CONDITIONS FOR CASK USE AND

TECHNICAL SPECIFICATIONS FOR THE

VSC-24 STORAGE CASK SYSTEM

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1.0 INTRODUCTION

This section presents the conditions with which a potential user (licensee) of the Ventilated Storage Cask (VSC-24) system must comply 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 U.S. Nuclear Regulatory Commission (NRC) staff as a result of the review of the Final Safety Analysis Report (FSAR), or are part of the regulatory requirements expressed in 10 CFR 72.212.

1.1 General Requirements and Conditions

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).

Section 72.212(b) requires that the licensee perform written evaluations, before use, establishing that: (1) conditions set forth in the Certificate of Compliance (CoC) 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 FSAR and the associated SERs, 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 that need verification by the system user are as follows:

1. The temperature of 75ºF as the maximum average yearly temperature, without solar incidence;

2. The steady state temperature extremes of 100ºF (average daily temperature) with incident solar radiation and minus 40ºF with no solar incidence;

3. The “accident” short-term temperature extreme of 125ºF with incident solar radiation;

4. The horizontal and vertical seismic acceleration levels of 0.25g and 0.17g, respectively;

5. The analyzed flood condition of 17.7 fps water velocity and submerged depth of 120 feet; and

6. The potential for fire and explosion should be addressed, based on site-specific considerations.

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 FSAR are considered appropriate 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 the use of the valves to permit a determination of 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 MSB 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

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 Technical Specification 1.2.3. The purpose of the test is to measure the heat removal performance of the VSC system and establish base-line data (FSAR Section 9.1.3). A letter report summarizing the results of the test and evaluation shall be submitted to 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 FSAR, with the lesser load as the only exception. The calculation and the measured temperature data shall be reported 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. The use of artificial thermal loads other than spent fuel to satisfy the above requirement is acceptable.

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 specified in Table 1 and Table 2.

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 FSAR bound those of the actual fuel being stored.

Action: For each spent fuel assembly to be loaded into an MSB, compliance with the parameter limits listed in Table 1 shall be independently verified and documented. Compliance with all of the Table 2 assembly parameter requirements for any one of the eight defined fuel assembly classes shall also be verified for each loaded assembly. 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.

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.[1]
Class/Type	• B&W, Mark B, 15 x 15, with and without BPRAs or TPAs • CE/Exxon 15 x 15 with and without poison clusters or plugging clusters • CE 16 x 16 • Westinghouse PWR 17 x 17 with and without BPRAs or TPAs • Westinghouse PWR 15 x 15 • Westinghouse PWR 14 x 14 with and without BPRAs or TPAs
Fuel Cladding	Zircaloy clad fuel with no known or suspected gross cladding failures.[2]
Decay Power Per Assembly	Less than or equal to 1 kW
Maximum Burnup	Less than or equal to 45,000 MWd/MTU
Post Irradiation Time	Greater than or equal to 5 years. Varies with assembly burnup and initial enrichment, as shown in FSAR Table 5.5-1.
Maximum Initial Enrichment	Less than or equal to 4.2 weight percent ^{235}U
Assembly Weight	Less than or equal to 1585 lb (720.5 kg.)
Number of Assemblies per VSC	24

Notes:

(1) High cobalt assemblies (i.e., assemblies with solid stainless steel or stainless steel clad rods in fuel rod locations, which contain 46.7 to 250 grams of initial cobalt within the fuel zone) must not be loaded into the 12 fuel sleeves located around the perimeter of the MSB.

(2) Failed BPRAs or TPAs may be loaded provided that they do not contain Ag-In-Cd or Hf poison material. BPRAs containing these poison materials must have intact fuel cladding, with no known or suspected defects.

Table 2 - Fuel Assembly Class Characterization Parameters (2 pages)

Parameter	Fuel Assembly Class			
	B&W 15x15	**W 14x14**	**W 15x15**	**W 17x17**
Assembly Lattice Layout	see Figure 1	see Figure 2	see Figure 3	see Figure 4
Assembly Pin Pitch	1.44272 cm	1.41224 cm	1.43002 cm	1.25984 cm
Fuel Density	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical
Fuel Pellet Diameter	from 0.92202 cm to 0.94996 cm	from 0.86360 cm to 0.94234 cm	from 0.89408 cm to 0.93980 cm	from 0.75946 cm to 0.82804 cm
Fuel Clad Material	zircaloy	Zircaloy	zircaloy	zircaloy
Fuel Clad Outer Diameter	≤ 1.10236 cm	≤ 1.08712 cm	≤ 1.08712 cm	≤ 0.96012 cm
Fuel Clad Thickness	≥ 0.06604 cm	≥ 0.05588 cm	≥ 0.05842 cm	≥ 0.05588 cm
Guide Tube Material	zircaloy	zircaloy	zircaloy	zircaloy
Guide Tube Outer Diameter	≤ 1.36 cm	≤ 1.37414 cm	≤ 1.4 cm	≤ 1.24 cm
Guide Tube Thickness	≤ 0.045 cm	≤ 0.04318 cm	≤ 0.06 cm	≤ 0.06 cm
Instrument Tube Material	zircaloy	zircaloy	zircaloy	zircaloy
Instrument Tube Outer Diameter	≤ 1.26 cm	≤ 1.37414 cm	≤ 1.4 cm	≤ 1.24 cm
Instrument Tube Thickness	≤ 0.07 cm	all	≤ 0.06 cm	≤ 0.06 cm
Active Fuel Length	≤ 370.84 cm	≤ 373.0 cm	≤ 370.0 cm	≤ 371.0 cm
Guide Bar Effective Diameter [see Note 1]	not applicable	not applicable	not applicable	not applicable
Control Component Rodlets Allowed in Guide Tubes	yes	yes	no	yes
Control Component Rodlet Clad Material	zircaloy or stainless steel	zircaloy or stainless steel	not applicable	zircaloy or stainless steel
Control Component Rodlet Outer Diameter	≤ 1.1176 cm	all	not applicable	≤ 0.9779 cm
Control Component Rodlet Fill Material	any non-hydrogen bearing material	any non-hydrogen bearing material	not applicable	any non-hydrogen bearing material

Note:

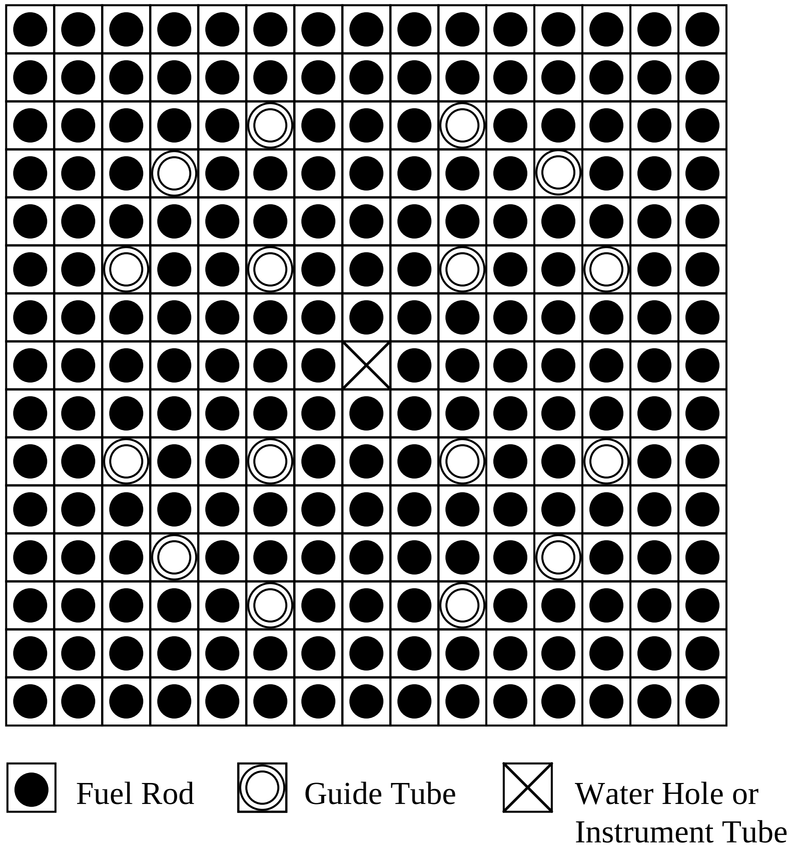
The guide bars may have either a rectangular or circular cross section. The guide bar effective diameter corresponds to the diameter of a circular region having an area equal to the actual guide bar cross-sectional area.

