ATTACHMENT A

CONDITIONS FOR CASK USE AND
TECHNICAL SPECIFICATIONS
DOCKET NO. 72-1007
CERTIFICATE OF COMPLIANCE NO. 1007
AMENDMENT 1

CONTENTS

TABLES

FIGURES

REVISION HISTORY

	Section	Change Description
Amendment 1	1.1	Changed text to reflect terminology in COC.
	1.1.2	Deleted specific name of valve manufacturer.
	1.1.7	Clarified requirement that first MSB shall be loaded with 24 spent fuel assemblies constituting a heat source of up to 24kW.
	1.2.1	Changed to include storage of BPRAs and clarified conditions for fuel loading.
	1.2.6	Added curve for spent fuel pool boron concentrations when loading B&W, Mark B, 15X15 fuel with BPRAs with initial enrichments over 3.3 weight percent uranium-235.
	1.2.8	Corrected typographical error.
	1.2.15	Reduced maximum lift height from 80 to 60 inches to correspond with the analyzed values provided in SNC calculations and corrected typographical errors.

1.0 INTRODUCTION

This section presents the conditions that a potential user (licensee) of the Ventilated Storage Cask (VSC-24) system must comply with, in order to use the system under a general license issued according to the provisions of 10 CFR 72.210 and 72.212. These conditions have either been proposed by the system vendor, imposed by the Nuclear Regulatory Commission staff as a result of the review of the Safety Analysis Report (SAR), or are part of the regulatory requirements expressed in 10 CFR 72.212.

1.1 General Requirements and Conditions for Cask Use

1.1.1 Regulatory Requirements

Regulatory requirements define a number of technical and administrative conditions for system use. Technical regulatory requirements for the licensee (user of the VSC-24 system) are contained in 10 CFR 72.212(b).

10 CFR 72.212(b) requires that the licensee perform written evaluations, before use, that establish that: (1) conditions set forth in the Certificate of Compliance have been met; (2) cask storage paths and areas have been designed to adequately support the static load of the stored casks; and (3) the requirements of 10 CFR 72.104, "Criteria for radioactive materials in effluents and direct radiation from an ISFSI or MRS," have been met. It also requires that the licensee review the SAR and the associated SER, before use of the general license, to determine whether or not the reactor site parameters (including earthquake intensity and tornado missiles), are encompassed in the cask design bases considered in these reports.

Site-specific parameters and analyses, identified in the SER, that need verification by the system user, are as follows:

1. The temperature of 75^{o} F as the maximum average yearly temperature, without solar incidence. (Reference SER Section 2.5);

2. The steady state temperature extremes of 100^{o} F, (average daily temperature) with incident solar radiation, and -40^{o} F, with no solar incidence. (Reference SER Section 2.5);

3. The "accident" short-term temperature extreme of 125^{0} F with incident solar radiation. (Reference SER Section 2.5);

4. The horizontal and vertical seismic acceleration levels of 0.25g and 0.17g, respectively. (Reference SER Section 2.5);

5. The analyzed flood condition of 25 fps water velocity and full submergence of the loaded ventilated concrete cask (VCC). (Reference SER Section 2.5); and

6. The potential for fire and explosion should be addressed, based on site-specific considerations. (Reference SER Section 2.6).

According to 10 CFR 72.212(b), a record of the written evaluations must be retained by the licensee until spent fuel is no longer stored under the general license issued under 10 CFR 72.210.

1.1.2 Operating Procedures

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The operating procedures suggested generically in the SAR are considered appropriate, as discussed in Section 11.0 of the SER, and should provide the basis for the user's written operating procedures. The following additional written procedures shall also be developed as part of the user operating procedures:

1. A procedure shall be developed for cask unloading, assuming damaged fuel. If fuel needs to be removed from the multi-assembly sealed basket (MSB), either at the end of service life or for inspection after an accident, precautions must be taken against the potential for the presence of oxidized fuel and to prevent radiological exposure to personnel during this operation. This activity can be achieved by using the valves to determine the atmosphere within the MSB before the removal of the structural and shield lids. If the atmosphere within

the MSB is helium, then operations should proceed normally, with fuel removal, either via the transfer cask or in the pool. However, if air is present within the MSB, then appropriate filters should be in place to permit the flushing of any potential airborne radioactive particulate from the MSB, via the valves. This action will protect both personnel and the operations area from potential contamination. For the accident case, personnel protection in the form of respirators or supplied air should be considered in accordance with the licensee's Radiation Protection Program.

2. A procedure shall be developed for the documentation of the characterizations performed to select spent fuel to be stored in the MSB. This procedure shall include a requirement for independent verification of each fuel assembly selection.

3. A procedure shall be developed for two independent determinations (two samples analyzed by different individuals) of the boron concentration in the water of the spent fuel pool and that used to fill the MSB cavity.

4. In preparing written operating procedures for handling the MSB over the VCC, the user shall include a consideration for reducing the likelihood of fracturing the ceramic tiles at the bottom of the VCC, as the MSB is lowered into position.

1.1.3 Quality Assurance

Activities at the independent spent fuel storage installation (ISFSI) shall be conducted in accordance with the requirements of 10 CFR Part 50 , Appendix B.

1.1.4 Heavy Loads Requirements

Lifts of the MSB in the multi-assembly transfer cask (MTC) must be made within the existing heavy loads requirements and procedures of the licensed nuclear power plant. The MTC design has been reviewed under 10 CFR Part 72 and found to meet NUREG-0612 and ANSI 14.6. However, an additional safety review (under 10 CFR 50.59) is required to show operational compliance with NUREG-0612 and/or existing plant-specific heavy loads requirements. Other spent fuel transfer systems, for loading the MSB and VCC within reactor fuel buildings, may be suitable for use in accordance with 10 CFR Part 50 operating licenses.

