Plate Welds Based on Welding Video Records" dated May 22, 2014 (RAI Response 1 Enclosure 8);
- Structural Integrity Associates Inc. Report 1301415.402, Revision 0, "Review of TRIVIS Inc. Welding Procedures used for Field Welds on The Transnuclear NUHOMS® 61BTH Type 1 & 2 Transportable Canister for BWR Fuel" (RAI Response 1 Enclosure 9); and

- RAI Response 2.

The NRC staff determined that, because the same welding process, welding equipment, and welding procedures were used by the personnel that conducted the ITCP and OTCP welds in DSCs 11-16, it is reasonable to conclude, based on engineering judgement that the types of defects in DSC 16 are representative of those that may be in DSCs 11–15. The NRC staff determined that, because the DSCs 11-16 are the same design, were fabricated to the same specifications, and were subjected to the same tests, the analysis conducted for DSC 16 is also applicable to DSCs 11–15.

The NRC staff reviewed the applicant's analysis for the OTCP welds and the description of the OTCP welding based on weld head video described in Exemption Request Enclosure 3, Structural Integrity Associates, Inc. Report 700388.401, Revision 1, "Evaluation of the Welds on DSC 11–15,'' Appendix B, "Outer Top Cover Plate Closure Weld Bead Sequence (Based on VID Observations)" and Appendix C, "Tabulated Review of Available VIDS for Monticello DSC-12 thru DSC-16." The NRC staff also reviewed the information included from the review of the general area video records included in Appendix D of Exemption Request Enclosure 3, "Monticello DSC Video Inspection." The NRC staff determined that due to the OTCP weld joint design and welding process used in the OTCP closure weld, the likely significant welding defects in the OTCP weld would be lack of fusion between the weld beads or at the interface of the OTCP weld and the OTCP or the interface of the OTCP weld and the DSC shell. Given the geometry of the weld joint, the number of welding passes required to fill the weld joint, the position of each welding pass, and the requirement for in-process visual inspection of the weld after each pass, the NRC staff determined that it is unlikely that a connected lack-of-fusion defect greater than the thickness of one pass would be present. The NRC staff determined that any lack-of-fusion defects in the OTCP would not be aligned because of the weld joint geometry and the positioning of the weld passes required to fill the OTCP weld joint.

With respect to the ITCP welds, the NRC staff reviewed the applicant's analysis for the ITCP welds and the description of the ITCP welding based on weld head video described in Exemption Request Enclosure 3, Structural Integrity Associates, Inc. Report 700388.401, Revision 1, "Evaluation of the Welds on DSC 11–15." The NRC staff also reviewed the following appendices to Exemption Request Enclosure 3: Appendix A, "Inner Top Cover Plate Closure Weld Bead Sequence (Based on VID Observations)"; Appendix C, "Tabulated Review of Available VIDS for Monticello DSC-12 through DSC-16"; and Appendix D "Monticello DSC Video Inspection."

The NRC staff notes that it is unclear whether some of the observations in Exemption Request Enclosure 3, Appendix C were in conformance with Procedure 12751-MNGP-OPS-01, Revision 0, "Spent Fuel Cask Welding: 61BT/BTH NUHOMS® Canisters" (RAI Response 1 Enclosure 6). In particular, the NRC staff note that Exemption Request Enclosure 3, Appendix C indicated there were two instances of blow through of the root pass on the OTCP weld of DSC-12. Procedure 12751-MNGP-OPS-01, Revision 0 states such an event would be treated as a major repair with additional NDE and documentation. However, in RAI Response 2, the applicant indicated that these events were weld craters and were not weld root blow through events. While NRC staff was not able to resolve whether these actions taken by the welder were in conformance with the applicable procedure, it was apparent from Exemption Request Enclosure 3, Appendix C that corrective actions were taken to address the weld defects. In addition, the NRC staff determined that either a blow through of the root pass or a weld crater is a localized defect that would, in the worst case, compromise a small length of the root pass. As such, the NRC staff determined that the reported observation of a possible root blow through in two locations is bound by the assumed size of the OTCP welds defects in the flaw evaluation.

The NRC staff determined that for the ITCP weld joint design the likely significant welding defects would be lack of fusion at the interface of the ITCP weld and the ITCP or the interface of the ITCP weld and the DSC shell. Given the geometry of the weld joint, the number of welding passes required to fill the weld joint, the position of each welding pass, and the requirement for in-process visual inspection of the weld after each pass, the NRC staff determined that lack of fusion between the ITCP weld and the DSC shell is likely to be the most significant type of weld defect in this joint. The NRC staff determined that the positioning of the welding electrode necessary to weld the root pass would minimize the chances of a lack-of-fusion defect located at the interface of the ITCP weld and the ITCP. The NRC staff determined that the positioning of the welding electrode necessary to weld the second fill pass would minimize the chances of a lack-of-fusion defect at the interface of the ITCP weld and the DSC shell.

Based on the review of the information provided by the applicant including the review of weld head video for all available DSCs, general area video for all available DSCs, and welding records; the allowable flaw size evaluation in the ITCP closure weld for DSC 16; and the ITCP and OTCP closure weld flaw evaluation for a 61BTH DSC based on the DSC 16 PAUT results, the NRC staff concludes that the applicant has adequately considered the sizes and location of potential weld flaws to evaluate the stress margins in the ITCP and OTCP welds of DSCs 11–15. The NRC staff structural review for the requested exemption follows the materials review.