Table 2 - Fuel Assembly Class Characterization Parameters (2 pages)

Parameter	Fuel Assembly Class			
	CE 15x15A	**CE 15x15B**	**CE 15x15C**	**CE 16x16**
Assembly Lattice Layout	see Figure 5	see Figure 6	see Figure 7	see Figure 8
Assembly Pin Pitch	1.397 cm	1.397 cm	1.397 cm	1.28524 cm
Fuel Density	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical	≤ 96% theoretical
Fuel Pellet Diameter	from 0.888 cm to 0.912 cm	from 0.888 cm to 0.912 cm	from 0.888 cm to 0.912 cm	from 0.81534 cm to 0.83566 cm
Fuel Clad Material	zircaloy	zircaloy	zircaloy	zircaloy
Fuel Clad Outer Diameter	≤ 1.062 cm	≤ 1.062 cm	≤ 1.062 cm	≤ 0.98044 cm
Fuel Clad Thickness	≥ 0.0508 cm	≥ 0.0508 cm	≥ 0.0508 cm	≥ 0.06096 cm
Guide Tube Material	not applicable	zircaloy	zircaloy	zircaloy
Guide Tube Outer Diameter	not applicable	≤ 1.06934 cm	≤ 1.06934 cm	≤ 2.54 cm
Guide Tube Thickness	not applicable	≤ 0.02032 cm	≤ 0.02032 cm	≤ 0.1778 cm
Instrument Tube Material	zircaloy	zircaloy	zircaloy	not applicable
Instrument Tube Outer Diameter	≤ 1.06934 cm	≤ 1.06934 cm	≤ 1.06934 cm	not applicable
Instrument Tube Thickness	≤ 0.08509 cm	≤ 0.08509 cm	≤ 0.08509 cm	not applicable
Active Fuel Length	≤ 336.0 cm	≤ 336.0 cm	≤ 336.0 cm	≤ 385.0 cm
Guide Bar Effective Diameter [see Note 1]	≤ 1.2006 cm	≤ 1.2006 cm	≤ 1.2006 cm	not applicable
Control Component Rodlets Allowed in Guide Tubes	not applicable	yes	yes	no
Control Component Rodlet Clad Material	not applicable	zircaloy or stainless steel	zircaloy or stainless steel	not applicable
Control Component Rodlet Outer Diameter	not applicable	all	all	not applicable
Control Component Rodlet Fill Material	not applicable	any non-hydrogen bearing material	any non-hydrogen bearing material	not applicable

Note:

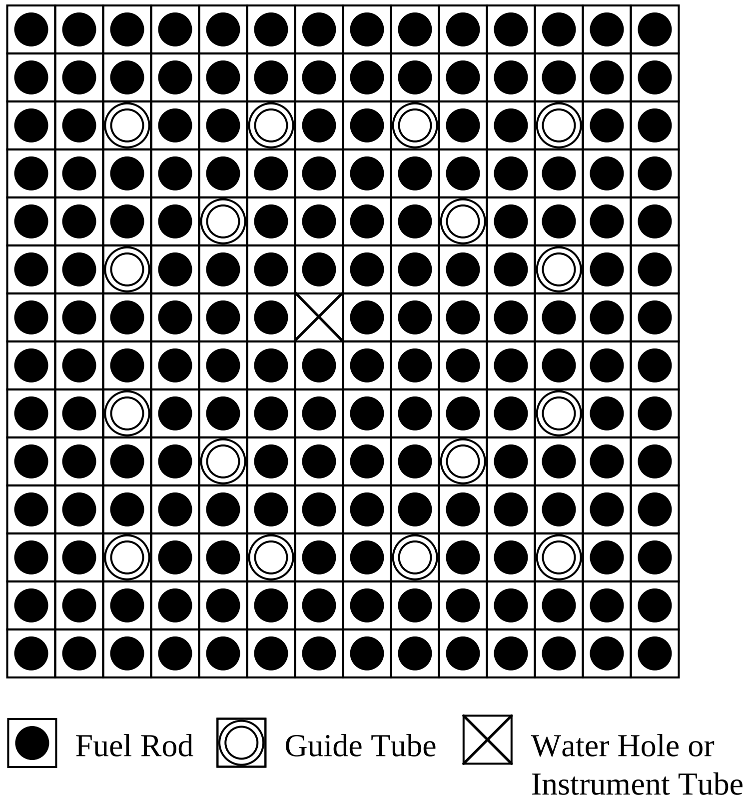
The guide bars may have either a rectangular or circular cross section. The guide bar effective diameter corresponds to the diameter of a circular region having an area equal to the actual guide bar cross-sectional area.



Note 1: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

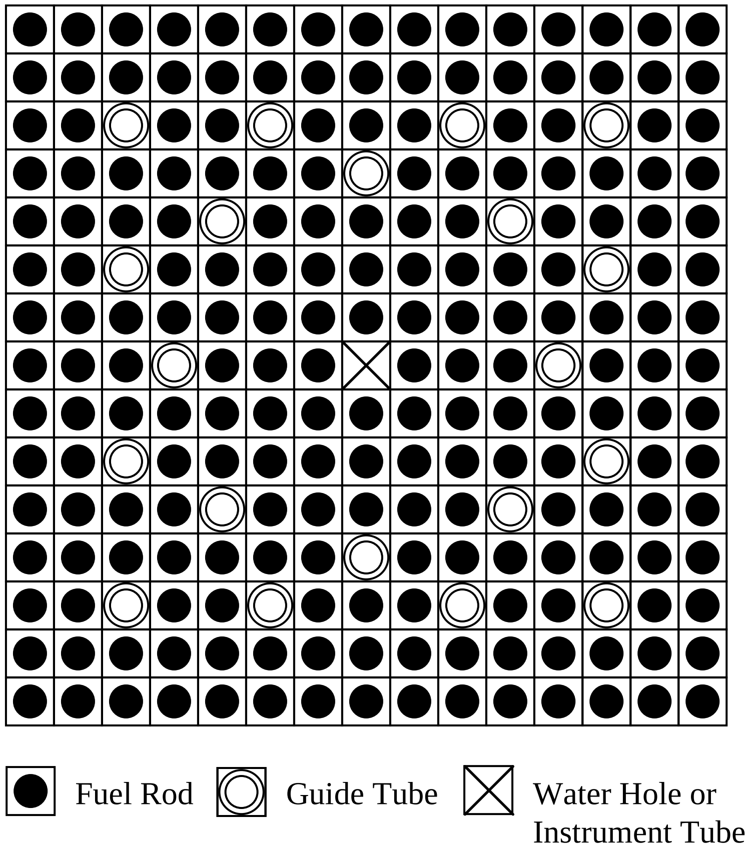
Figure 1 - B&W 15x15 Assembly Class Lattice Layout



Note 1: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

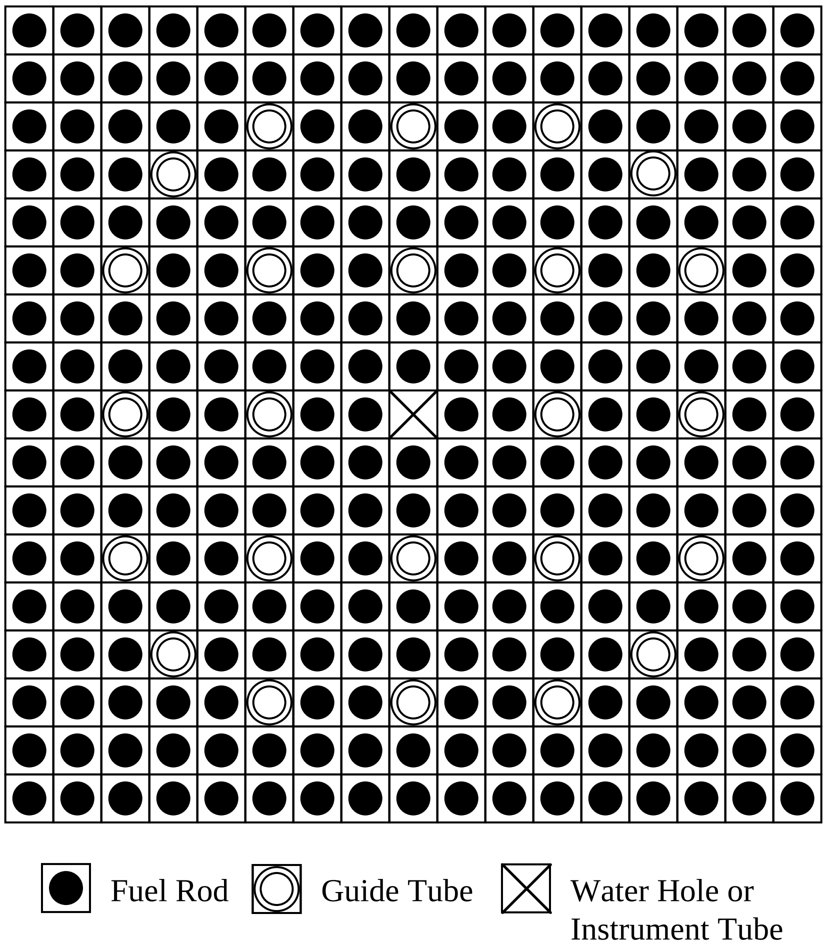
Figure 2 - W 14x14 Assembly Class Lattice Layout



Note 1: Although control component rodlets are (conservatively) modeled in the criticality evaluation, inserted control components are not allowed for the W 15x15 assembly, as shown in Table 2.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

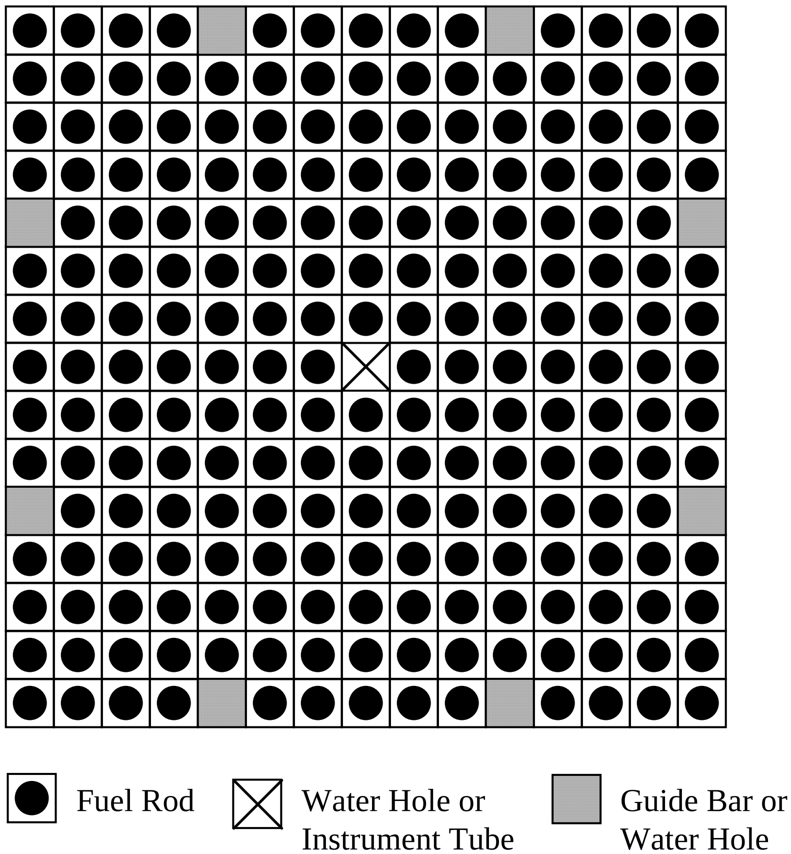
Figure 3 - W 15x15 Assembly Class Lattice Layout