1.1.5 Training Module

A training module shall be developed for the existing licensee's training program, establishing an ISFSI training and certification program. This module shall include the following:

1. VSC-24 Design (overview);
2. ISFSI Facility Design (overview);
3. Certificate of Compliance Conditions (overview);
4. Fuel Loading, MTC Handling, MSB Lowering Procedures; and
5. Off-Normal Event Procedures.

1.1.6 Training Exercise

A dry run of the MSB loading, MTC handling, and MSB lowering shall be held. This dry run shall include, but not be limited to, the following:

1. Moving an MSB and MTC into and out of the pool;
2. Loading a fuel assembly;
3. MSB sealing and cover gas backfilling operations (using a mock-up MSB);
4. Lowering the MSB into the concrete cask;
5. Returning the MSB to the fuel pool; and
6. Opening an MSB (using a mock-up MSB).

1.1.7 Requirement for First Cask in Place

Based on the conclusions of SER Section 12.0, the following measurements are required for the first VSC placed in service:

The first MSB shall be loaded with 24 spent fuel assemblies, constituting a heat source of up to 24 kW, and then the MSB shall be loaded into the VCC to measure the cask thermal performance by measuring the air inlet and outlet temperatures for normal air flow, according to the specification in Section 1.2.3. The purpose of the test is to measure the heat removal performance of the VSC system and establish base-line data (SAR Section 9.1.3). A letter report summarizing the results of the test and evaluation shall be submitted to the NRC within 30 days of placing the cask in service in accordance with 10 CFR 72.4.

Should the first user of the system not have spent fuel capable of producing a 24 kW heat load, the user may use a lesser load for the test, provided that a calculation of the temperature difference between the inlet and outlet temperatures is performed, using the same methodology and inputs documented in the SER and SAR, with the lesser load as the only exception. The calculation and the measured temperature data shall be reported to the NRC in accordance with 10 CFR 72.4. The calculation and comparison need not be reported for casks that are subsequently loaded with lesser heat sources than the test case. However, for the first or any other user, the process needs to be reported for any higher heat sources, up to 24 kW, which is the maximum allowed under this Certificate of Compliance. NRC will also accept the use of artificial thermal loads other than spent fuel, to satisfy the above requirement.

1.1.8 Surveillance Requirements Applicability

The specified frequency for each Surveillance Requirement is met if the surveillance is performed within 1.25 times the interval specified in the frequency, as measured from the previous performance.

For frequencies specified as "once," the above interval extension does not apply.

If a required action requires performance of a surveillance or its completion time requires periodic performance of "once per...," the above frequency extension applies to the repetitive portion, but not to the initial portion of the completion time.

Exceptions to these requirements are stated in the individual specifications.

1.2 Technical Specifications, Functional and Operational Limits

1.2.1 Fuel Specification

Limit/Specification:

The characteristics of the spent fuel allowed to be stored in the VSC-24 system are restricted to those included in Table 1.

Applicability: The specification is applicable to all fuel to be stored in the VSC-24 system.

Objective: The specification is prepared to ensure that the peak fuel rod temperatures, maximum surface doses, and nuclear criticality effective neutron multiplication factor are below the design values. Furthermore, the fuel weight and type ensures that structural conditions in the SAR bound those of the actual fuel being stored.

Action: Each spent fuel assembly to be loaded into an MSB shall have the parameters listed in Table 1 independently verified and documented. Fuel not meeting this specification shall not be stored in the VSC-24 system.

Surveillance: Immediately before insertion of a spent fuel assembly into an MSB, the identity of each fuel assembly shall be independently verified and documented.

Basis: The specification is based on consideration of the design basis parameters included in the SAR and limitations imposed as a result of the staff review. Such parameters stem from the type of fuel analyzed, physical and structural limitations, criteria for criticality safety, criteria for heat removal, and criteria for radiological protection. The VSC-24 system is designed for dry, vertical storage of irradiated pressurized water reactor (PWR) fuel. The principal design parameters of the fuel to be stored are found in SAR Section 2.1. The VSC-24 cask can accommodate standard PWR fuel assembly classes of the designs manufactured by Combustion Engineering (CE), Exxon, Westinghouse, and Babcock and Wilcox (B&W), including burnable poison rod assemblies (BPRAs) in B&W 15X15 fuel assemblies, that are listed in Table 1, and is limited to fuel assemblies bounded by these standard designs. The analyses presented in the SAR are based on non-consolidated, zircaloy-clad fuel, with no known or suspected gross cladding failures.

The physical parameters that define the mechanical and structural design of the VCC and the MSB are the fuel assembly dimensions and weight provided in SAR Table 2.1-1. They represent the heaviest fuel, so that the calculated stresses bound the PWR fuel designs to be stored.

The design basis for nuclear criticality safety is based on the most reactive fuel assemblies, of the types listed, with initial enrichments up to 4.2 wt. percent ^{235}U (see SER Section 7.0). The criticality design criteria ensure that the k_{eff} remain subcritical under normal, off-normal, and accident conditions assuming the loading of unirradiated fuel.

Primary protection against accidental criticality is provided by operational procedures to prevent the introduction of water into the cask not containing the minimum specified concentration of dissolved boron (see Technical Specification 1.2.6). Technical Specification 1.2.6 requires that, prior to the introduction of water into the MSB, two water samples be taken and chemically analyzed by two individuals to independently verify the boron concentration in the water. The likelihood of one test failing to detect the correct boron concentration is small and the probability of two independent tests failing, concurrently, is highly unlikely. A secondary level of protection is provided by the burned nature of the permitted fuel as described below.

The secondary level of protection is provided by loading only fuel with a reactivity of less than or equal to 1.35 wt. percent ^{235}U unirradiated fuel. Combinations of initial enrichment and fuel irradiation that satisfy these criteria are shown in Figure 1. This Fuel Criticality Acceptance Curve represents the minimum burnup required for fuel assemblies of various initial enrichments to achieve a reactivity equivalent to an enrichment less than or equal to 1.35 weight percent ^{235}U. For irradiated fuel assemblies with an initial enrichment and burnup that fall above the curve, K_{eff} will be less than 0.95 when flooded with fresh water. Therefore, the fuel assemblies are acceptable for storage in the VSC-24 system, provided they also comply with the other characteristics of the Table 1 specifications. Irradiated fuel assemblies with initial enrichment and burnup that fall below the curve are unacceptable for loading in the VSC-24 cask.