Additional Strain Margins in Welds (Exemption Request Enclosures 6 through 9): The applicant stated that additional analysis was performed to maximize the size of flaws present in locations consistent with the results of the DSC 16 PAUT to demonstrate substantial margin to account for potential flaw uncertainties. In addition, the applicant stated that DSCs 11–15 site-specific heat load conditions were applied to demonstrate additional weld margin exists and is available to account for any remaining flaw uncertainty. The applicant stated that the analysis used design basis loads with flaws present in locations consistent with the DSC 16 PAUT results and maximized in size such that the weld flaws approach acceptable design limits.

The applicant stated that the two maximum modeled weld flaws for OTCP to DSC shell weld are 0.43 inch and 0.42 inch in height, which represents about 85% through-wall of the 0.5-

inch minimum weld throat. The applicant stated that the maximum modeled full-circumferential weld flaws for ITCP to DSC shell weld are 0.11 inch in height at the ITCP weld to the ITCP interface and 0.14 inch in height at the ITCP weld to DSC shell interface, which represent respectively 58% and 74% through-wall of the 0.19-inch minimum weld throat. The applicant stated that each of the four assumed flaws represent defects spreading over more than one weld bead.

The NRC staff reviewed the applicant's analysis for the ITCP and OTCP weld flaws along with the applicant's summary of the welding video recordings and the PAUT examination results for DSC 16. For the ITCP weld, the NRC staff assessed the geometry of the weld joint, the positioning of the welding electrode in both the root and the final fill pass along with the requirement for in-process visual inspection of the weld after each pass. For the OTCP weld, the NRC staff assessed the geometry of the weld joint, the number of welding passes required to fill the weld joint, the position of each welding pass, along with the requirement for in-process visual inspection of the weld after each pass. The NRC staff determined that any lack-of-fusion defects in the ITCP and OTCP would not be aligned and would not result in a defect greater than the thickness of one pass given the weld joint geometry and the positioning of the weld passes required to fill the ITCP and OTCP weld joints. Thus, the NRC staff determined that the flaws assessed in Exemption Request Enclosure 6 are both unlikely to occur in any of the DSCs loaded in the 2013 campaign and the flaws assessed in Exemption Request Enclosure 6 conservatively bound any possible welding defects that are likely to exist in the DSC 11–15 OTCP welds.

Based on the review of the information provided by the applicant including the analysis of flaws analyzed from the PAUT examination of the ITCP and OTCP welds of DSC 16 and the assumed maximized flaws that exceed the weld bead deposit thickness, the NRC staff concludes that the applicant's analysis of stress margins in the ITCP and OTCP welds of DSCs 11–15 conservatively assumed weld flaws that are much larger than would be reasonably expected. This is due to the combination of the materials of construction, weld joint designs, and the welding process used for the ITCP and OTCP welds.

Structural Review for the Requested Exemption: The exemption request states that there is a reasonable assurance of safety to grant the requested exemption to continue the storage of DSCs in their respective HSMs. As noted by the applicant, one of the many factors contributing to this assertion is the structural integrity of the DSC top cover plates-to-shell closure welds. The *Structural Review* is based on the conclusion of the *Materials Review* where the NRC staff determined among other findings that, because the DSCs 11-16 are of the same design, were fabricated to the same specifications, and were subjected to the same tests, the analyses conducted for DSC 16 may also be applied to DSCs 11–15.

For the DSC 11–15 closure weld structural functions assessment, which was done by analysis, the applicant noted that the previous evaluations to demonstrate adequate strain margins of safety of the DSC 16 closure welds also support the current exemption request. These evaluations were provided in the following reports:

- SIA Report 1301415.301, Revision 0, “Development of an Analysis Based Stress Allowable Reduction Factor (SARF) – Dry Shielded Canister (DSC) Top Closure Weldments” (Exemption Request Enclosure 2);
- AREVA Calculation 11042-0204, Revision 3, “Allowable Flaw Size Evaluation in the Inner Top Cover Plate Closure Weld for DSC #16” (Exemption Request Enclosure 4); and
- AREVA Calculation 11042-0205, Revision 3, “61BTH ITCP and OTCP Closure Weld Flaw Evaluation” (Exemption Request Enclosure 5).

The evaluations performed on the DSC 16 closure welds included: (1) a structural analysis using an analysis-based stress allowance reduction factor and theoretically-bounding full-circumferential flaws to demonstrate that finite element analysis (FEA) simulation is suitable for analyzing the structural performance of the weld as a continuum with multiple embedded flaws; (2) a calculation that documents the allowable critical flaw size in the ITCP closure weld based on the maximum design basis radial stresses in the welds; and (3) a structural analysis demonstrating large weld strain margins of safety with conservative assumptions of flaw distribution and size derived from the DSC 16 PAUT examination results.

However, to demonstrate adequate strain margin and to accommodate flaws in the DSCs 11–15 closure welds, the applicant provides a FEA simulation evaluation in SIA Report, 700388.401, Revision 1, "Evaluation of the Welds on DSCs 11–15," (Exemption Request Enclosure 3) to support that the flaw distribution and size based on the PAUT examination results for the DSC 16 closure weld performance can be used to conservatively represent the closure weld flaws for DSCs 11–15. As noted in the *Materials Review*, the NRC staff reviewed the applicant's evaluation and determined that the flaws used in analyzing the DSC 16 closure welds are a reasonable representation for the closure welds for all DSCs 11-16. This finding provides the basis for the NRC staff to review the two calculation packages: Calculations 11042-0207 and 11042-0208, which used the maximized weld flaws that are essentially the same in distribution but are much larger in size than those used for the DSC 16 evaluation.