Note 1: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

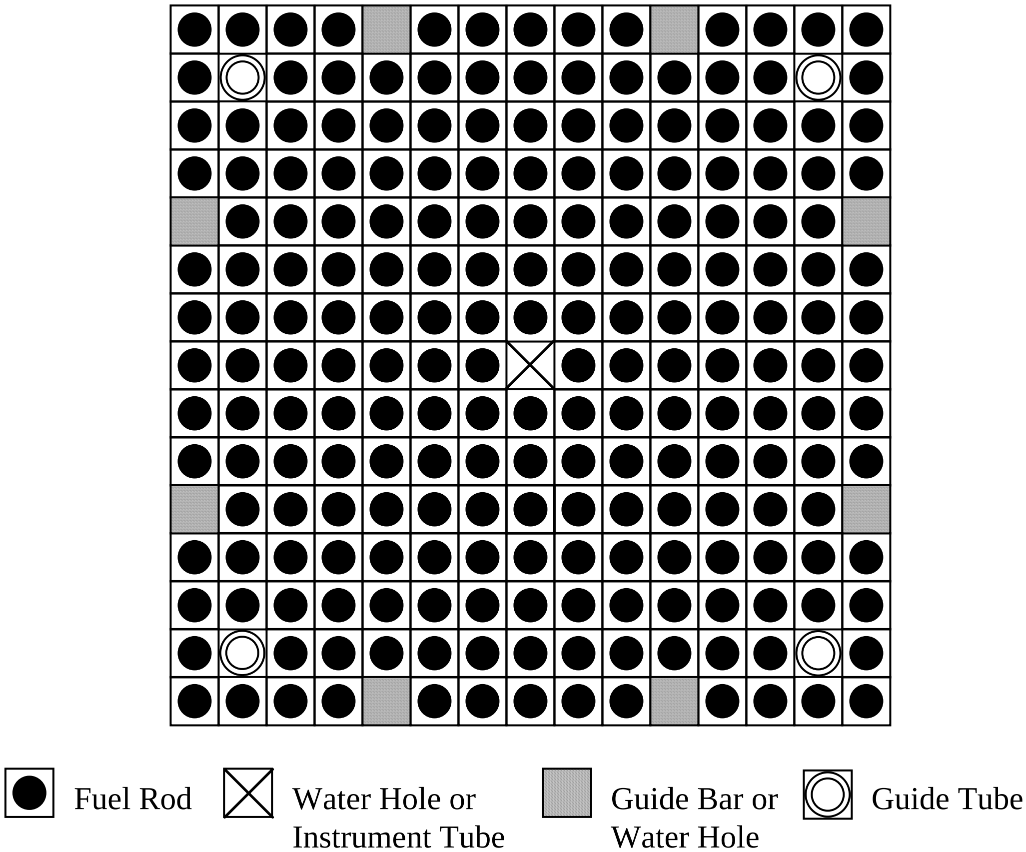
Figure 4 - W 17x17 Assembly Class Lattice Layout



Note 1: The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Figure 5 - CE 15x15A Assembly Class Lattice Layout

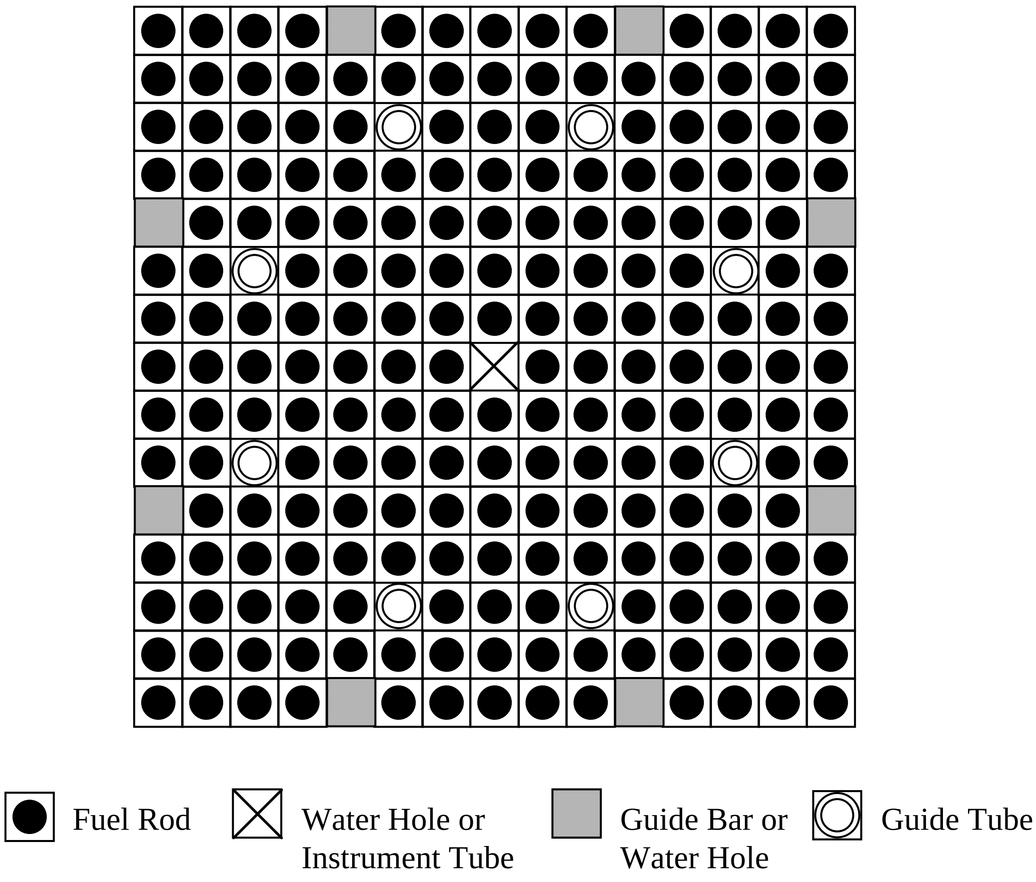


Note 1: The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Note 3: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Figure 6 - CE 15x15B Assembly Class Lattice Layout

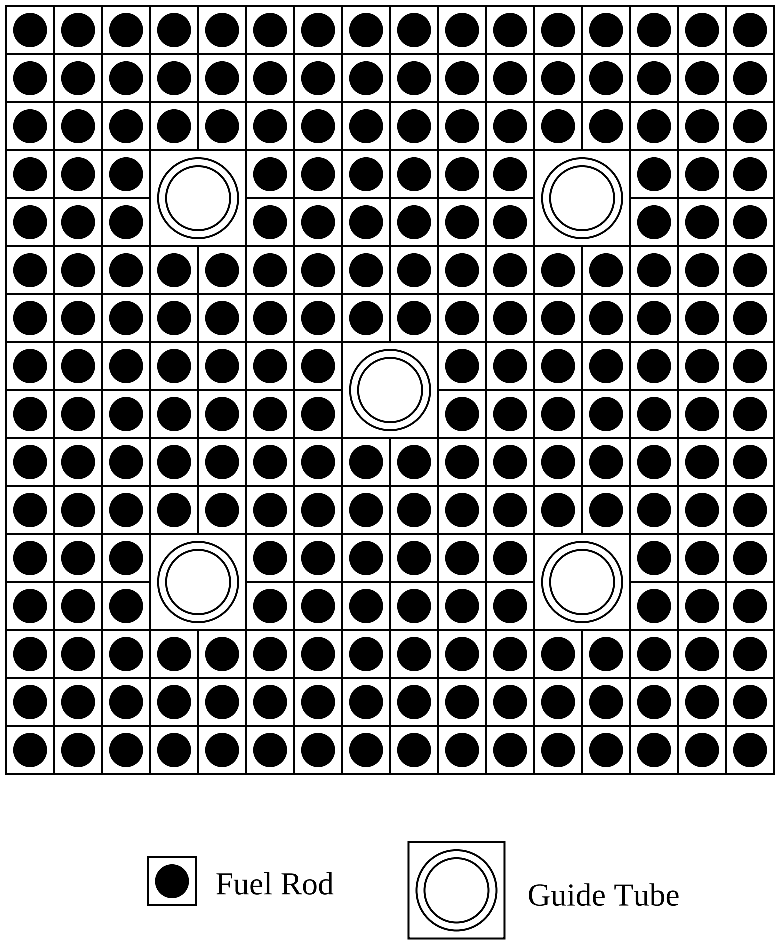


Note 1: The guide bars as shown may contain any form of bar or rod as long as the cross-sectional area is less than or equal to that corresponding to a solid rod with the specified guide bar effective diameter.

Note 2: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Note 3: The guide tubes as shown may contain any form of control component rodlet as long as the control component rodlet does not contain a hydrogen-bearing material.

Figure 7 - CE 15x15C Assembly Class Lattice Layout



Note 1: The fuel rods as shown may be replaced with any other type of rod as long as the outer diameter of the replacement rod is less than or equal to that of the fuel rod.

Figure 8 - CE 16x16 Assembly Class Lattice Layout

1.2.2 Maximum Permissible MSB Leak Rate

Limit/Specification:

Less than or equal to 1.0×10^{-4} standard cubic centimeters per second (scc/sec) at 0.5 atm differential pressure.

