72-1015

NAC INTERNATIONAL

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October 17, 2000

U.S. Nuclear Regulatory Commission
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Attn: Document Control Desk

Subject: Docket No. 72-1015

Request for Amendment of the Certificate of Compliance for the NAC-UMS® Universal Storage System to Incorporate Nonfuel Components and Uncanistered Damaged Fuel Rods as Approved Contents

References:
1. Safety Analysis Report for the NAC-UMS® Universal Storage System, Revision 5, NAC International, September 29, 2000
2. Draft Certificate of Compliance for Amendment 1 to the NAC- UMS® Universal Storage System, United States Nuclear Regulatory Commission, October 2000

NAC International (NAC) herewith requests that Reference 2 (Draft Certificate of Compliance No. 1015 for the NAC-UMS® Universal Storage System) be amended to incorporate nonfuel components and uncanistered damaged fuel rods as approved contents. This amendment is being requested to support the Maine Yankee Decommissioning Project. These changes have been discussed previously with members of the NRC Spent Fuel Projects Office staff.

This submittal includes 10 copies of the request for amendment and the Revision UMSS-00L changed pages for the NAC-UMS® Universal Storage System Safety Analysis Report (SAR), which incorporates the requested amendment. The List of Effective Pages and the Master Table of Contents for the SAR are updated to incorporate the text and page changes in the body of the SAR.

The results of this amendment demonstrate the acceptability of the hypothetical release into the transportable storage canister (TSC) cavity of fuel debris from up to 24 fuel rods that are uncanned, i.e., not in Maine Yankee fuel cans. The amendment also evaluates and demonstrates the acceptability for dry storage of nonfuel components – Boronometer sources, Control Element Assembly (CEA) fingertips and a 24-inch segment of an in-core instrumentation (ICI) thimble. These two categories of "contents" are evaluated for the structural, thermal, shielding, and criticality disciplines, and the Technical Specifications have been revised to include these "contents." The transportable storage canister has been shown to be leaktight; therefore, containment need not be evaluated further.

In addition to the above changes, SAR Chapters 9 and 10 have been revised to remove the daily inspection requirements for the loaded vertical concrete casks (VCCs) and the required maintenance for the first storage system placed in service, since these requirements are addressed by LCO 3.1.6 in the Technical Specifications in Chapter 12. The Surveillance Requirement (SR) 3.1.6.1, which verifies the difference between the average air outlet temperature for the VCC and the ISFSI ambient

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temperature, will confirm the heat removal performance of the system and will identify any off-normal condition requiring closer inspection of a specific VCC (At the Maine Yankee site, the temperature measurements can be recorded and evaluated remotely for improved ALARA considerations). LCO 3.1.6 is revised to delete Action B.2.2 because it does not relate to restoring the VCC system operability. In Surveillance Requirement 3.2.1.1 of LCO 3.2.1, the words "containing fuel" are deleted to provide general applicability to GTCC canisters, as well as fuel canisters. Tables 12B2-6 and 12B2-7 are revised to properly categorize all of the fuel limits.

In accordance with NRC/NAC discussions, implementation of the NAC-UMS® Universal Storage System is a critical path item for successful completion of the decommissioning of the Maine Yankee site. Therefore, NAC requests that the NRC perform the technical review and complete the regulatory approval of this amendment during the second quarter of calendar year 2001 to support Maine Yankee's decommissioning schedule.

If you have any comments or questions, please contact me at (770) 447-1144.