Specifically, in Calculation 11042-0207, the applicant asserts that there are adequate strain margins in the welds to accommodate flaws for DSCs 11–15. The DSCs are subject to the design basis temperature, pressure, and side-drop loading conditions and are analyzed per the ASME Code Section III criteria, using the limit load and elastic-plastic analyses. In Calculation 11042-0208, the applicant asserts additional strain margin in the DSCs 11–15 closure welds. The maximum flaws, the analysis methodology and the evaluation criteria are the same as those of Calculation 11042-0207. However, in lieu of the design basis loading, the analysis used the as-loaded DSC cavity pressure, which is site-specific and temperature dependent. The at-temperature material yield strengths are used, which are higher than those associated with the design basis loading.

It is noted that the exemption request also included Calculation 11042-0209 (Exemption Request Enclosure 8) to demonstrate additional weld strain margin for DSCs 11–15 subject to the site-specific side-drop loading condition. The NRC staff neither approves, nor rejects, and is not expressing any view related to the material in the calculation, as it did not enter into the NRC evaluation.

The NRC staff reviewed the above two calculation reports on the structural performance of the DSC 11–15 closure welds. In Calculation 11042-0207, the applicant followed the same analysis method used in Calculation 11042-0205 for DSC 16 to demonstrate adequate strain margin in DSCs 11–15 closure welds. The applicant noted that the finite element model details and structural performance acceptance criteria are the same except that the maximized flaw configuration is postulated to result in much larger flaws than those associated with DSC 16 to provide additional insights into the weld structural performance.

To arrive at the maximized configuration, the flaws modeled in Calculation 11042-0205 for DSC 16 were first modified slightly, including replacing conservatively the 0.11 inch-long flaw inside the ITCP with an equivalent-height flaw at the interface between the ITCP and the 3/16-inch ITCP-to-shell weld. However, the size and location of all other welds were unchanged. Next, an elastic-plastic analysis of flaw length introduced increasingly larger flaw sizes in each analysis iteration to simulate higher localized plastic strain.   As noted by the applicant, the iteration analysis was considered complete for the maximized flaws determination for which the peak equivalent plastic strain for the most critically stressed flaws would be calculated to be somewhat below the ASME code weld material elongation limit of 28 percent. The applicant performed the elastic-plastic iteration analysis using a 150-percent design basis side-drop of 112.5 g (75 × 1.5 = 112.5) to arrive at the maximized flaws. Specifically, the maximized, 360^o full-circumferential flaws are of 0.43 inch and 0.42 inch in height for the two flaws associated with the OTCP, which represent about 85% through-wall of the 0.5-inch minimum throat for OTCP-to-DSC shell weld. The maximized full-circumferential flaws for ITCP-to-DSC shell weld are 0.11 inch and 0.14 inch each in height, which represent respectively 58% and 74% through-wall of the 0.19-inch minimum weld throat. The NRC staff reviewed the iteration analysis for arriving at the maximized flaws for the DSCs 11–15 closure welds. Because the maximized flaws are essentially the same in locations as those used for DSC 16 and the resulting flaw sizes are much larger than the corresponding ones used for DSC 16, the NRC staff concludes that the postulated maximized flaws are conservative and appropriate for evaluating the strain performance of the DSCs 11–15 closure welds.

Using the maximized flaws, the applicant performed limit load analyses in Calculation 11042-0207 for two DSC design basis internal pressures of 32 psi and 65 psi for the ASME Code Service Level A/B and Service Level D evaluations, respectively. The analyses resulted in the calculated collapse pressures of 86.3 psi for Service Level A/B and 122.2 psi for Service Level D. The collapse pressures are acceptable because they are greater than the respective ASME Code limit-load analysis acceptance criteria of 60 psi and 90.2 psi. Similarly, for the design basis DSC side-drop of 75 g, the applicant used the 3D half-symmetric model to perform a Service Level D limit load analysis. The applicant determined the side-drop collapse load to be approximately 179.5 g, which includes an off-normal DSC design basis internal pressure of 20 psi as a boundary condition. This determination is acceptable because the collapse load is greater than the required side-drop load of 104 g to satisfy the ASME Code limit-load analysis acceptance criteria.

To address the potential material rupture associated with high plastic strain concentrations at the weld flaws, the applicant performed elastic-plastic analyses in Calculation 11042-0207 to quantify strain margins of safety for the DSCs 11–15 with maximized flaws. This concern was addressed by considering a Ramberg-Osgood idealization of the stress-strain curve for SA-240 Type 301 stainless steel, which recognizes strain hardening effects for the FEA modeling. The elastic-plastic analyses resulted in the peak equivalent plastic strains of 7.4 percent and 11.1 percent for the Service Level D design basis pressure of 65 psi and side-drop of 75 g, respectively. For the strain margin evaluation, the applicant continued to use the same DSC 16 weld strain acceptance criterion of not exceeding the 28 percent elongation limit, which is a reduction from the ASME B&PV Code specified weld elongation limit of 35 percent by a factor of 0.8 ($0.35 \times 0.8 = 0.28$). Considering the 28 percent elongation limit, the strain margins of safety corresponding to the calculated peak equivalent plastic strains are 2.78 $\{(0.28/0.074) – 1 = 2.78\}$ and 1.52 $\{(0.28/0.111) – 1 = 1.52\}$, respectively. Because the margins of safety are all positive (i.e. greater than zero), the NRC staff concludes that there are adequate strain margins in the welds to accommodate flaws for DSCs 11–15.