Applicability: MSB inner seal (shield lid weld) 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 all performed on the MSB outer surface.

Surveillance: The MSB shall be tested after the inner seal weld (shield lid 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, addition 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.

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°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 FSAR.

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 FSAR. That section shows that if the air temperature increase (for 24 kW) is below 110°F (expected to be 89°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°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°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 FSAR. If fuel assemblies meeting the fuel specification in Technical Specification 1.2.1 have been loaded into the cask and the temperature difference is greater than 110°F, or that predicted for less heat loads, then this condition is not addressed in the FSAR and will require additional measurements and analysis to determine that the actual performance of the cask is within the limits analyzed in the FSAR. 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 hours until the cask has reached thermal equilibrium. After reaching thermal equilibrium, thermal performance of each cask shall be verified on a daily basis through visual inspection of the VCC vent screens in accordance with Technical Specification 1.3.1.

1.2.4 Maximum External Surface Dose Rate

Limit/Specification:

The external surface average dose rate from all types of radiation will be less than 100 mrem per hour on the sides and 200 mrem/hr on the top. Dose rates at the air inlets and outlets will be below 350 and 100 mrem/hr, respectively.

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 FSAR, to provide verification for plant personnel that radiation levels are acceptably low, and to satisfy the 10CFR72 dose rate limit of 25 mrem/year at the ISFSI controlled area boundary.

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.

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.

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, the concentration specified (as a function of assembly initial enrichment for each defined assembly class) in Figures 9 through 16.

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 analyzed chemically 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.