Sincerely,

T. C. Thompson

Thomas C. Thompson
Director, Licensing
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Enclosure

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12412-SAR-002

DOCKET NO. 72-1015

UMS®

UNIVERSAL MPC SYSTEM®

SAFETY ANALYSIS REPORT

for the

UMS® UNIVERSAL STORAGE CASK

Amendment for

MAINE YANKEE ATOMIC POWER COMPANY

Site Specific Spent Fuel

October 2000 UMSS-00L

NAC INTERNATIONAL

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Table 1-1 Terminology

Universal Storage System	The storage component of the Universal MPC System (UMS®) designed by NAC for the storage and transportation of spent nuclear fuel.
Universal Transport Cask	The packaging consisting of a Universal Transport Cask body with a closure lid and energy-absorbing impact limiters. The Universal Transport Cask is used to transport a Transportable Storage Canister containing spent fuel. The cask body provides the primary containment boundary during transport.
Confinement System	The components of the Transportable Storage Canister intended to retain the radioactive material during storage.
Contents	Twenty-four PWR fuel assemblies, or fifty-six BWR fuel assemblies. The fuel assemblies may be configured as Site Specific Fuel. The fuel assemblies are contained in a Transportable Storage Canister.
Standard Fuel	Irradiated fuel assemblies with a burnup less than, or equal to, 45,000 MWD/MTU and having the same configuration as when originally fabricated consisting generally of the end fittings, fuel rods, guide tubes, and integral hardware. For BWR fuel, the channel is considered to be integral hardware. The design basis fuel characteristics and analysis are based on the standard fuel configuration.
Consolidated Fuel	A nonstandard fuel configuration in which the individual intact fuel rods from one or more fuel assemblies are placed in a single container or a lattice structure that is dimensionally similar to a fuel assembly.
Intact Fuel (Assembly or Rod) (Undamaged Fuel)	A fuel assembly or fuel rod with no fuel rod cladding defects, or with known or suspected fuel rod cladding defects not greater than pinhole leaks or hairline cracks.

Table 1-1 Terminology (Continued)

Damaged Fuel (Failed Fuel)	A fuel assembly or fuel rod with known or suspected cladding defects greater than pinhole leaks or hairline cracks.
High Burnup Fuel	A fuel assembly having a burnup between 45,000 and 50,000 MWD/MTU, which must be preferentially loaded in periphery positions of the basket. An intact high burnup fuel assembly in which no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and in which no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, as determined by measurement and statistical analysis, may be stored as intact fuel. High burnup fuel assemblies not meeting these cladding criteria are classified as damaged fuel.
Site Specific Fuel	Spent fuel configurations that are unique to a site or reactor due to the addition of other components or reconfiguration of the fuel assembly at the site. It includes fuel assemblies, which hold nonfuel-bearing components, such as control components or instrument and plug thimbles, or which are modified as required by expediency in reactor operations, research and development or testing. Modification may consist of individual fuel rod removal, fuel rod replacement of similar or dissimilar material or enrichment, the installation, removal or replacement of burnable poison rods, or containerizing damaged (failed) fuel. Site specific fuel includes irradiated fuel assemblies designed with variable enrichments and/or axial blankets, fuel that is consolidated and fuel that exceeds design basis fuel parameters.

Table 1-1 Terminology (Continued)

Maine Yankee Fuel Can	A specially designed stainless steel screened can sized to hold an intact fuel assembly, consolidated fuel, or damaged fuel. The can screens permit draining and drying, while precluding the release of gross particulates into the canister cavity. The Maine Yankee Fuel Can may only be loaded in a Class 1 Canister.
Uncanned Damaged Fuel	Fuel classified as damaged that is not loaded in a Maine Yankee Fuel Can. Fuel assemblies with up to a total of 24 fuel rods classified as damaged (in one or more fuel assemblies) may be loaded in each canister using one or more of the periphery fuel loading positions.
Transportable Storage Canister (Canister)	The stainless steel cylindrical shell, bottom end plate, shield lid, and structural lid that contain the fuel basket structure and the contents.
Shield Lid	A thick stainless steel disk that is located directly above the fuel basket. The shield lid comprises the first part of a double-welded closure system for the Transportable Storage Canister. The shield lid provides a containment/confinement boundary for storage and shielding for the contents.
- **Drain Port**	A penetration located in the shield lid to permit draining of the canister cavity.
- **Vent Port**	A penetration located in the shield lid to aid in draining and in vacuum drying and backfilling the canister with helium.
- **Port Cover**	The stainless steel covers that close the vent and drain ports, and that are welded in place following draining, drying, and backfilling operations.
- **Quick Disconnect**	The valved nipple used in the vent and drain ports to facilitate operations.