Additionally, similar to the analysis used to supplement qualification of the DSC 16 closure welds, the applicant considered a 150 percent of the design basis loading to evaluate the DSCs 11–15 welds. The analysis used a DSC internal pressure of 100 psi ($65 \times 1.5 = 97.5 < 100$ psi) and a side-drop of 112.5 g ($75 \times 1.5 = 112.5$ g), which are beyond the ASME B&PV Code, Section III, Paragraph NB-3228.3 Plastic Analysis provisions. The calculated peak equivalent plastic strains are 13.6 percent and 23.0 percent for the respective pressure and side-drop loading cases. For the weld strain margin evaluation, the applicant continued to use the same 28 percent weld elongation limit which resulted in the weld strain margins of safety of 1.06 $\{(0.28/0.0136) – 1 = 1.06\}$ and 0.22 $\{(0.28/0.23) – 1 = 0.22\}$, respectively. Because all margins of safety are positive, even in loading conditions that are 50 percent beyond those required for evaluating localized strains by the elastic-plastic analysis, the NRC staff concludes that there are adequate strain margins on the welds to accommodate flaws for DSCs 11–15.

The applicant noted that there are additional strain margins in the closure welds of DSCs 11–15 owing to the site-specific as-loaded temperature and DSC internal pressure conditions at MNGP, which are less severe than those associated with the design basis conditions. In Calculation 11042-0208 (Exemption Request Enclosure 7), the applicant performed evaluations using the temperature and pressure conditions specific to DSCs 11–15. The evaluation follows the same Calculation 11042-0207 analysis method and acceptance criteria, including the same maximized flaws. The applicant indicated that the evaluations were intended to address any remaining uncertainties related to potential flaws that may be present in DSCs 11–15 by demonstrating existence of additional strain margins in the closure welds.

Using the site-specific 370^{o}F at-temperature material yield strength of 21.2 ksi for the SA-240 Type 304 stainless steel, the applicant determined the Service Level D limit load collapse pressure is 144.1 psi. This pressure is significantly higher than the DSC at-temperature internal pressure of 45.9 psi and the ASME Code limit-load collapse pressure acceptance criteria of 90.2 psi. Correspondingly, using the site-specific 237^{o}F at-temperature material yield strength of 24.0 ksi, together with the off-normal at-temperature internal pressure of 10.9 psi as a boundary condition, the applicant determined the collapse side-drop g-load to be 204 g. This site-specific collapse side-drop is also much greater than the ASME Code limit-load collapse side-drop g-load acceptance criteria of 104 g associated with the design basis 500^{o}F at-temperature material yield strength of 19.4 ksi.

To determine the strain margins of safety for the site-specific temperature and pressure, the applicant performed elastic-plastic analyses for DSCs 11–15 with the maximized flaws in the OTCP- and ITOP-to-shell welds. Using the analysis approach in Calculation 11042-0207, the applicant calculated the peak equivalent plastic strains of 4.4 percent and 9.8 percent for the Service Level D internal pressure of 45.9 psi and the design basis side-drop of 75 g, respectively. For the same weld elongation limit of 28 percent, the corresponding strain margins of safety are calculated to be 5.36 {(0.28/0.044) – 1 = 5.36} and 1.86 {(0.28/0.098) – 1 = 1.86}. Similar to the analysis used in Calculation 11042-0207 for a supplement qualification of the DSC 16 closure welds with a more conservative loading assumption, the applicant also considered 150 percent of the site-specific loading to evaluate the weld flaws using a DSC internal pressure of 69 psi (45.9 × 1.5 = 69 psi) and side-drop load of 112.5 g. The resulting peak equivalent plastic strains are 7.1 percent and 19.0 percent, which correspond to the strain margins of safety of 2.94 {(0.28/0.071) – 1 = 2.94} and 0.47 {(0.28/0.19) – 1 = 0.47}, respectively. For the

MNGP site-specific evaluation, because the margins of safety are all positive, the NRC staff concludes that the DSCs 11–15 weld strains have additional margins beyond the design basis conditions.

On the basis of the review above, the NRC staff concludes that the limit load and elastic-plastic analysis results showed that the welds would undergo localized plastic deformation. The applicant's evaluation indicated that no weld material rupture or breach of the DSCs 11–15 confinement boundary at the closure welds is expected because of the adequate margins of safety against the weld elongation limits. For this reason, the NRC staff has reasonable assurance to conclude that the ITCP and OTCP welds of DSCs 11–15 have adequate structural margins of safety for the ASME Code Service Level D design criteria, which bound the normal, off-normal, and accident (including natural phenomenon) conditions for the subject weld structural integrity evaluation. The NRC staff also finds that the retrievability of DSCs 11–15 is ensured based on the demonstration of adequate weld strain margins of safety discussed above.