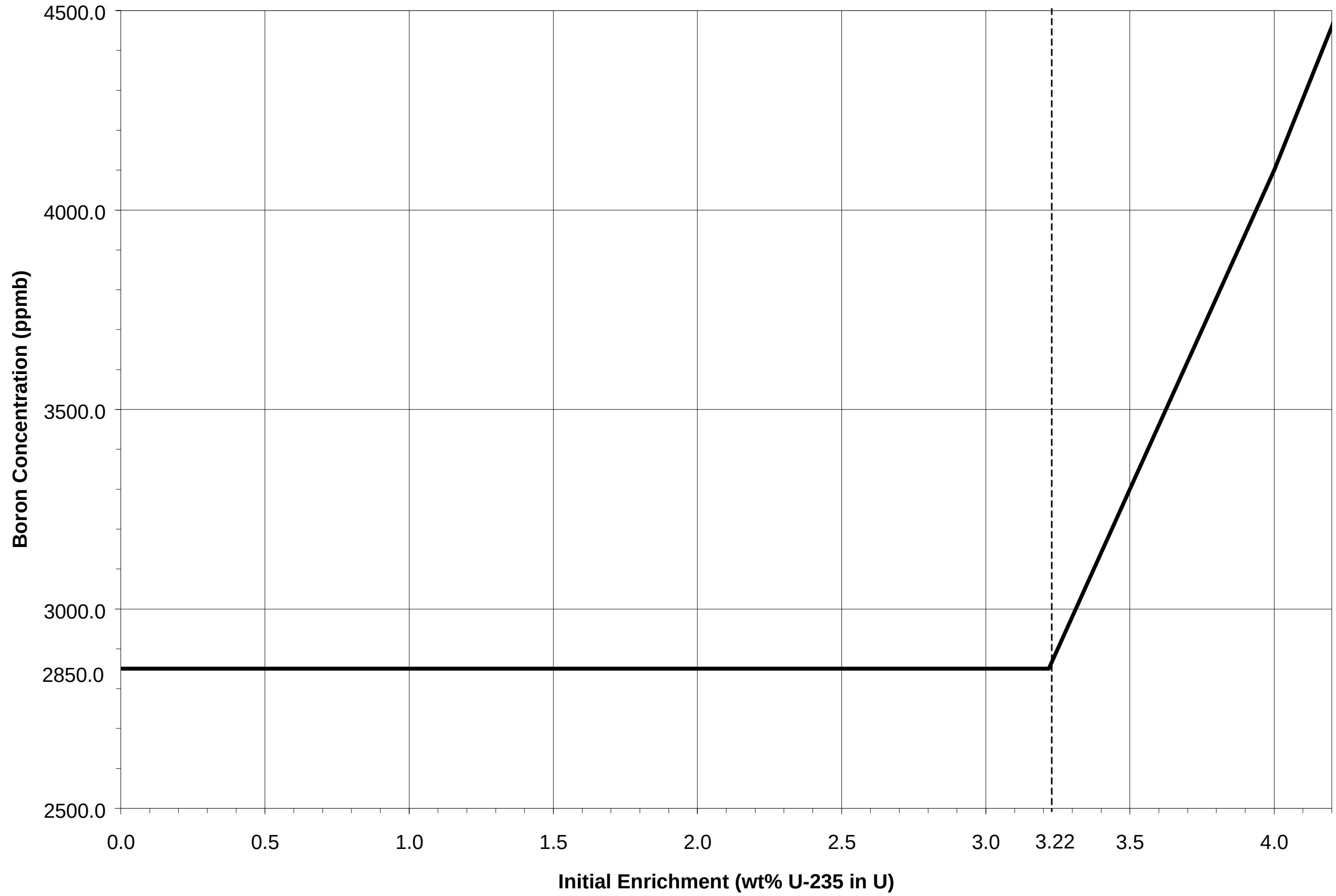


Figure 9 - B&W 15x15 Assembly Class Minimum Required Soluble Boron

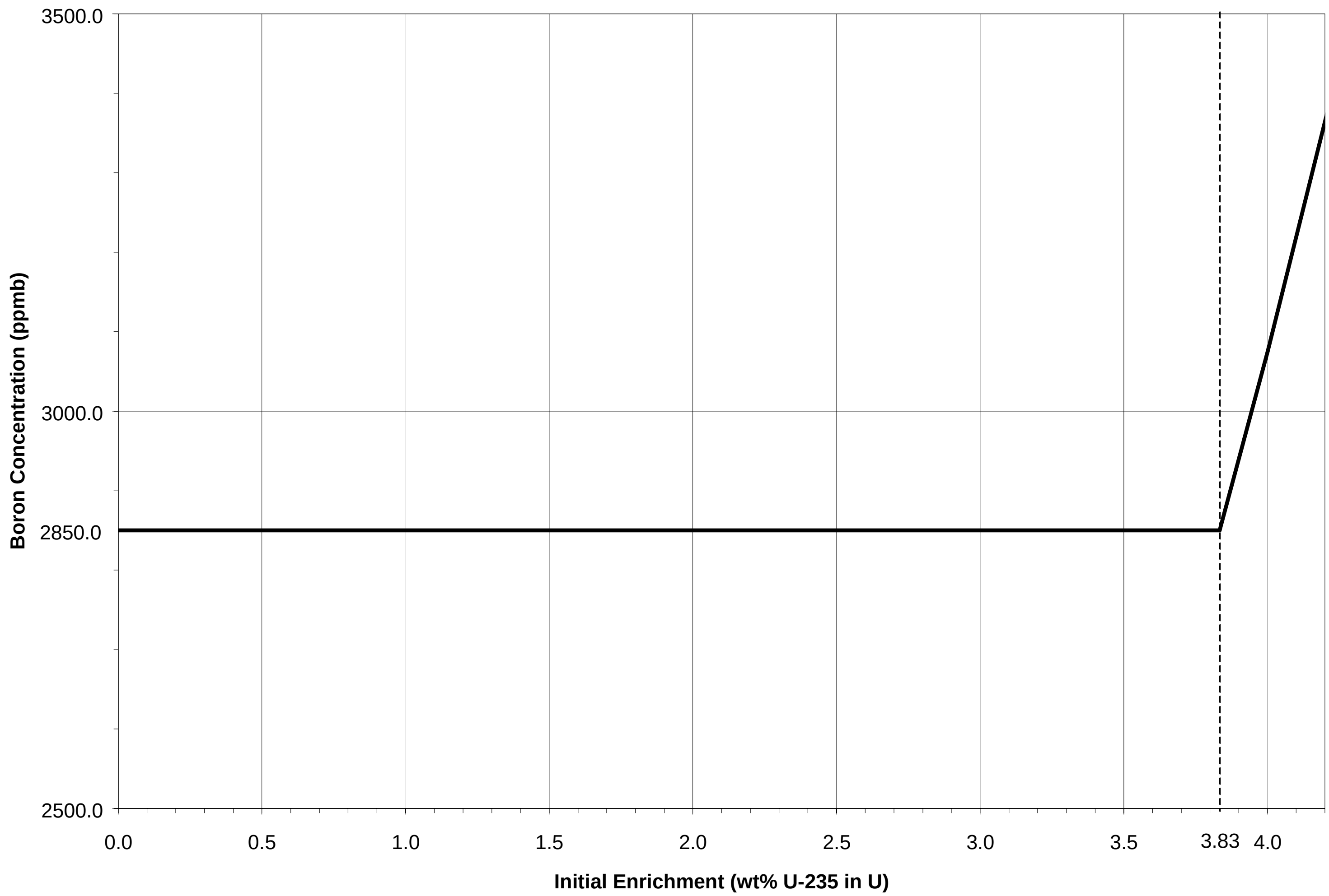


Figure 10 - W 14x14 Assembly Class Minimum Required Soluble Boron

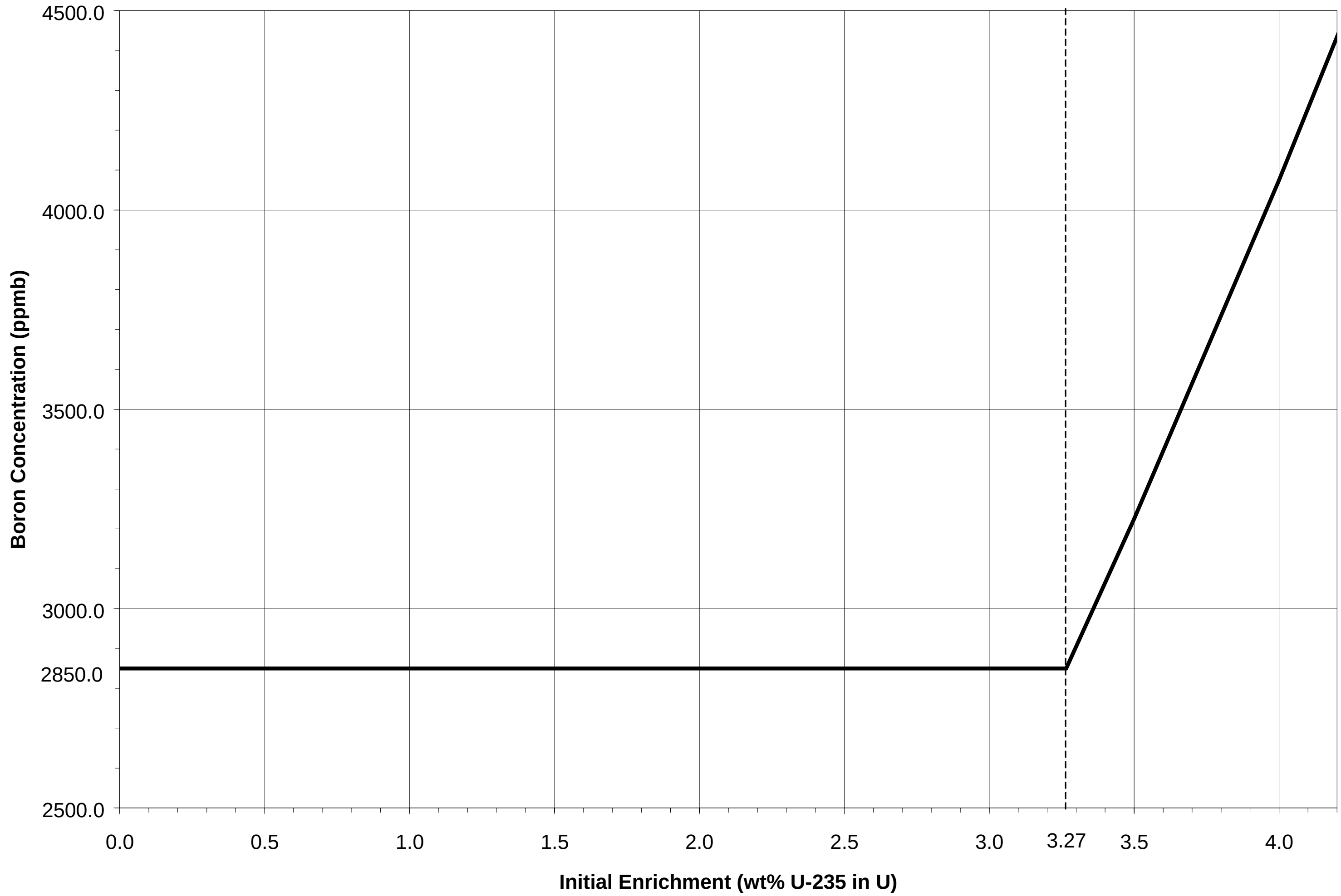


Figure 11 - W 15x15 Assembly Class Minimum Required Soluble Boron

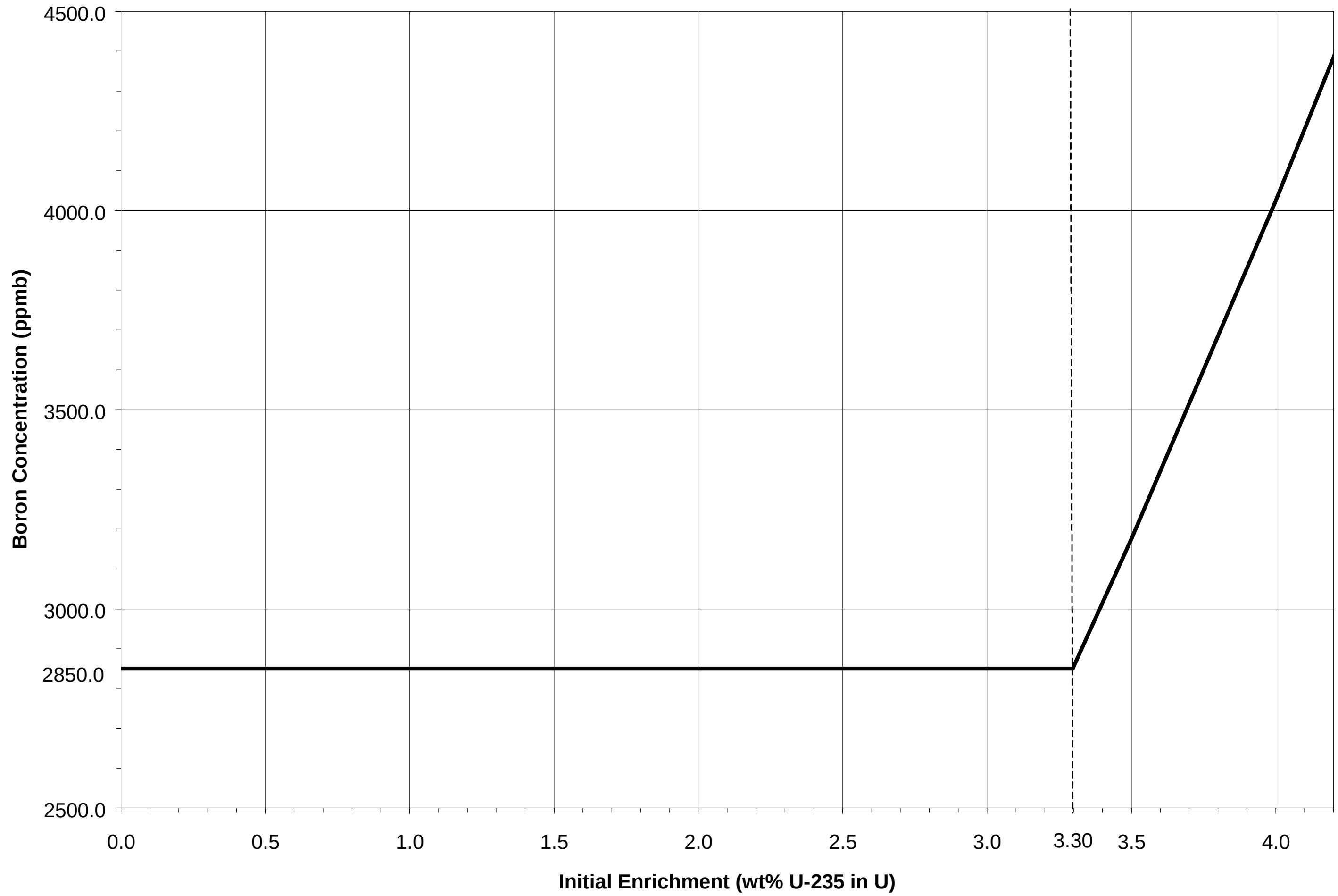


Figure 12 - W 17x17 Assembly Class Minimum Required Soluble Boron Results

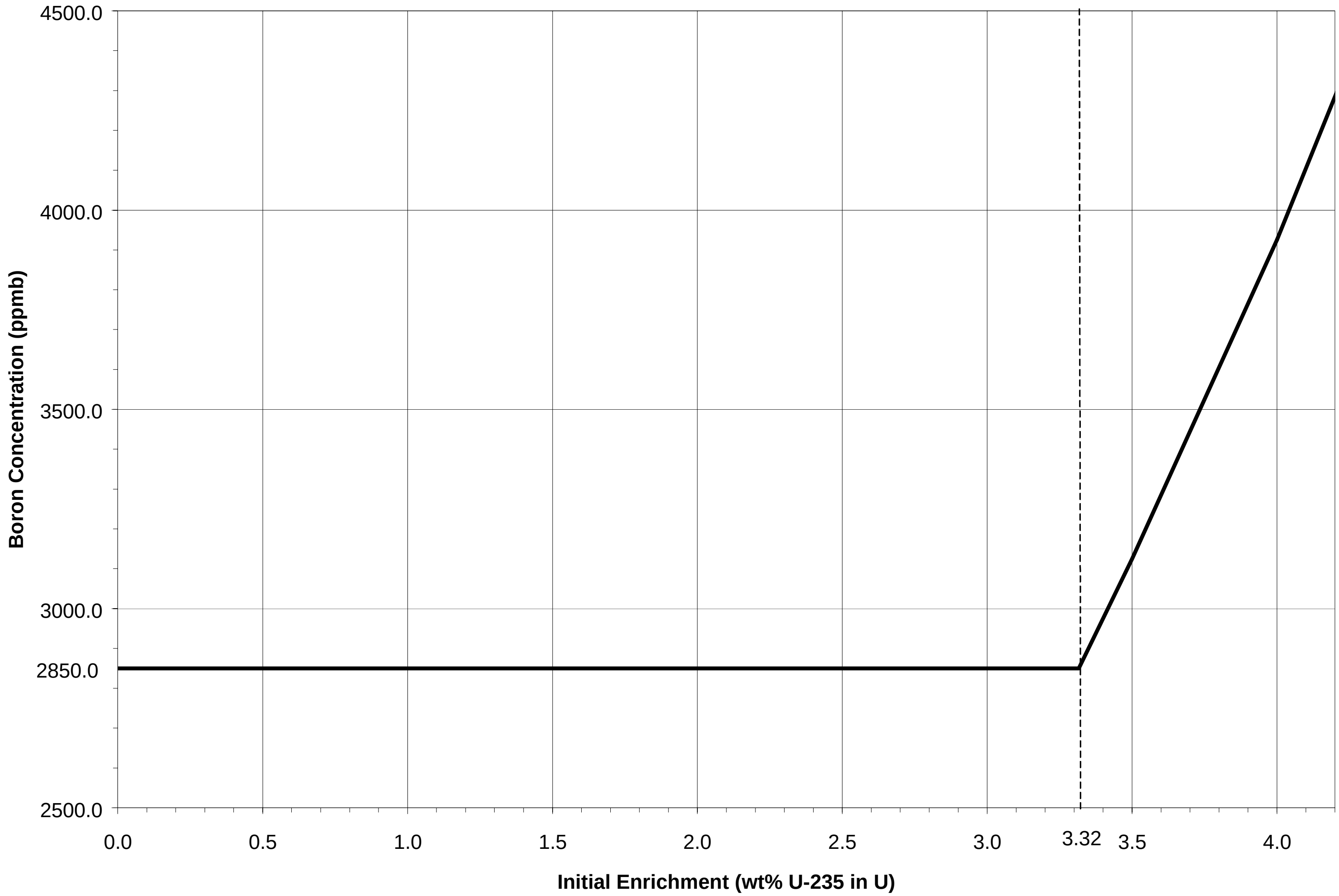


Figure 13 - CE 15x15A Assembly Class Minimum Required Soluble Boron Results

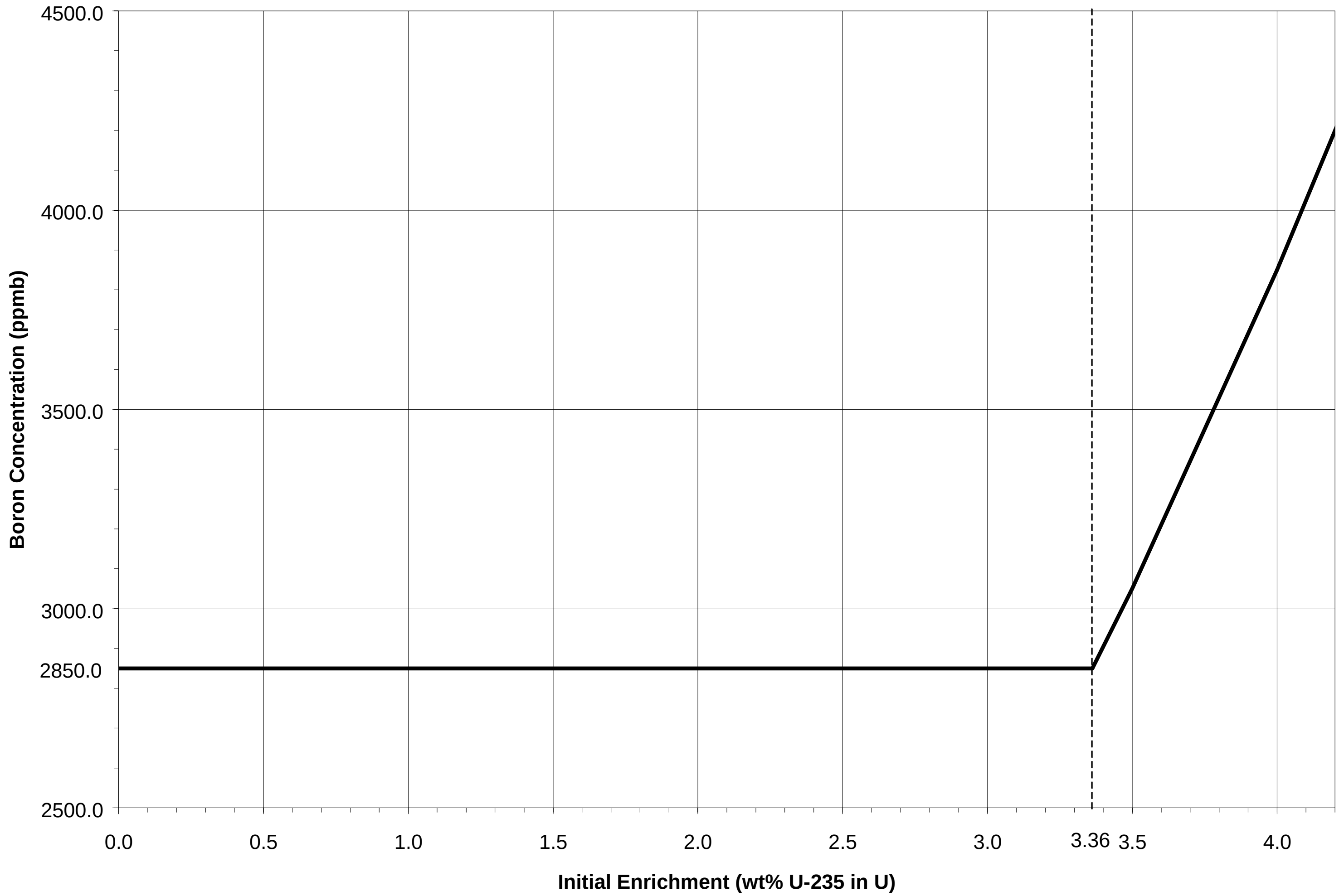


Figure 14 - CE 15x15B Assembly Class Minimum Required Soluble Boron Results

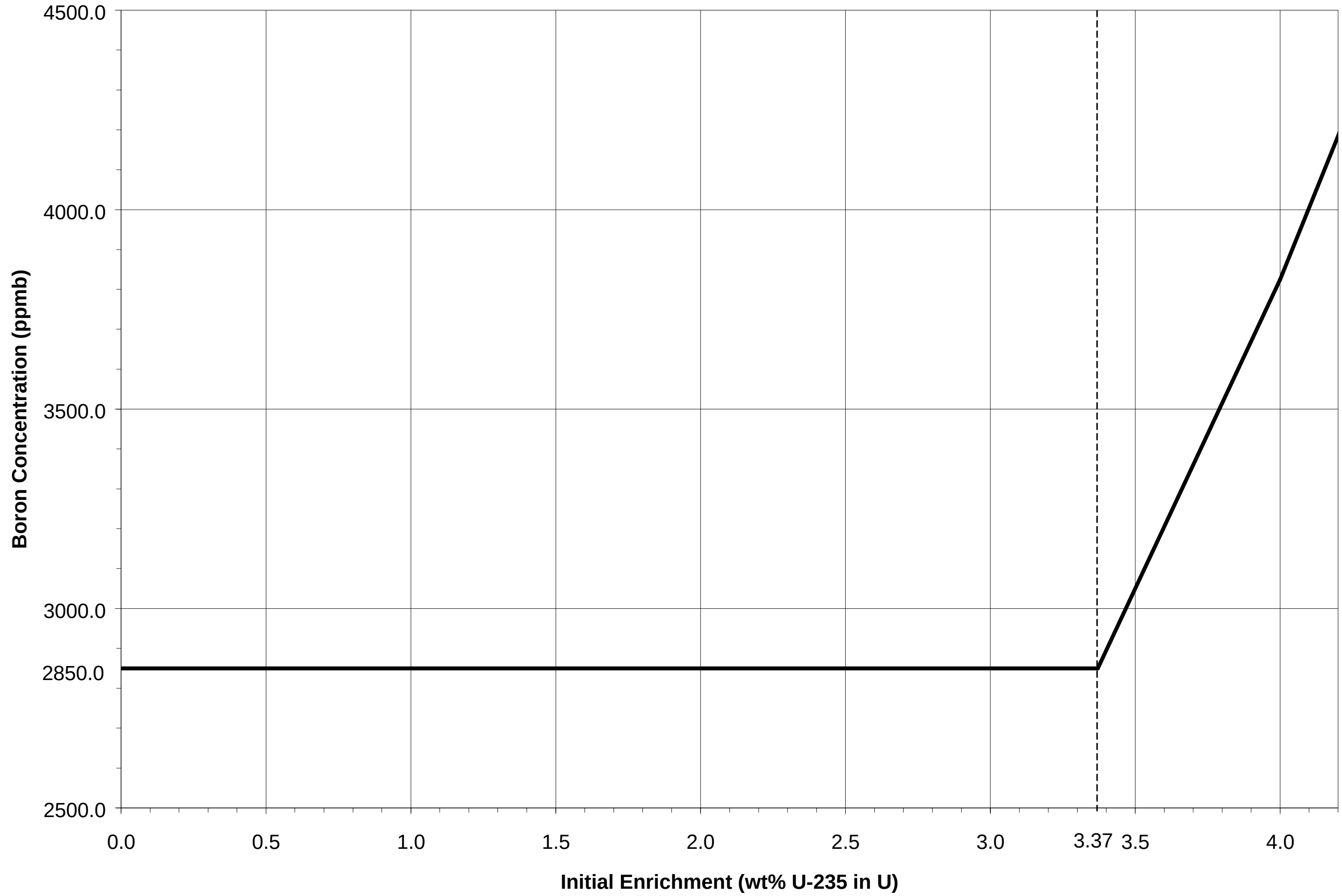


Figure 15 - CE 15x15C Assembly Class Minimum Required Soluble Boron Results

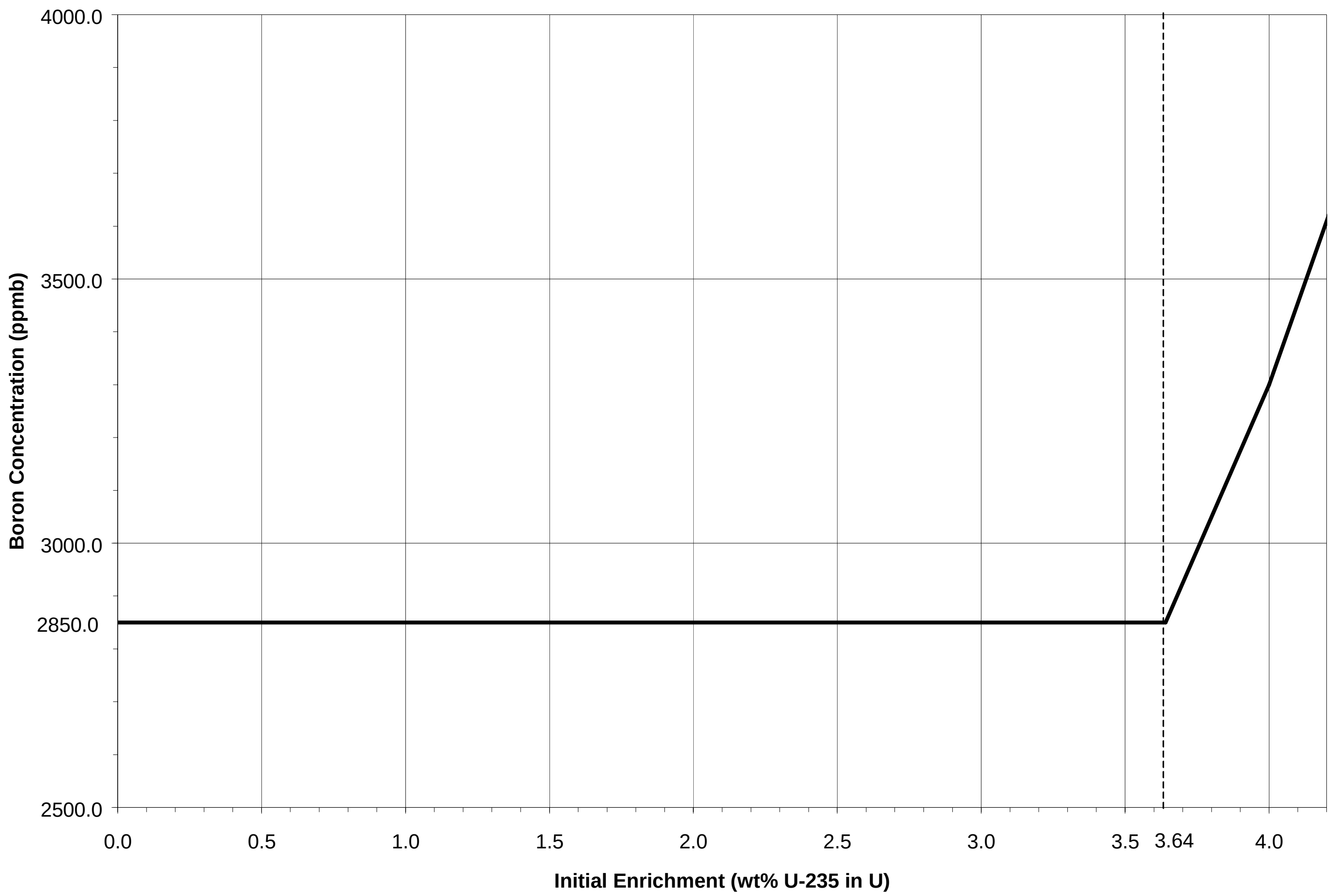


Figure 16 - CE 16x16 Assembly Class Minimum Required Soluble Boron Results

1.2.7 MSB Vacuum Pressure During Drying

Limit/Specification:

Vacuum Pressure:	Less than or equal to 3 mm Hg
Time at Pressure:	Greater than or equal to 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, and 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; or
3. Check and reseal the shield lid fitting(s).

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

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 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.

1.2.9 Non-Destructive Examination of Shield and Structural Lid Seal Welds

Limit/Specification:

The MSB pressure boundary shield lid, structural lid and valve cover plate closure welds shall be liquid penetrant tested (PT) in accordance with the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Article NC-5000 (1986 edition, 1988 addenda). The PT acceptance standards shall be as described in Subsection NC-5350.

In addition, the MSB structural lid-to-shell weld shall be examined by ultrasonic testing (UT) in accordance with the criteria defined in the "Guideline Requirements for the Time-of-Flight Diffraction Ultrasonic Examination of the VSC-24 Structural Lid to Shell Weld," VMSB-98-001, latest version.

The specified PT and UT examination shall begin no sooner than two hours after completion of the weld to be examined.

Applicability: For all MSBs, the PT examination is applicable to:

1. Root and final weld surfaces between the shield lid and shell, and between the structural lid and shell; and

2. Final weld surfaces between the structural lid and shield lid, and between the valve cover plate and structural lid.

The confirmatory UT examination is applicable to the completed MSB structural lid-to-shell weld.