Table 1-1 Terminology (Continued)

Structural Lid	A thick stainless steel disk that is positioned on top of the shield lid and welded to the canister. The structural lid is the second part of a double-welded closure system for the Transportable Storage Canister. The structural lid provides a confinement boundary for storage, shielding for the contents, and canister lifting/handling capability.
Fuel Basket (Basket)	The structure located within the Transportable Storage Canister that provides structural support, criticality control, and primary heat transfer paths for the fuel assemblies.
- Support Disk	The primary lateral load-bearing component of the fuel basket. The PWR support disk is a circular stainless steel plate with 24 square holes machined in a symmetrical pattern. The BWR support disk is a circular carbon steel plate with 56 square holes machined in a symmetrical pattern. Each square hole is a location for a fuel tube.
- Heat Transfer Disk	A circular aluminum plate with 24 (PWR basket) or 56 (BWR basket) square holes machined in a symmetrical pattern. The heat transfer disk enhances heat transfer in the fuel basket.
- Fuel Tube	A stainless steel tube having a square cross-section with enclosed BORAL neutron poison material on its exterior surfaces. One fuel tube is inserted through each square hole in the support disks and heat transfer disks. Fuel assemblies are loaded into the fuel tube.
- Tie Rod	A stainless steel rod used to align, retain, and support the support disks and the heat transfer disks in the fuel basket structure. The tie rods extend from the top weldment to the bottom weldment.
- Spacer	Installed on the tie rod between the support disks (BWR only) or between the support disks and top and bottom weldments (BWR and PWR) to properly position the disks and provide axial support for the support disks.

Table 1-1 Terminology (Continued)

- Split Spacer	Spacers installed on the tie rod between the support disks and the heat transfer disks to properly position the disks and provide axial support for the support disks and the heat transfer disks.
Vertical Concrete Cask (Concrete Cask)	A concrete cylinder that contains the Transportable Storage Canister during storage. The Vertical Concrete Cask is formed around a steel inner liner and base and is closed by a shield plug and lid.
- Shield Plug	A thick carbon steel plug installed in the top end of the Vertical Concrete Cask to reduce skyshine radiation. The shield plug contains a 1-inch thick neutron shield.
- Lid	A thick carbon steel plate that serves as the bolted closure for the Vertical Concrete Cask. The lid precludes access to the canister and provides additional radiation shielding.
- Liner	A thick carbon steel shell that forms the annulus of the concrete cask. The liner serves as the inner form during concrete pouring and provides radiation shielding of the canister contents.
- Base	A carbon steel weldment that contains the air inlets, the concrete cask jacking points and the pedestal that supports the canister inside of the concrete cask.
Transfer Cask	A shielded lifting device for handling of the Transportable Storage Canister during loading of spent fuel, canister closure operations, and transfer of the canister into or out of the Vertical Concrete Cask during storage, or into or out of the Universal Transport Cask during transportation. The transfer cask incorporates bottom doors that permit the vertical loading of the storage and transport casks.

Table 1-1 Terminology (Continued)

- Transfer Cask Lifting Trunnions	Four low alloy steel trunnions used to lift and move the transfer cask.
Adapter Plate	A carbon steel plate assembly that attaches to the top of the transport or concrete cask to facilitate installation and alignment of the transfer cask. It also provides the operating mechanism for the transfer cask bottom doors.
NS-4-FR	A solid, borated, hydrogenous, synthetic, polymer material with neutron absorption capabilities, similar to those of borated water. Developed by BISCO Products, Inc., NS-4-FR is now supplied by Japan Atomic Power Company and its product licensees.
Air Pad Rig Set (Air Pallet)	A device used to lift the Vertical Concrete Cask by using high volume air.
Heavy Haul Trailer	The trailer used to transport the empty or loaded Vertical Concrete Cask.
Margin of Safety	An analytically determined value defined as the "factor of safety" minus 1. Factor of safety is also analytically determined, and is defined as the allowable stress or displacement of a material divided by its actual (calculated) value.