Thermal Review for the Requested Exemption: The applicant stated that even though nonconforming examinations exist for the primary confinement welds, satisfactory completion of the required helium leak test conducted on DSCs 11–15 has demonstrated the integrity of the primary confinement boundary (ITCP and siphon/vent cover plate) welds. These tests specifically demonstrated that the primary confinement boundary field welds are "leak tight" as defined in ANSI N14.5-1997. The applicant stated that, in this respect, the helium leak test demonstrated the basic integrity of the primary confinement boundary and the lack of a through-weld flaw in the field closure welds that would lead to a loss of cavity helium in DSCs 11–15. The applicant stated that the field closure welds indirectly support the thermal design function by virtue of their confinement function (as demonstrated by the helium leak test conducted on DSCs 11–15) which assures the helium atmosphere in the DSCs 11–15 cavity is maintained in order to support heat transfer. The applicant also stated that the satisfactory completion of two required vacuum pump-downs conducted on the DSCs demonstrated weld integrity of the ITCP confinement boundary. These pump-downs establish a differential pressure across the ITCP and siphon/vent block welds of approximately one atmosphere, which exceeds the magnitude of the 10 psig design pressure used in stress analyses for normal conditions. Although the vacuum pump-down imparts a pressure differential in a reverse direction from the confinement function, according to the applicant, the pump-down demonstrates the basic function of the confinement boundary and the lack of a through-weld flaw in the ITCP and siphon/vent block welds sufficient to cause a loss of cavity helium when in service.

The NRC staff reviewed the applicant's exemption request and also evaluated its effect on DSCs 11–15 thermal performance. The NRC staff concludes that the cask thermal performance is not affected by the exemption request because the applicant has shown that a satisfactory helium leak test was conducted on DSCs 11–15, which is integral to ensuring integrity of the primary confinement boundary. Integrity of the primary confinement boundary assures the spent fuel is stored in a safe inert environment with unaffected heat transfer characteristics that assure peak cladding temperatures remain below allowable limits. The NRC staff also concludes that the applicant demonstrated the lack of a through-weld flaw in the ITCP and siphon/vent block weld sufficient to cause a loss of cavity helium. This satisfies 10 CFR 72.236(f) which requires that the cask be designed to have adequate heat removal capacity without active cooling systems and 10 CFR 72.122(h) which states that the fuel cladding during storage must be protected against degradation and gross rupture. Therefore, based on the NRC staff's review of the applicant's evaluation and technical justification, the NRC staff finds the exemption request acceptable by virtue of the demonstrable structural integrity of the ITCP and siphon/vent plate welds.

The NRC staff finds that the thermal function of DSCs 11–15, loaded under CoC No. 1004, Amendment No. 10, addressed in the exemption request remains in compliance with 10 CFR part 72.

Shielding and Criticality Safety Review for the Requested Exemption: The NRC staff reviewed the criticality safety and radiation protection effectiveness of DSCs 11–15 presented in the applicant's exemption request. The NRC staff finds that the criticality safety and radiation protection of DSCs 11–15 are not affected by the nonconforming PT examinations for the following reasons: (1) the interior of DSCs 11–15 will continue to prevent water in-leakage which means that the system will remain subcritical under all conditions; and (2) the nonconforming PT examinations do not affect the radiation source term of the spent fuel contents, or the configuration and effectiveness of the shielding components of the Standardized NUHOMS® system containing the 61BTH DSC, meaning that the radiation protection performance of the system is not altered.

The NRC staff finds that the criticality safety and shielding function of DSCs 11–15, loaded under CoC No. 1004, Amendment No. 10, addressed in the exemption request remains in compliance with 10 CFR part 72.

Confinement Review for the Requested Exemption: The objective of the confinement evaluation was to confirm that DSCs 11 through 15 loaded at the MNGP met the confinement-related requirements described in 10 CFR part 72. NRC staff relied on the information provided by the applicant in their Exemption Request dated October 18, 2017.

As described in the applicant's "Exemption Request for Nonconforming Dry Shielded Canister Dye Penetrant Examinations" (Exemption Request Enclosure 1), certain elements of the DSCs 11–15 closure weld PT examinations did not comply with examination procedures associated with TS 1.2.5. To support the exemption request, the applicant noted that a helium leakage rate test of the closure's confinement boundary, including ITCP weld, siphon cover plate weld, and vent port cover plate weld, were conducted per TS 1.2.4a and demonstrated that the primary confinement barrier field welds met the TS acceptance criterion of leaktight as defined by ANSI N14.5-1997. The applicant noted that the confinement integrity is not affected by the non-compliant PT examination procedures. The NRC staff concludes that not performing the PT examination procedures relevant to this exemption request would not change the results of the helium leakage test, which is integral to ensuring closure confinement integrity, and therefore, the closure confinement integrity is unaffected. The structural and material acceptability of DSCs 11 through 15 welds is discussed in the *Structural Review* and the *Materials Review* described previously.

It is noted that a dose-related analysis was included as Enclosure 10 of the Exemption Request. NRC staff neither approves, nor rejects, and is not expressing any view related to the material in that enclosure, as it did not enter into the evaluation.