The thermal design criterion of the fuel to be stored is that the maximum heat generation rate per assembly be such that the fuel cladding temperature is maintained within established criteria during normal and off-normal conditions. Fuel cladding temperature criteria were established by the applicant based on methodology in PNL-6189 and PNL-6364 (SAR References 1.1 and 4.1), which was described in Appendix 7 of the SAR. Based on this methodology, the staff has accepted that a maximum heat generation rate of 1 kW per assembly is a bounding value for the PWR fuel to be stored.

The radiological design criterion is that the gamma and neutron source strength of the irradiated fuel assemblies must be bounded by values of the neutron and gamma ray source strengths used by the vendor in the shielding analysis.

The design basis source strengths were derived from a burnup analysis of 3.2 weight percent ^{235}U enriched fuel, irradiated to an average fuel burnup of 35,000 MWd/MTU, and a post-irradiation time of 5 years.

The criteria provided in Table 1 applies to the fuel assembly and the BPRAs, in the case of the B&W, Mark B, 15X15 fuel. For example, the determination of the decay power per assembly includes the fuel and the BPRAs.

In the case of the B&W 15X15 fuel assembly, the total gamma source strength from the fuel assembly and BPRA is less than the specified gamma source strength limit. Furthermore, the gamma spectrum from the BPRA source regions is bounded by the corresponding gamma spectrum from the fuel assembly. The combined decay power from the B&W fuel with BPRAs inserted is less than the specified limit for decay power per assembly. The B&W 15X15 fuel with BPRAs also represents the heaviest assembly weight.

BPRAs with cladding failures are acceptable for loading into the VSC-24 system. Analyses of the B&W 15X15 BPRAs, with cladding failures, loaded in the MSB were used to determine the minimum required soluble boron concentration to ensure a k_{eff} of less than 0.95. These criticality analyses conservatively modeled the presence of the individual rods of the BPRAs as guide tubes filled with depleted B_4C (i.e., $(B_{11})_4C$). Thus, the highly borated moderator between the rods of the BPRA and the guide tubes was displaced by $(B_{11})_4C$, which has a positive net effect on reactivity. A failed BPRA loaded in the VSC-24 system would be depressurized and present a lower MSB accident pressure than that of an intact BPRA. Any release from a failed BPRA would not have an adverse effect on the internals of the MSB or the fuel assemblies stored in the MSB.

Table 1

Characteristics of Spent Fuel to Be Stored in the VSC-24 System

Fuel	Only intact, unconsolidated PWR fuel assemblies meeting the requirements listed below
Class/Type	B&W, Mark B, 15 x 15, with and without BPRAs CE/Exxon 15 x 15 CE 16 x 16 Westinghouse PWR 17x 17 Westinghouse PWR 15 x 15 Westinghouse PWR 14 x 14
Fuel Cladding	Zircaloy clad fuel with no known or suspected gross cladding failures
Decay Power Per Assembly	≤ 1 kW
Maximum Burnup	≤ 51,800 MWd/MTU (1)
Minimum Burnup	Determined by the fuel criticality acceptance curve shown in Figure 1.
Post Irradiation Time	≥ 5 years
Maximum Initial Enrichment	≤ 4.2 weight percent ^{235}U
Gamma Source	6.8×10^{15} photons/sec-assembly, with spectrum bounded by Table 5.2-1 of the SAR
Neutron Source	1.2×10^{8} neutrons/sec-assembly, with spectrum bounded by Table 5.2-2 of the SAR
Assembly Weight	≤ 1576 lb (716 kg)
Number of Assemblies per VSC	24

Notes:

(1) For casks loaded with fuel assemblies having burnups greater than 35,000 MWd/MTU, specific analyses must be performed, as described in SAR Section 2.1, to demonstrate that initial fuel clad temperature criteria are not exceeded and that neutron and gamma source strengths do not exceed the specified criteria tabulated above.