1.3.2.1 Maine Yankee Site Specific Spent Fuel

The configurations of Maine Yankee site specific fuel assemblies that have been evaluated and found to be acceptable contents are:

- Fuel assemblies with up to 176 fuel rods removed from the assembly lattice.
- Fuel assemblies with fuel rods replaced with stainless steel rods, solid Zircaloy rods or fuel rods enriched to 1.95 wt %.
- Fuel assemblies with burnable poison rods replaced with hollow Zircaloy tubes.
- Fuel assemblies that are variably enriched with a maximum fuel rod enrichment of 4.21 wt % ^{235}U and that also have a maximum planar average enrichment of 3.99 wt % ^{235}U.
- Fuel assemblies with variable enrichment and/or annular axial blankets.
- Fuel assemblies with a control element inserted.
- Fuel assemblies with an instrument thimble inserted in the center guide tube.
- Fuel assemblies with up to two fuel rods inserted in any or all of the guide tubes.
- Fuel assemblies with inserted non-fuel components, including startup sources.
- Consolidated fuel.
- Fuel assemblies having up to 100% of the rods damaged in each assembly.
- Fuel assemblies having a burnup of greater than 45,000 MWD/MTU but less than 50,000 MWD/MTU.

These site specific fuel configurations are evaluated against the limits established for the UMS® Storage System based on the design basis fuel. The site specific fuel is either shown to be bounded by the evaluation of the design basis fuel or is separately evaluated to establish limits which are maintained by preferential loading administrative controls. Where applicable to specific configurations, the preferential loading controls are described in Section 2.1.3.1.1. The preferential loading controls take advantage of design features of the UMS® Storage System to allow the loading of fuel configurations that may have higher burnup or additional hardware or fuel source material that is not specifically considered in the design basis fuel evaluation.

The Transportable Storage Canister loading procedures will indicate that the loading of a fuel configuration with removed fuel or poison rods, damaged or consolidated fuel in a Maine Yankee fuel can, or fuel with burnup greater than 45,000, but less than 50,000, MWD/MTU is administratively controlled in accordance with Section 2.1.3.1 and Table 2.1.3.1-1. As shown in the table, only one consolidated fuel lattice is loaded in any single canister. Preferential loading positions in the canister basket are shown in Figure 2.1.3.1-1.

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Table 1.5-1 NUREG-1536 Compliance Matrix (Continued)

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
1. Testing and Maintenance	a. The SAR must describe the applicant's program for preoperational testing and initial operations. [10 CFR 72.24(p)]	Section 9.1 presents the acceptance testing for the system.
	b. The cask design must permit maintenance as required. [10 CFR 72.236(g)]	Section 9.2 presents the maintenance activities for the system.
	c. Structures, systems, and components (SSCs) important to safety must be designed, fabricated, erected, tested, and maintained to quality standards commensurate with the importance to safety of the function they are intended to perform. [10 CFR 72.122(a), 10 CFR 72.122(f), 10 CFR 72.128(a)(1), and 10 CFR 72.24(c)]	The acceptance tests and maintenance activities presented in Sections 9.1 and 9.2 are performed to verify compliance with the design bases and criteria, and that the system continues to perform as designed.
	d. The applicant or licensee must establish a test program to ensure that all required testing is performed to meet applicable requirements and acceptance criteria. In addition, at least 30 days before the receipt of spent fuel, the licensee must submit to the NRC a report concerning the pre-operational test acceptance criteria and test results. [10 CFR 72.162 and 10 CFR 72.82(e)]	The testing and maintenance provided in Sections 9.1 and 9.2 are intended to be used by an ISFSI user in the development of site-specific programs.
	e. The applicant or licensee must evaluate the cask and its systems important to safety, using appropriate tests or other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions. [10 CFR 72.236(l)]	The acceptance tests presented in Section 9.1 demonstrate that the system will maintain confinement of the spent fuel under normal, off-normal, and accident conditions.
	f. The applicant or licensee must inspect the cask to ascertain that there are no cracks, pinholes, uncontrolled voids, or other defects that could significantly reduce confinement effectiveness. [10 CFR 72.236(j)]	As described in Section 9.1.1, the canister is visually and non-destructively examined prior to use.
	g. The applicant must perform, and make provisions that permit the Commission to perform, tests that the Commission deems necessary or appropriate. [10 CFR 72.232(b)]	Provisions shall be made, as necessary, to facilitate additional NRC imposed testing as required.
	h. The general licensee must accurately maintain the record provided by the cask supplier showing any maintenance performed on each cask. This record must include evidence that any maintenance and testing have been conducted under an NRC-approved quality assurance (QA) program. [10 CFR 72.212(b)(8)]	Records of maintenance activities would be maintained by the ISFSI user, and thus are not applicable.
	The applicant or licensee must assure that the casks are conspicuously and durably marked with a model number, unique identification number, and the empty weight. [10 CFR 72.236(k)]	As specified in Section 9.1.8, each system is to be marked with the model number, unique cask number, empty weight, and additional information.