Risk Assessment for the Requested Exemption: In support of the applicant's request, the applicant submitted a risk assessment, Jensen Hughes Report 016045-RPT-01, "Risk Assessment of MNGP DSCs 11–15 Welds Using NUREG-1864 Methodology" (Exemption Request Enclosure 11). The risk assessment compares the calculated risk of leaving the five DSCs in storage "as is" at the MNGP ISFSI versus transferring the DSCs back into the reactor building to perform PAUT of the welds and then returning them to their storage locations. The risk for each potential accident, regardless of likelihood, can be generally summarized by the following equation:

$$\begin{array}{c} \text{Initiating} \\ \text{Event} \\ \text{Frequency} \\ \text{(Per Year)} \end{array} \times \begin{array}{c} \text{Probability} \\ \text{of Canister} \\ \text{Release} \end{array} \times \begin{array}{c} \text{Probability of} \\ \text{Containment} \\ \text{Release} \end{array} \times \begin{array}{c} \text{Consequences} \\ \text{(Cancer} \\ \text{Fatality)} \end{array} = \text{Risk}$$

The process to transfer a DSC to the reactor building refueling floor for PAUT incurs added potential for accidental drops due to the lifting and subsequent lowering operations. For 20-year storage, the risk is the sum of all potential accident risks for the duration. Each DSC handling operation is independent. For five canisters, the total risk value is multiplied by five.

NUREG-1864, "A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant" (ADAMS Accession No. ML071340012) provides guidance for assessing the risk to the public and for identifying the dominant contributors to risk for performing probabilistic risk assessments (PRAs) of a dry cask storage system located at a nuclear power plant site. NUREG-1864 documents a pilot PRA conducted for a dry cask storage system (Holtec International HI-STORM 100) at a Boiling Water Reactor (BWR) Mark 1 plant. The risk assessment estimated the annual off-site risk for one cask in terms of individual probability of a prompt fatality and a latent cancer fatality. It does not consider risk to workers or future off-site transportation of DSCs.

The applicant applied the methodology and results in NUREG-1864 to perform the risk assessment. The risk assessment compared the NUHOMS® and HI-STORM-100 dry spent fuel storage systems and determined the designs are similar with a few basic differences. Both storage systems include canisters for confining dry spent fuel. The canisters have similar design and dimensions and are made of stainless steel of similar thickness and are required to meet the same ASME class (ASME B&PV, Section III, and Subsection NB). The HI-STORM 100 system consists of a multipurpose canister (MPC) that confines spent fuel assemblies, a transfer overpack that provides shielding during canister preparation, and a vertical, cylindrical storage overpack that provides shielding during long-term storage.

Both MNGP and Hatch (the plant selected for the Pilot PRA) are BWR, Mark 1 plants; therefore, the storage systems are exposed to similar handling hazards. The potential drop heights for loaded TCs moving across the refueling floor, or lowering from the height of refueling floor to the ground floor of the equipment hatch are very similar. The potential impact surfaces are also similar.

The NUHOMS® system is comprised of a DSC, a TC, and an HSM. A transfer trailer is used to move the loaded TC. Two key differences exist between the NUHOMS® and the HI-STORM dry spent fuel storage operations. First, the NUHOMS® TC is placed horizontally on the transfer trailer and is not subject to accidental drops when moving between the ISFSI and fuel building. Second, transferring NUHOMS® DSC between the TC and the HSM is done horizontally; thus, the NUHOMS® DSC is not subject to any potential vertical drop. During storage on an ISFSI pad, the horizontal-storage design of the HSM eliminates the risk of tip over caused by seismic activities or wind-driven missiles. Aircraft impact on the HSM is limited to only large aircrafts and the methodology considered the distance to local airfields and planes that operate in the area. The NUREG-1864 frequency estimate for meteorite strikes per unit area is used in this assessment, and the analysis is adjusted for the larger horizontal surface area of the HSM.

In the risk assessment, the potential radiological consequences are based on a comparison of the spent fuel in the MNGP DSC and the spent fuel modeled in NUREG-1864. In NUREG-1864, the HI-STORM 100 MPC contained 68 BWR fuel assemblies with 10-year-old high-burnup (50 GWD/MTU) fuel. The MNGP NUHOMS® DSC contains 61 BWR fuel assemblies with 15.5-year-old fuel of 41 GWD/MTU (not high burnup) fuel. The plume heat content for a cask release is estimated to be that of the spent fuel. NUREG-1864 estimates the maximum decay heat load to be 264 watts per assembly. The estimated maximum decay heat load for MNGP DSC is approximately 220 watts per assembly. The risk assessment analysis assumes that the source term from NUREG-1864 adequately represents or bounds those of the MNGP configuration. The NRC staff agrees that this is reasonable based on the applicant's assessment which shows NUREG-1864 radionuclide inventory is 7.0 times higher than that of MNGP DSC.

The NUREG-1864 evaluation of misload concluded MPC integrity would not be affected unless a gross series of errors occurred. The errors would have to result in nearly every fuel assembly loaded into the MPC being incorrect and insufficiently cooled. NUREG-1864 concluded this gross misload scenario was not credible. Therefore, the risk assessment did not explore risk from misloading of spent fuel.

The applicant's risk assessment assumes the annual risk for a DSC while stored on the ISFSI would be the same for both alternatives. The risk assessment identified three types of mechanical failure that could cause significant radiological releases to the environment: drop accidents, meteorite strikes, and overflight aircraft accidents. The primary difference in risk between the two alternatives, continued storage at the ISFSI versus moving a DSC back to the spent fuel pool area for PAUT, are potential drop accidents during lifting and lowering of a DSC between the ground floor and the height of the refueling floor.

The applicant's risk assessment accounted for possible added risk from a potential flaw around the canister lid by assuming the probability of lid failure would be same as for the DSC shell in drop accidents. This assumption doubles the estimated probability for a release from drop accidents. Strain analysis in NUREG-1864 reports the most highly stressed regions of the MPC for a drop accident are in areas near the base of the cylindrical shell and in the weld joining the shell to the baseplate. Since the top side of a canister is not expected to experience significant strain, the NRC staff agrees that the assumption is conservative and bounds the probability of a release occurring following a drop accident.