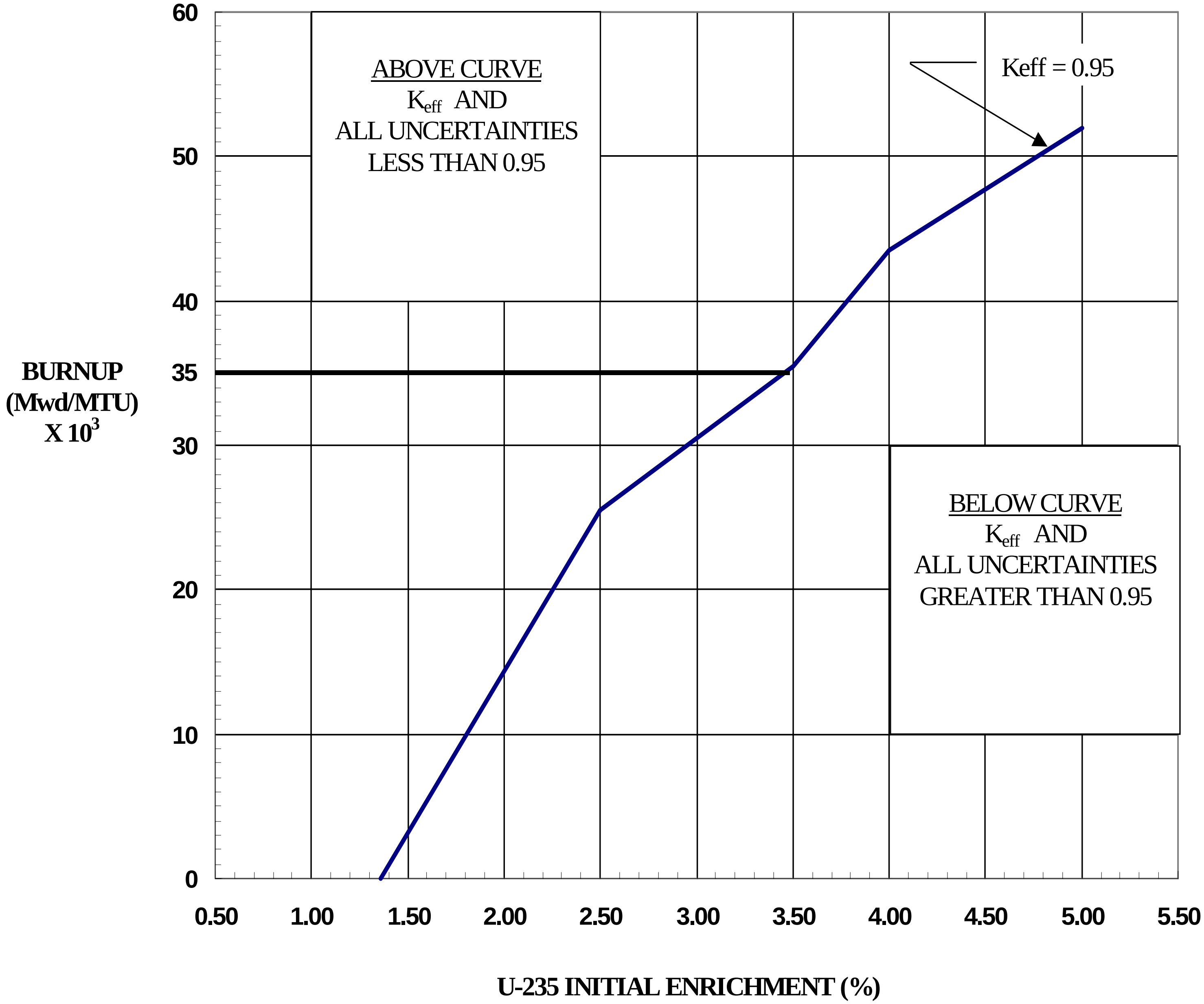


FIGURE 1

FUEL CRITICALITY ACCEPTABLITY CURVE

1.2.2 Maximum Permissible MSB Leak Rate

Limit/Specification:

$\leq$1.0 x 10^{-4} standard cubic centimeters per second (scc/sec) at 0.5 atm differential pressure.

Applicability: MSB inner seal confinement boundary.

Objective:

1. To limit the total radioactive doses normally released by each cask to negligible levels. Should fission gases escape the fuel cladding, they will remain confined by the MSB confinement boundary.

2. To retain helium cover gases within the MSB and prevent oxygen from entering the MSB. The helium improves the heat dissipation characteristics of the VSC and prevents any oxidation of fuel cladding.

Action: The leak rate shall be checked using calibrated instruments and written procedures. Procedures should be prepared to ANSI N14.5 (standard for leak testing of shipping cask) or equivalent. If the leak rate exceeds 10^{-4} scc/sec, the leak point must be found and repaired. The confinement boundary of the MSB itself may be easily repaired, since the field welding is performed only on the MSB outer surface.

Surveillance: The MSB shall be tested after the inner seal weld has been completed. The MSB will be pressurized with helium to 1.5 atm and a hand-held sniffer may be used (per manufacturer's instructions) to determine a leak rate. If the rate is within the limit, additional testing and surveillance are not required, since there are no normal or accident conditions that will breach the structural integrity and leak tightness of the MSB.

Basis: If the MSB leaked at the largest undetectable leak rate (10^{-4} scc/sec), then only 1 percent of the helium would escape over a 20-year span. This amount would be negligible.

1.2.3 Maximum Permissible Air Outlet Temperature

Limit/Specification:

The equilibrium air temperature at the outlet of a fully loaded VSC (24 kW) shall not exceed ambient by more than 110^0 F.

Applicability: This temperature limit applies to all VSCs stored in the ISFSI. If a cask is placed in service with a heat load less than 24 kW, the limiting temperature difference between outlet and ambient shall be determined by a calculation performed by the user using the same methodology and inputs documented in the SER.

Objective: The objective of this limit is to ensure that the temperatures of the fuel cladding and the VSC concrete do not exceed the temperatures calculated in Section 4.0 of the SAR. That section shows that if the air temperature increase (for 24kW) is below 110^0 F (expected to be 89^0 F for normal operation), the fuel cladding and concrete will be below both their temperature criteria for normal operation and the maximum heat load transient (125^0 F ambient, full solar and full thermal load). An additional objective of the temperature measurements is to confirm the thermal performance of the cask and provide base-line data.

Action: If an air temperature rise of greater than 110^0 F, or greater than predicted, is observed for any VSC placed in service, the first action should be to check all inlet and outlet ducts for airflow blockage. If environmental factors can be ruled out as the cause of the excessive cask temperatures, this condition indicates that the fuel assemblies may be producing heat at a rate higher than specified in Section 2.0 of the SAR. If fuel assemblies meeting the fuel specification in Section 1.2.1 have been loaded into the cask and the temperature difference is greater than 110^0 F, or that predicted for less heat loads, then this condition is not addressed in the SAR and will require additional measurements and analysis to determine that the actual performance of the cask is within the limits analyzed in the SAR. If the excessive temperatures cause the cask to perform in an unacceptable manner, or the temperatures cannot be controlled to within acceptable criteria, the cask shall be unloaded and a letter report shall be submitted to NRC within 30 days.

Surveillance: The ambient temperature and cask outlet air temperatures for the first VSC shall be measured and recorded daily for a period of 1 week after the VSC has been placed in service. The ambient temperature and cask outlet temperatures for the rest of the VSCs shall be measured and recorded upon placement in service and at intervals not to exceed 48 hrs until the cask has reached thermal equilibrium. After reaching thermal equilibrium, thermal performance of each cask shall be verified on a daily basis in accordance with specification 1.3.4.