Objective: To ensure that the MSB is adequately sealed and leak-tight, and to confirm the integrity of the structural lid-to-shell weld.

Action: If the PT examination indicates that a weld is unacceptable:

1. The weld shall be repaired in accordance with Article NC-4000 Fabrication and Installation, ASME Boiler and Pressure Vessel Code, Section III – Division 1, Subsection NC (1986 edition, 1988 addenda); and

2. The repaired weld shall be re-examined in accordance with the requirements of this specification.

If indications are found as a result of the UT examination:

1. Evaluate the flaw proximity per ASME Section XI, IWA-3300 (1989 edition);

2. Compare each flaw to the flaw screening criteria provided below:

Acceptable Flaw Depth (for lengths less than or equal to 0.7 in.)	Acceptable Flaw Depth (for lengths greater than 0.7 in.)
0.37 inches	0.16 inches

3. If a flaw is unacceptable, perform further flaw-specific evaluations (i.e., linear-elastic fracture mechanics or elastic-plastic fracture mechanics) per VMSB-98-001, latest revision, to determine whether the flaw is acceptable for continued operation; or

4. Repair the weld in accordance with Article NC-4000 Fabrication and Installation, ASME Boiler and Pressure Vessel Code, Section III - Division 1, Subsection NC (1986 edition, 1988 addenda); and

5. Re-examine the repaired weld in accordance with this specification.

Surveillance: During MSB closure operations.

1.2.10 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.

1.2.11 Average Ambient Temperature

Limit/Specification:

The yearly average ambient temperature shall be 75°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 FSAR, all United States power plant sites should be bounded by the value of 75°F.

1.2.12 Minimum Temperature for Moving the Loaded MSB

Limit/Specification:

A VCC containing a loaded MSB shall only be moved at ambient temperatures of 0ºF or above, coincident with a structural lid-to-shell weld temperature of 30°F or above.

Objective: To conform with design basis criteria for brittle fracture.

Action: Confirm before moving the VCC containing the loaded MSB that the ambient temperature is at 30ºF or above.

If the ambient is less than 30ºF but 0ºF or greater, confirm that the structural lid-to-shell weld is at 30ºF or above. Physical measurement should be used to determine the structural lid-to-shell weld temperature. Alternately, calculations similar to those presented in Chapter 4 of the FSAR may be used for the specific fuel to determine the minimum MSB shell temperature for any particular ambient condition.

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

1.2.13 Minimum Temperature for Lifting the MTC