The NRC staff reviewed the applicant's risk assessment and agrees the mechanical failures identified and the radiological inventory from NUREG-1864 would be bounding for each of the MNGP DSCs. The risk assessment concludes that the risks are significantly lower than the level considered "negligible" by the Quantitative Health Guidelines (QHG) established in "Risk-Informed Decisionmaking for Nuclear Material and Waste Applications," Revision 1 (ADAMS Accession No. ML080720238). The QHG considers public individual risk of latent cancer fatality risk of less than 2×10^{-6} per year as negligible. The pilot PRA (NUREG-1864) concluded that there is no prompt fatality risk, and the calculated risk is extremely small. NUREG-1864 reports the increase in risk (individual probability of latent cancer fatality) from the first year as 1.8×10^{-12}, and for subsequent years as 3.2×10^{-14} per year per MPC. The total risk for Monticello as calculated by Jensen Hughes took into account the characteristics of the spent fuel and the site, as well as the differences between the MNGP and Hatch ISFSIs. For the five DSCs over a period of 20-year storage, risk would be: Alternative 1, continue storage as-is, Risk = 1.4×10^{-12}; Alternative 2, move DSCs back up to the refueling floor for PAUT then return to storage location, Risk = 2.3×10^{-12}; with a difference in risk between the two proposed alternatives of 9.3×10^{-13}.

The assessment of difference in risk between the proposed alternatives was performed based on evaluation data from NUREG-1864. The MNGP off-site consequence is based on individual risk and not absolute population difference. Based on the considerations taken into account for the difference between the NUREG-1864 MPC and the MNGP DSCs in this assessment, the NRC staff finds the risk assessment calculation to be reasonable because the applicant used accepted methods and the site-specific considerations were addressed in an appropriately conservative manner.

The purpose of this assessment is to compare the risk associated with leaving these DSCs as-is at the ISFSI versus transferring the five DSCs back to the refueling floor for PAUT, and then returning them to the ISFSI for storage. The process of returning the five DSCs to the refueling floor for PAUT incurs additional crane operation. The inadvertent drop frequency for heavy loads (NUREG-1774, "A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002", ADAMS Accession No. ML032060160) is $5.6X10^{-5}$/lift. The probability of release from a DSC drop accident, assuming defective weld, is 4.0×10^{-2}. This operation occurs inside a closed building with probability of release value of 1.5×10^{-4}. The consequence value for a release is 3.6×10^{-4}. The risk for a drop while lifting a DSC up to the refueling floor can be calculated as:

$$(5.6 \times 10^{-5})(4.0 \times 10^{-2})(1.5 \times 10^{-4})(3.6 \times 10^{-4}) = 1.2 \times 10^{-13} \text{ cancer fatality/year}$$

The risk for a drop while lowering a DSC (assuming no weld flaw, probability of release is $2.0X10^{-2}$) through the equipment hatch back to ground level can be calculated as:

$$(5.6 \times 10^{-5})(2.0 \times 10^{-2})(1.5 \times 10^{-4})(3.6 \times 10^{-4}) = 6.0 \times 10^{-14} \text{ cancer fatality/year}$$

The additional risk from performing PAUT for five DSCs would be five times the sum of risk for lifting and lowering one DSC.

$$5 \times [(1.2 \times 10^{-13}) + (6.0 \times 10^{-14})] = 9.3 \times 10^{-13} \text{ cancer fatality/year}$$

Probabilistic risk assessments are typically used to evaluate risks greater than 1.0×10^{-6}. In light of the calculated risk values, the NRC staff finds the off-site risk as too small to be accurately discernable. Based on the discussion presented above, the NRC staff concludes that risk to the public for the two options provided by Jensen Hughes, “continued storage as-is” and “transfer, perform PAUT, and return to storage,” are essentially equivalent.

Otherwise in the Public Interest

In considering whether granting the exemption is in the public interest, the NRC staff considered the alternative of not granting the exemption. If the exemption were not granted, in order to comply with the CoC, either (1) DSCs 11–15 would have to be removed from their respective HSMs, opened and unloaded, and the contents loaded in new DSCs, with each of those new DSCs welded and tested, or (2) removed from the HSMs to allow access to the OTCP to be machined off, and the ITCP weld machined down to the root weld; and each DSC, ITCP and OTCP inspected to determine if there was any damage as a result of the machining (which would then necessitate the actions detailed in option 1); or (3) conduct PAUT by opening the HSMs to conduct in-situ testing (which is limited to less than 360° of the weld circumference) or transferring to a TC for testing on the ISFSI pad or in the reactor building (essentially Alternative 2 in the *Risk Assessment*). Options 1 and 2 would entail a higher risk of cask handling accidents, additional personnel exposure, and greater cost to the applicant. As noted above in the *Risk Assessment*, Option 3 does not increase the risk by a discernible amount. All options would generate additional radioactive contaminated material and waste from operations. For options 1 and 2, the lid would have to be removed, which would generate cuttings from removing the weld material that could require disposal as contaminated material. For option 3, radioactive wastes would be generated from radioactively contaminated consumables and anti-contamination clothing used during the examination. Also, radioactive waste would be generated from the cleanup of any coupling fluid (of the PAUT) that it combines with and then transports resulting in contamination from the surface of the DSC. This radioactive waste would be transported and ultimately disposed of at a qualified low-level radioactive waste disposal facility, potentially exposing it to the environment.