Basis: If the air temperature rise is 110^0 F [21^0 F more than the 89^0 F rise calculated for the 75^0 F ambient case (SAR Table 4.1-1)], the maximum concrete and fuel cladding temperatures can be expected to be less than 21^0 F hotter than predicted (due to the non-linearity of radiation heat transfer). For a cask load of 24 kW, this condition would result in a maximum concrete temperature of 207^0 F and a maximum cladding temperature of 708^0 F. Both of these values are below the acceptable criteria (225^0 F for concrete and 712^0 F for 5-year cooled fuel).

1.2.4 Maximum External Surface Dose Rate

Limit/Specification:

The external surface dose rate from all types of radiation will be less than 20 mrem per hour on the sides and 50 mrem/hr on the top. Dose rates at the air inlets or outlets will be below 50 mrem/hr.

Applicability: This dose rate limit shall apply to the entire external surface of the VCC, except the bottom surface.

Objective: The external dose rate is limited to this value, to ensure that the cask has not been inadvertently loaded with fuel not meeting the specifications in Section 2.0 of the SAR, and to provide verification for plant personnel that radiation levels are acceptably low.

Action: If the measured dose rates are above those values listed above, correct fuel loading shall be verified. If correct fuel is loaded, specific analyses must demonstrate compliance with 10 CFR Part 20 and 10 CFR Part 72 radiation protection requirements, or appropriate action must be taken to comply with the acceptable limits. A letter report, summarizing the action taken and the results of investigation conducted to determine the cause of the high dose rates, shall be submitted to the NRC within 30 days. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to the administrator of the appropriate NRC regional office.

Surveillance: The external surface dose rate shall be measured after loading the MSB in the VCC and before transfer to the storage pad. The side dose rate shall be measured at a distance of 5 feet from the bottom of the VCC and at four equally spaced radial locations. The top dose rate shall be measured at the VCC lid center and the VCC outer lid edge. The dose rate measurement shall account for the effects of background radiation on the absolute dose rate measurements.

Basis: The basis for this limit is the shielding analysis presented in Section 5.0 of the SAR.

1.2.5 Maximum MSB Removable Surface Contamination

Limit/Specification:

10^{-4} µCi/cm^2 gamma-beta

10^{-5}µCi/cm^2 alpha

Applicability: MSB external surface.

Objective: Keep removable surface contamination level low enough so that offsite doses will be below 1 mrem, even in the event that contamination became loose and behaved as a particulate or gaseous release.

Action: If the limit is exceeded, the MSB exterior shall be washed by flushing the MSB-MTC gap with water, or other suitable decontamination solution, and additional contamination surveys taken until the limit is met.

Surveillance: Contamination surveys shall be taken on the MSB exterior, within 6 inches of the top of the MSB. Contamination surveys shall be taken on the MTC interior and bottom exterior surfaces after the MSB has been transferred to the VCC. The contamination surveys for removable surface contamination shall be conducted after the loaded MSB is removed from the pool and before the VSC is moved to the storage pad.

Basis: If the MSB were covered over its entire surface with 2.1 x 10^7 dpm/cm^2 (9.5 µCi/cm^2) of Co-60, and all the contamination became loose and were released as a gaseous particulate cloud under the worse meteorological conditions, the dose at 200 meters would be less than 1 mrem. This basis and the analysis are presented in Section 11.1.4 of the SAR. Therefore, using 10^{-4} µCi/cm^2 is a conservative limit, and ensures that the offsite dose limits in 10 CFR Parts 20, 50 (Appendix I), and 72 can be met. Significant amounts of residual contamination on the MTC surfaces above the specification are an indication that the MSB was not thoroughly decontaminated.

1.2.6 Boron Concentration in the MSB Cavity Water

Limit/Specification:

The MSB cavity shall be filled only with water having a boron concentration equal to, or greater than, 2850 ppm. The higher concentration of either 2850 ppm or that specified in Figure 2 shall be used when loading or unloading MSBs with B&W, Mark B, 15X15 fuel containing BPRAs.

Applicability: This specification is applicable to the loading and unloading of all MSBs.

Objective: To ensure a subcritical configuration is maintained in the case of accidental loading of the MSB with unirradiated fuel.

Action: If the boron concentration is below the required weight percentage concentration (gm boron/10^6 gm water), add boron and re-sample, and test the concentration until the boron concentration is shown to be greater than that required.

Surveillance: Written procedures shall be used to independently determine (two samples analyzed by different individuals) the boron concentration in the water used in the spent fuel pool and that used to fill the MSB cavity.