Limit/Specification:

The MTC containing a loaded MSB shall only be moved at ambient temperatures of 40°F or above.

Objective: To conform to the design criteria for brittle failure.

Action: Confirm that the ambient temperature is 40°F or above before movement of the MTC containing a loaded MSB.

Surveillance: The MTC ambient temperature shall be determined before movement of the MTC containing a loaded MSB.

1.2.14 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.

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 3 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.

1.3.2 Exterior VCC Surface Inspection

Surveillance: The VCC exterior surface shall 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.

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.

Table 3 - Summary of Surveillance Requirements

	Surveillance	Period	Technical Specification or Surveillance Requirement
1.	Spent Fuel Assembly Identification	L	1.2.1
2.	Weld Leak Testing	L	1.2.2
3.	Air Outlet Temperature	L, AN	1.2.3
4.	Dose Rates	L	1.2.4
5.	MSB Surface Contamination	L	1.2.5
6.	Boron Concentration	PL	1.2.6
7.	Vacuum Pressure	L	1.2.7
8.	MSB Helium Backfill Pressure	L	1.2.8
9.	Weld Nondestructive Examination	L	1.2.9
10.	Ambient Temperature (VCC Movement)	AN	1.2.12
11.	Ambient Temperature (MTC Lift)	AN	1.2.13
12.	VCC, MSB Drop	AN	1.2.14
13.	Air Inlet and Outlet Surveillance	D	1.3.1
14.	Cask Exterior (normal)	Y	1.3.2
15.	Cask Interior	AN	1.3.3

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.

TECHNICAL SPECIFICATIONS BASES

FOR THE

VSC-24 STORAGE CASK SYSTEM

B.1.2.1 Fuel Specification

Basis: The specification is based on consideration of the design basis parameters included in the FSAR. 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 FSAR Section 2.1. Allowable parameters of stored fuel are described in detail in Table 2. The VSC-24 cask can accommodate the standard PWR fuel designs manufactured by Combustion Engineering (CE), Exxon, Westinghouse, and Babcock and Wilcox (B&W). The specific designs accommodated by the VSC-24 are listed in Table 1. For a given assembly to qualify for loading into the cask, however, it must meet all of the geometry specifications given for that assembly type in Table 2.

For all of the listed assembly designs except W 15x15 and CE 16x16, assemblies containing burnable poison rod assemblies (BPRAs) and thimble plug assemblies (TPAs) may be loaded into the VSC-24 cask. The poison rods or plugging rods may be Zircaloy or stainless steel clad. Plugging (TPA) rods may be either hollow or solid metal tubes. CE 15x15 assemblies may contain poison rods or poison “clusters” in any number of the assembly guide tubes.