The proposed exemption to permit continued storage of DSCs 11–15 in their respective HSMs for the service life of the canisters at the MNGP ISFSI is consistent with NRC's mission to protect public health and safety. Approving the requested exemption reduces the opportunity for a release of radioactive material compared to the alternatives to the proposed action, because there will be no operations involving the opening of the DSCs, which confine the spent nuclear fuel, and there will be no operations involving the opening of the HSMs potentially exposing radioactive waste to the environment. Therefore, the exemption is in the public interest.

Environmental Consideration

The NRC staff also considered in the review of this exemption request whether there would be any significant environmental impacts associated with the exemption. The NRC staff determined that this proposed action fits a category of actions that do not require an environmental assessment or environmental impact statement. Specifically, the exemption meets the categorical exclusion in 10 CFR 51.22(c)(25).

Granting this exemption from 10 CFR 72.212(a)(2), 72.212(b)(3), 72.212(b)(5)(i), 72.214, and 72.212(b)(11) only relieves the applicant from the inspection or surveillance requirements associated with performing PT examinations with regard to meeting TS 1.2.5 of Attachment A of CoC No. 1004. A categorical exclusion for inspection or surveillance requirements is provided under 10 CFR 51.22(c)(25)(vi)(C) if the criteria in 10 CFR 51.22(c)(25)(i)-(v) are also satisfied. In its review of the exemption request, the NRC staff determined, as discussed above, that, under 10 CFR 51.22(c)(25): (i) granting the exemption does not involve a significant hazards considerations because granting the exemption neither reduces a margin of safety, creates a new or different kind of accident from any accident previously evaluated, nor significantly increases either the probability or consequences of an accident previously evaluated; (ii) granting the exemption would not produce a significant change in either the types or amounts of any effluents that may be released offsite because the requested exemption neither changes the effluents nor produces additional avenues of effluent release; (iii) granting the exemption would not result in a significant increase in either occupational radiation exposure or public radiation exposure, because the requested exemption neither introduces new radiological hazards nor increases existing radiological hazards; (iv) granting the exemption would not result in a significant construction impact, because there are no construction activities associated with the requested exemption; and; (v) granting the exemption would not increase either the potential or consequences from radiological accidents such as a gross leak from the closure welds, because the exemption neither reduces the ability of the closure welds to confine radioactive material nor creates new accident precursors at the MNGP ISFSI. Accordingly, this exemption meets the criteria for a categorical exclusion in 10 CFR 51.22(c)(25)(vi)(C).

IV. Availability of Documents

The documents identified in the following table are available to interested persons through one or more of the following methods, as indicated.

DOCUMENT	ADAMS ACCESSION NO.
Federal Register Notice Issuing Exemption from Nonconforming Dye Penetrant Examinations of Dry Shielded Canister (DSC) 16, June 8, 2016	ML16159A227
Exemption Request for Nonconforming Dye Penetrant Examinations of Dry Shielded Canisters (DSCs) 11 through 15, October 18, 2017	ML17296A205
First Request for Additional Information for Review of Exemption Request for Five Nonconforming Dry Shielded Canisters 11 through 15 (CAC No. 001028, Docket No. 72-58, EPID L-2017-LLE-0029), March 6, 2018	ML18065A545
Monticello Nuclear Generating Plant - Response to Request for Additional Information Regarding Exemption Request for Nonconforming Dye Penetrant Examinations of Dry Shielded Canisters (DSCs) 11 through 15, April 5, 2018	ML18100A173

Supplement to Exemption Request for Nonconforming Dye Penetrant Examinations of Dry Shielded Canisters (DSCs) 11 through 15 (CAC No. 001028, EPID L-2017-LLE-0029)	ML18151A870
NUREG-1774, “A Survey of Crane Operating Experience at U.S. Nuclear Power Plants from 1968 through 2002”	ML032060160
Risk-Informed Decisionmaking for Nuclear Material and Waste Applications, Revision 1	ML080720238
NUREG-1536, Revision 1 “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility”	ML101040620
NUREG-1864, “A Pilot Probabilistic Risk Assessment of a Dry Cask Storage System at a Nuclear Power Plant”	ML071340012
Attachment A, Technical Specifications, Transnuclear, Inc., Standardized NUHOMS® Horizontal Modular Storage System Certificate of Compliance No. 1004, Renewed Amendment No. 10, Revision 1	ML17338A114

V. Conclusion

Based on the foregoing considerations, the NRC staff has determined that, pursuant to 10 CFR 72.7, the exemption is authorized by law, will not endanger life or property or the common defense and security, and is otherwise in the public interest. Therefore, the NRC grants the applicant an exemption from the requirements of 10 CFR 72.212(a)(2), 72.212(b)(3), 72.212(b)(5)(i), 72.212(b)(11), and 72.214 only with regard to meeting TS 1.2.5 of Attachment A of CoC No. 1004 for DSCs 11–15.

This exemption is effective upon issuance.

Dated at Rockville, Maryland, this 13th day September, 2018.

For the Nuclear Regulatory Commission.

John McKirgan,

Branch Chief,

Spent Fuel Licensing Branch,

Division of Spent Fuel Management,

Office of Nuclear Material Safety and Safeguards.

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