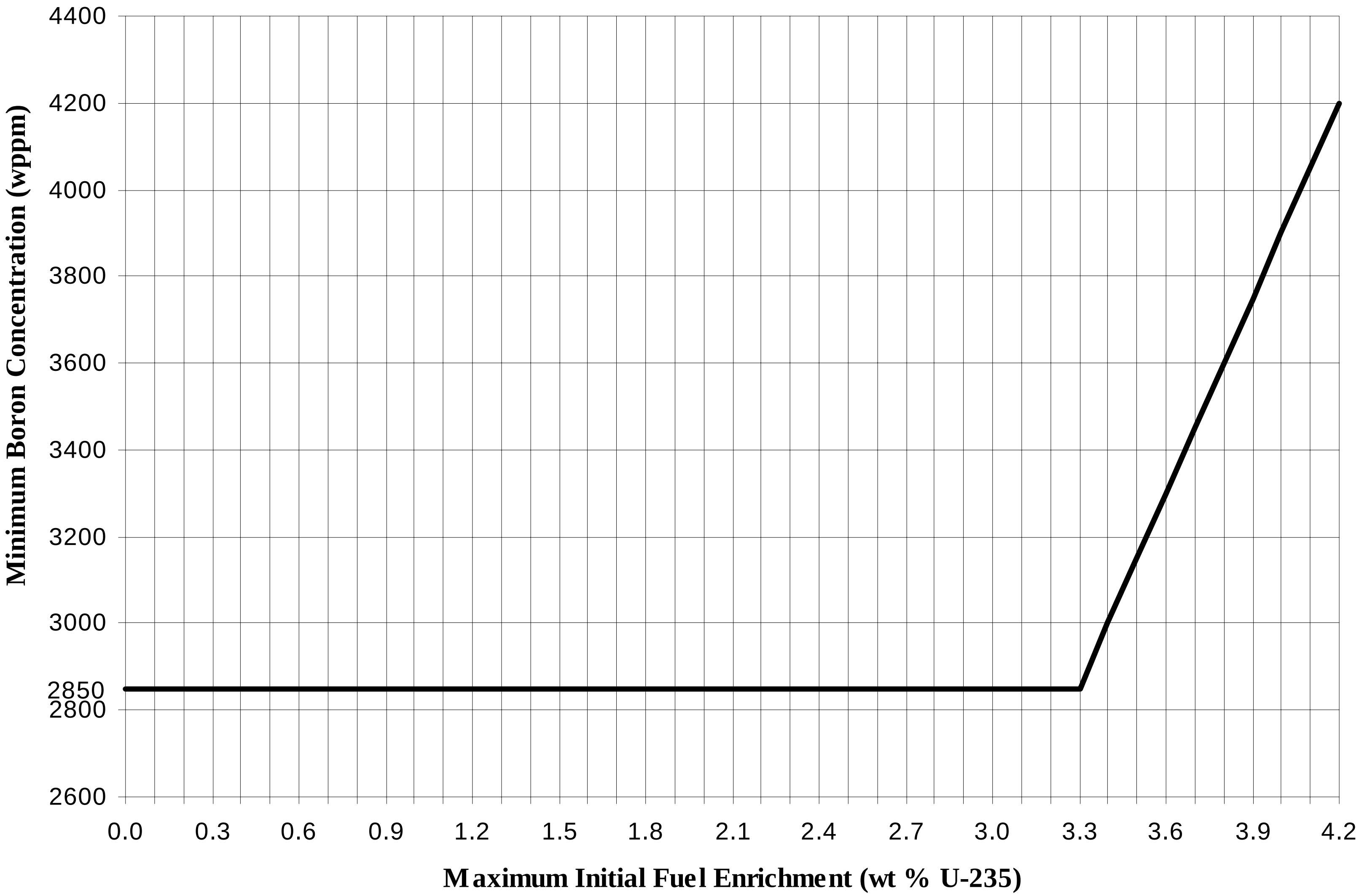
1. Within 4 hours before insertion of the first fuel assembly into the MSB, the dissolved boron concentration in water in the spent fuel pool and in the water that will be introduced into the MSB cavity shall be independently determined (two samples chemically analyzed by two individuals).

2. Within 4 hours before flooding the MSB cavity for unloading the fuel assemblies, the dissolved boron concentration in water in the spent fuel pool and in the water that will be introduced into the MSB cavity shall be independently determined (two samples chemically analyzed by two individuals).

3. The dissolved boron concentration shall be reconfirmed at intervals not to exceed 48 hours until such time as the MSB is removed from the spent fuel pool or the fuel is removed from the MSB.

Basis: The required boron concentration is based on the criticality analysis for an accidental misloading of the MSB with unburned fuel, maximum enrichment, and optimum moderation conditions.

Figure 2
Boron Concentration Curve



1.2.7 MSB Vacuum Pressure During Drying

Limit/Specification:

Vacuum Pressure:	$\leq$3 mm Hg
Time at Pressure:	$\geq$30 min.
Number of Pump-Downs:	2

Applicability: This specification is applicable to all MSBs.

Objective: To ensure a minimum water content.

Action: Once the required vacuum pressure specification is obtained, perform helium backfill to 22.1 psia ± 0.5 psia, MSB He leak test, then repeat evacuation.

If the required vacuum pressure cannot be obtained:

1. Check and repair, or replace, the vacuum pump;
2. Check and repair the vacuum tubing as necessary; and
3. Check and repair the weld between the MSB structural lid and the outer shell and the fillet weld between the structural lid and the shield plug.

Surveillance: No maintenance or tests are required during normal storage. Surveillance of the vacuum gauge is required during the vacuum drying operation.

Basis: The value of 3 mm Hg for absolute pressure was selected to allow the use of standard vacuum pumps. If the only gas contained within the MSB cavity is considered to be super-heated steam at a pressure 3 mm Hg and 450^{0}F, the moisture content of the MSB cavity is approximately 0.729 moles (assuming a perfect gas) and, hence, only 0.364 moles of O_2 are available (if 100 percent radiolysis is assumed). This O_2 could react with 1.09 moles of UO_2 (295 grams). However, the reaction of 295 grams of UO_2 would be negligible, compared to the 2225 grams of UO_2 in a single rod. Therefore, oxidation of 295 grams does not represent a threat to the safe operation of the VSC system.

However, since a multiple pump-down is performed, the O_2 partial pressure after backfilling with helium will not exceed (760) x $(3/760)^2$ = 0.01 mm Hg.

1.2.8 MSB Helium Backfill Pressure

Limit/Specifications:

Helium 14.5 psia ± 0.5 psia backfill pressure (stable for 30 minutes after filling).

Applicability: This specification is applicable to all MSBs.

Objective: To ensure that: (1) the atmosphere surrounding the irradiated fuel is a non-oxidizing inert gas; (2) the atmosphere is favorable for the transfer of decay heat; and (3) the MSB does not become over-pressurized.

Action: If the required pressure cannot be obtained:

1. Check and repair or replace the pressure gauge;
2. Check and repair or replace the pressure tubes, connections, and valves;
3. Check and repair or replace the helium source; and
4. Check and repair the welds on MSB structural lid.

If pressure exceeds the criterion:
Release a sufficient quantity of helium to lower the cavity pressure.

Surveillance: No maintenance or tests are required during the normal storage. Surveillance of the pressure gauge is required during the helium backfilling operation.

Basis: The value of 14.5 psia was selected to ensure that the pressure within the MSB is within the design limits during any expected off-normal operating condition.