BPRAs consist of a group of hollow metal rods that are filled with boron carbide (B_4C), aluminum oxide (Al_2O_3), a silver-indium-cadmium mixture (Ag-In-Cd), borosilicate glass, or halfnium (Hf) poison material. These rods are inserted into the assembly guide tubes. The metal rods are made of stainless steel or zircaloy. TPAs contain a group of solid or hollow metal rods. These (relatively short) rods are inserted into the top section of the assembly guide tubes (in order to “plug” the guide tubes). TPA rods consist of stainless steel or zircaloy. Both BPRAs

and TPAs have additional metal hardware (a “spider assembly”) at the top end which binds the group of rods together and provides a handle to allow the BPRA or TPA to be engaged by fuel assembly handling equipment. The top end hardware is made of stainless steel, zircaloy, and/or nickel alloys such as inconel.

The VSC-24 may also accommodate PWR assemblies with any number of solid zircaloy rods, poison rods, or Gd_2O_3 rods in place of standard fuel rods, given that the diameter of the replacement rods does not exceed that of the standard fuel rods. Poison rods (like BPRA rods) are hollow stainless steel or zircaloy rods filled with B_4C, Al_2O_3, Ag-In-Cd, borosilicate glass, or Hf poison material. Gd_2O_3 rods are hollow stainless steel or zircaloy rods that are filled with Gd_2O_3 poison material instead of UO_2 fuel.

For fuel assemblies with stainless-steel-clad or solid stainless steel rods in place of fuel rods, or with solid stainless steel rods in guide tube locations, an evaluation must be performed to verify that the assembly fuel zone cobalt content lies within the analysis basis discussed in FSAR Section 5.2.1.2 (and in FSAR Table 5.2-2). If the overall assembly fuel zone cobalt quantity does not exceed the design basis quantity of 46.7 grams (given in FSAR Table 5.2-2), the assembly may be loaded into any location inside the MSB. If the assembly fuel zone cobalt quantity exceeds the design basis amount of 46.7 grams, but does not exceed the “high” assembly cobalt quantity of 250 grams (defined and qualified in FSAR Section 5.5.2), the assembly may be loaded into any one of the inner 12 fuel sleeves in the MSB (i.e., it may not be loaded into any of the 12 sleeves that lie along the MSB edge).

The analyses presented in the FSAR 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

FSAR 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 assembly types defined in Table 2, with initial enrichments up to 4.2 wt. percent ^{235}U. The criticality design criteria ensure that the MSB remains subcritical (k_{eff} less than or equal to 0.95) under normal, off-normal, and accident conditions assuming the loading of unirradiated fuel. The assembly initial enrichment is defined as the maximum lattice average enrichment that occurs at any axial location within the assembly.

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.

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 (FSAR References 1.1 and 4.1). Based on this methodology, 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 not be so high as to cause the cask exterior dose rate limits in Technical Specification 1.2.4 to be exceeded.

The design basis shielding calculations show that these dose rate limits are not exceeded for a cask loaded with 3.2 weight percent ^{235}U enriched PWR fuel, irradiated to an average fuel burnup of 35,000 MWd/MTU, with a post-irradiation time of 5 years.

Additional shielding analyses determine the cooling time required to meet the specified dose limits for different combinations of assembly burnup and initial enrichment. The minimum required cooling time to meet the 1.0 kW assembly decay power limit is also determined.

The minimum assembly cooling times at which both the 1.0 kW assembly decay power limit and the Technical Specification 1.2.4 dose rate limits are met are shown in FSAR Table 5.5-1.

The cooling times provided in FSAR Table 5.5-1 apply to the fuel assemblies with and without BPRAs or TPAs. The calculated assembly and radiation sources include both the fuel and the inserted BPRAs and TPAs.

The criticality analyses used to determine the minimum required soluble boron concentration and ensure a k_{eff} of less than 0.95, consider BPRAs inserted into the fuel assemblies. These criticality analyses conservatively modeled the presence of the individual rods of the BPRAs as zircaloy tubes filled with depleted B_4C (i.e., $(B_{11})\ _4C$).

BPRAs with cladding failures are acceptable for loading in the VSC-24 system. 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. An exception to this is poison rods containing Ag-In-Cd or Hf poison materials, which could possibly interact galvanically with the MSB internal components. For this reason, the cladding of BPRA rods or inserted poison rods that contain Ag-In-Cd or Hf poison material must be intact (i.e., must have no known or suspected gross failures).

B.1.2.2 Maximum Permissible MSB Leak Rate

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.

B.1.2.3 Maximum Permissible Air Outlet Temperature

Basis: If the air temperature rise is 110°F [21°F more than the 89°F rise calculated for the 75°F ambient case (FSAR Table 4.1-1), the maximum concrete and fuel cladding temperatures can be expected to be less than 21°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 214°F and a maximum cladding temperature of 705°F. Both of these values are below the acceptable criteria (225°F for concrete and 712°F for 5-year cooled fuel).

B.1.2.4 Maximum External Surface Dose Rate

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

B.1.2.5 Maximum MSB Removable Surface Contamination

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 FSAR. 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.

B.1.2.6 Boron Concentration in the MSB Cavity Water

Basis: The required boron concentration is based on the criticality analysis for an MSB with unburned fuel, maximum enrichment, and optimum moderation conditions. Required boron concentrations are calculated versus initial enrichment for each defined PWR assembly class. The sets of allowable values for assembly physical parameters important to criticality safety are defined for each PWR assembly class in Table 2.

The required boron concentrations apply only to assemblies that meet all of the physical parameter restrictions given in Table 2 for that assembly type. The boron concentrations are also applicable for assemblies containing BPRAs, TPAs, poison rods in place of fuel rods, or solid zircaloy or steel replacement rods.

The initial enrichments shown in Figures 9 through 16, upon which the minimum required boron concentrations are based, are defined as the maximum lattice average enrichment that occurs for any axial position within the assembly. Most PWR assembly types use a uniform fuel rod enrichment throughout the assembly lattice. For assemblies that have multiple fuel rod enrichments, an evaluation must be performed to verify that the actual fuel rod enrichment pattern is less reactive than a uniform fuel rod enrichment pattern at the lattice-average enrichment. Alternatively, the maximum enrichment of the individual fuel rods in the assembly lattice may be used to determine the required boron concentration.

B.1.2.7 MSB Vacuum Pressure During Drying

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°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 the 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.

B.1.2.8 MSB Helium Backfill Pressure

Basis: The value of 14.5 psia was selected to assure that the pressure within the MSB is within the design limits during any expected off-normal operating condition. The 14.5 psia backfill pressure assumes that the average helium temperature in the MSB is greater than 200°F when the MSB is sealed. The combination of pressure equal to 14.5 psia and temperature greater than 200°F assures an upper limit on the moles of helium in the MSB.

The MSB backfill helium shall be high-purity grade helium (99.995%).

B.1.2.9 Non-Destructive Examination of Shield and Structural Lid Seal Welds

Basis: Article NC-5000 Examination, ASME Boiler and Pressure Vessel Code, Section III – Division 1, Subsection NC (1986 edition, 1988 addenda).

Two hour delay in initiation of the specified PT and UT examinations ensures that closure welds will be inspected after any potential delayed hydrogen-induced cracking.

The leak tightness analysis for the MSB is based on welds being leak-tight to 10^{-4} scc/sec. These examinations are performed to ensure compliance with the leak tightness design criteria, and to confirm the integrity of the structural lid-to-shell weld.

B.1.2.10 Placement of the VSC on the Storage Pad

Basis: The access requirements are based on engineering judgement. 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 most all heat is removed from the VSC by the natural draft circulation (not thermal radiation) from the exterior sides.

B.1.2.11 Average Ambient Temperature

Basis: The thermal analysis presented in the FSAR used 75°F as the long-term average temperature. However, it should be noted that significant margin exists (e.g., 45°F for concrete temperatures and 28°F for fuel temperatures). The thermal analysis presented in the FSAR used 100ºF as the highest steady state ambient conditions with 125ºF the maximum short-term temperature extreme (12 hrs).

B.1.2.12 Minimum Temperature for Moving the Loaded MSB

Basis: Movement of the loaded MSB at a 30ºF ambient temperature or above conservatively satisfies this specification. Restricting movement of the loaded MSB below the temperatures specified is necessary to conform with the design criteria for brittle failure.

Each MSB shell material will have shown, during fabrication, by Charpy test (per ASTM A370) that it has 15 foot-pounds of absorbed energy at minus 50°F.

The temperature limit for the structural lid-to-shell weld effectively increases material toughness and permits allowable flaw size in the weld to be governed by primary stress criteria rather than by brittle fracture limits.

Specifications for future procurement and fabrication of MSB pressure retaining materials, including base materials and weld metal, shall specify a minimum Charpy V-notch impact absorbed energy value of 15 foot-pounds at minus 50ºF. Additionally, for the MSB shell, lid, and weld materials associated with the structural lid-to-shell weld, the minimum Charpy V-notch impact absorbed energy shall be 45 foot-pounds at 0ºF. These requirements define minimum values for material toughness and produce adequate margins of safety relative to the potential for brittle fracture under the most severe handling conditions.

B.1.2.13 Minimum Temperature for Lifting the MTC

Basis: The MTC material will have shown, during fabrication, that it has 15 ft-lb of absorbed energy at 0ºF. Having Charpy test results, at 0°F, which show ductility (or other appropriate test to show that the Nil Ductility Temperature is lower than 0°F), is necessary to conform to the design criteria for brittle failure when the cask is moved at 40°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.

B.1.2.14 MSB Handling Height

Basis: Drops up to 60 inches, of the MSB inside the VCC, can be sustained without breaching the confinement boundary, preventing 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.

B.1.3.1 Visual Inspection of Air Inlets and Outlets

Basis: The concrete temperature could exceed 350°F in the accident circumstances of complete blockage of all vents. Concrete temperatures over 350°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°F in 30 hours.

B.1.3.2 Exterior VCC Surface Inspection

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.

B.1.3.3 Interior VCC Surface Inspection

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