1.2.9 MSB Dye Penetrant Test of Shield and Structural Lid Seal Welds

Limit/Specification:

The MSB shield and structural lid seal welds shall be dye penetrant tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code Section III, Division 1, Article NC-5000 (1986). The liquid penetrant test acceptance standards shall be those described in Sub-section NC-5350 (1983) of the Code.

Applicability: This specification is applicable to the root and final pass welds on the inner shield lid and structural lid seal welds for all MSBs.

Objective: To ensure that the MSB is adequately sealed and leak tight.

Action: If the liquid penetrant test indicates that the weld is unacceptable:

1. The weld shall be ground down and re-welded; and
2. The new weld shall be re-examined in accordance with this specification.

Surveillance: During MSB closure operations.

Basis: Article NC-5000 Examination, ASME Boiler and Pressure Vessel Code, Section III - Division I, Sub-Section NC (1986).

The safety analysis of leak tightness of the MSB is based on a weld being leak-tight to 10^{-4} scc/sec. This check is done to ensure compliance with this design criterion.

1.2.10 Time Limit for Draining the MSB

Limit/Specification:

The water inside the MSB shall be drained within 47 hours after the MSB is removed from the fuel pool, when loaded with fuel producing 24 kW of heat.

The time limit for MSBs loaded with fuel producing less than 24 kW of heat may be determined by inverse linear calculation

y hours = 47 hr [(24 kW)/(X kW)]

where X is the total heat generated by the assemblies loaded into the MSB.

Applicability: This specification is applicable to all MSBs.

Objective: To provide added assurance that significant changes in moderator density cannot occur, and to support the double contingency criteria for criticality safety.

Action: If the water cannot be drained within the specified time, the MSB must be placed back into the fuel pool and allowed to cool.

Surveillance: The time from removing the MSB from the fuel pool until it is drained will be monitored.

Basis: The 47-hour limit was determined from an adiabatic heat-up calculation and hence, is considered conservative.

1.2.11 Placement of the VSC on the Storage Pad

Limit/Specification:

Each VSC shall be placed in a storage array with at least 15-ft ± 1 ft, center-to-center, spacings.

Applicability: This specification applies to all VSCs.

Objective: To provide easy access between casks, and to meet the thermal analysis.

Action: The center-to-center spacing shall be measured upon placement.

Basis: The access requirements are based on engineering judgment. The 15-ft, center-to-center spacing was also used to determine thermal radiation view factors in the thermal analysis. The ± 1 ft will not significantly affect the view factors, or the heat transfer, because almost all heat is removed from the VSC by the natural draft circulation (not thermal radiation) from the exterior sides.

1.2.12 Average Ambient Temperature

Limit/Specification:

The yearly average ambient temperature shall be 75^0 F, or less. Yearly average temperature is to be determined as follows, or by equivalent methodology. Yearly average may be hourly, daily, or monthly average temperatures added together and divided by 8760, 365, or 12, respectively.

The average daily ambient temperature shall be 100° F, or less.

Applicability: This specification applies to every site where the VSC will be deployed.

Objective: To ensure that the long-term ambient conditions are bounded by the analysis.

Action: The yearly average ambient temperature is to be determined from suitable site data, Federal or local government agency data, or other sources. Based on information in the SAR, all United States power plant sites should be bounded by the value of 75^0 F.

Basis: The thermal analysis presented in the SAR used 75^0 F as the long-term average temperature. However, it should be noted that significant margin exists (e.g., 65^0 F for concrete temperatures and 24^0 F for fuel temperatures) (see Table 4.1-1 of the SAR). The thermal analysis presented in the SAR used 100°F as the highest steady state ambient conditions with 125°F the maximum short-term temperature extreme (12 hrs).

1.2.13 Minimum Temperature for Moving the MSB

Limit/Specification:

Movement of the MSB inside the VCC will only be allowed at ambient temperatures of 0^0 F or above.

Objective: To avoid the potential for brittle failure.

Action: Confirm that the ambient temperature is above 0^0 F immediately before moving the MSB, while inside the VCC.

Surveillance: The ambient temperatures shall be measured before movement of the MSB.

Basis: Each MSB shell material will have shown, during fabrication, by Charpy test (per ASTM A370) that it has 15 ft-lb of absorbed energy at -50^0 F; and, therefore, movement of the MSB at temperatures above 0^0 F will avoid the potential for brittle fracture. Calculations show that the MSB shell minimum temperature will be substantially above the ambient temperature (e.g., 20 ^{0}F for 25-year-old fuel). However, for conservatism and simplicity, it is recommended that the ambient be selected as the minimum MSB movement temperature. It is highly unlikely that any MSB movement activity would take place at temperatures below zero. Nevertheless, if movement at a temperature below that specified is necessary, calculations (similar to those presented in Chapter 4 of the SAR) may be used to estimate the minimum MSB shell temperature for any particular ambient condition.

1.2.14 Minimum Temperature for Lifting the MTC

Limit/Specification:

The MTC shall be allowed to be used to move the MSB if the ambient temperature is 40^0 F or above.

Objective: To avoid the potential for brittle failure.

Action: Confirm that the ambient temperature is above 40^0 F before moving the MSB inside the MTC.

Surveillance: The MTC ambient temperature shall be determined before movement of the MSB in the MTC.

Basis: The MTC material will have shown, during fabrication, that it has 15 ft-lb of absorbed energy at 0^0 F. Having Charpy test results, at 0^0 F, which show ductility (or other appropriate test to show that the Nil Ductility Temperature is lower than 0^0 F), will avoid the potential for brittle failure when the cask is moved at 40^0 F or higher. The MTC shell will have a temperature higher than ambient due to the heat source from the irradiated fuel. However, for conservatism and simplicity, it is recommended that the ambient temperature be used as the minimum shell temperature. If movement at lower temperatures is ever required, additional specific analysis or other actions that meet the approval of the NRC must be provided.

1.2.15 MSB Handling Height

Specification:

1. The loaded VCC shall not be handled at a height greater than 60 inches.

2. In the event of a drop of a loaded VCC from a height greater than 18 inches: (a) fuel in the MSB shall be returned to the reactor spent fuel pool; (b) the MSB shall be removed from service and evaluated for further use; and (c) the VCC shall be inspected for damage.

Applicability: The specification applies to handling the VCC, loaded with the MSB, on route to, and at, the storage pad.

Objective:

1. To preclude a loaded VCC drop from a height of greater than 60 inches.

2. To maintain spent fuel integrity, according to the spent fuel specification for storage, continued containment integrity, and VCC functional capability, after a tipover or drop of a loaded VCC from a height greater than 18 inches.

Surveillance: In the event of a loaded VCC drop accident, the system will be returned to the reactor fuel handling building. After the fuel has been returned to the reactor spent fuel pool, the MSB and the VCC will be inspected and evaluated for future use.

Basis: The design basis calculations demonstrated that a fully loaded VCC can be dropped 60 inches without breaching the confinement boundary, requiring removal of spent fuel assemblies, or causing a criticality accident. This specification ensures that handling height limits will not be exceeded in transit to, or at the storage pad. Acceptable damage may occur to the VCC, MSB, and the fuel stored in the MSB for drops of height greater than 18 inches. The specification ensures that the spent fuel will continue to meet the requirements for storage, the MSB will continue to provide confinement, and the VCC will continue to provide its design functions of cooling and shielding. Based on linear-elastic analysis methods, drops up to a height of 18 inches and less are not judged to be of concern.

1.3 Surveillance Requirements

Surveillances required to implement the requirements of a number of specifications were included as part of the specifications in the previous sections. Additional surveillances, required for normal operation and after accident conditions, are described below. Table 2 summarizes all the surveillance requirements, including those discussed in previous sections.

1.3.1 Visual Inspection of Air Inlets and Outlets

Surveillance: A visual surveillance of the wire mesh screens covering the air inlets and outlets shall be conducted daily.

Action: If the surveillance shows signs of degradation, breach of the screens or other possible sources of blockage such as insect infestation, a close-up inspection of the air inlets and outlets shall be conducted to determine possible blockage and removal if present.

Basis: The concrete temperature could exceed 350^{o} F in the accident circumstances of complete blockage of all vents. Concrete temperatures over 350^{o} F in accidents (without the presence of water or steam) are undesirable as they have uncertain impact on strength and durability. A conservative analysis (adiabatic heat case) of complete blockage of all air inlets or outlets indicates that the concrete can reach the accident temperature limit of 350^{o} F in 30 hours.

1.3.2 Exterior VCC Surface Inspection

Surveillance: The VCC exterior surface shall also be inspected annually for any damage (chipping, spalling, etc.).

Action: Any defects larger than one-half inch in diameter (or width) and deeper than one-quarter of an inch shall be repaired by re-grouting, according to the grout manufacturer's recommendations.

Basis: This action maintains the surface condition of the concrete exterior, preventing degradation of the concrete interior, and avoids any adverse impact on shielding performance.

1.3.3 Interior VCC Surface Inspection

Surveillance: The VCC interior surfaces and MSB exterior surfaces of the first VSC unit placed in service at each site shall be inspected, to identify potential air flow blockage and material degradation after every 5 years in service.

Action: Results of the surveillance shall be documented, and a letter report, summarizing the findings, shall be submitted to the NRC within 30 days. The report must be submitted using instructions in 10 CFR 72.4 with a copy sent to administration of the appropriate regional office.

Basis: To identify degradation mechanisms affecting system performance that were not identified in the SAR. However, this surveillance is a conservative but prudent check to ensure suitable conditions remain in the VCC/MSB annulus.

1.3.4 Cask Thermal Performance

Surveillance: Verify a temperature measurement of the thermal performance, for each cask, on a daily basis.

Action: If the temperature measurement shows a significant unexplained difference, so as to indicate the approach of materials to the concrete or fuel clad temperature criteria, take appropriate action to determine the cause and return the cask to normal operation. If the measurement or other evidence suggests that the VCC concrete accident temperature criteria (350° F) has been exceeded for more than 24 hours, the VCC must be removed from service unless the licensee can provide test results in accordance with ACI-349, Appendix A.4.3, demonstrating that the structural strength of the VCC has an adequate margin of safety.

Basis: The temperature measurement should be of sufficient scope to provide the licensee with an positive means to identify conditions which threaten to approach temperature criteria for proper cask operation. The temperature measurement could be any parameter such as (1) a direct measurement of the VCC inner liner temperatures, (2) a direct measurement of the MSB temperatures, (3) a comparison of the inlet and outlet temperature difference to predicted temperature differences for an individual cask, or (4) other means that would identify and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria. If used, air temperatures must be measured in such a manner as to obtain representative values of annulus inlet and outlet air temperatures.

Table 2

Summary of Surveillance Requirements

	Surveillance	Period	Reference Section
1.	Air Outlet Temperature	L, AN	1.2.3
2.	Weld Leak Testing	L	1.2.2
3.	Weld Dye Penetrant Inspection	L	1.2.9
4.	Dose Rates	L	1.2.4
5.	MSB Surface Contamination	L	1.2.5
6.	Vacuum Pressure	L	1.2.7
7.	MSB Helium Backfill Pressure	L	1.2.8
8.	Boron Concentration	PL	1.2.6
9.	VCC, MSB Surveillance (off-normal)	AN	1.2.15
10.	Air Inlet and Outlet Surveillance	D	1.3.1
11.	Cask Exterior (normal)	Y	1.3.2
12.	Cask Interior	AN	1.3.3
13.	Cask Thermal Performance	D	1.3.4

Legend

L During or within 24 hours of loading and before movement to storage pad.
PL Before loading and unloading.
D Daily -- At least once per 24 hours
W Weekly -- At least once per 7 days
M Monthly -- At least once per 31 days
Y Yearly -- At least once per 366 days
AN As necessary/as required.