Table 1.5-1 NUREG-1536 Compliance Matrix (Continued)

Chapter 9 – Acceptance Test and Maintenance Program		
Area	**Regulatory Requirement**	**Description of Compliance**
2. Resolution of Issues Concerning Adequacy or Reliability	The SAR must identify all SSCs important to safety for which the applicant cannot demonstrate functional adequacy and reliability through previous acceptable evidence. For this purpose, acceptable evidence may be established in any of the following ways: • prior use for the intended purpose • reference to widely accepted engineering principles • reference to performance data in related applications In addition, the SAR should include a schedule showing how the applicant or licensee will resolve any associated safety questions before the initial receipt of spent fuel. [10 CFR 72.24(i)]	As described in Sections 3.1 and 3.3, the design of the system is based on industry standard codes and standards for materials and margins of safety. The acceptance tests specified in Section 9.1 are performed to demonstrate the adequacy of each fabricated system in accordance with applied Codes and Standards. The system does not rely on any materials or design standards that lack acceptable evidence of functional adequacy.
3. Cask Identification	The applicant or licensee must conspicuously and durably mark the cask with a model number, unique identification number, and empty weight. [10 CFR 72.236(k)]	As specified in Section 9.1.8, each system is to be marked with the model number, unique cask number, empty weight, and additional information.
Confinement System	American Society of Mechanical Engineers (ASME), "Boiler and Pressure Vessel (B&PV) Code," Section III, Subsection NB or NC "American National Standard for Radioactive Materials— Leakage Tests on Packages for Shipment" (ANSI N14.5-1987)	As specified in Section 3.1.2, the canister is designed in accordance with the ASME Code, Section III, Subsection NB. Exceptions to the Code are provided in Appendix 12B, Table 12B3-1. The confinement system is leak tested in accordance with ANSI N14.5 following shield lid welding as specified in Appendix 12A, LCO 3.1.5.
Confinement Internals (e.g., basket)	ASME B&PV Code, Section III, Subsection NG	As specified in Section 3.1.2, the basket structure is designed in accordance with the ASME Code, Section III, Subsection NG.
Metal Cask Overpack	ASME B&PV Code, Section VIII	Not applicable.
Concrete Cask Overpack	American Concrete Institute (ACI) Standards 318 and 349, as appropriate	As stated in Section 3.1.2, the VCC is designed in accordance with ACI-349 and ANSI/ANS-57.9.
Other Metal Structures	ASME B&PV Code, Section III, Subsection NF American Institute of Steel Construction (AISC), "Manual of Steel Construction"	Not applicable.

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2.1 Spent Fuel To Be Stored

The Universal Storage System is designed to safely store up to 24 PWR spent fuel assemblies, or up to 56 BWR spent fuel assemblies, contained within a Transportable Storage Canister. Each canister is specifically designed to accommodate one of three classes of PWR fuel assemblies or one of two classes of BWR fuel assemblies. The classification of the fuel assemblies is based primarily on fuel assembly length and cross section. The classes of major fuel assemblies to be stored in the Vertical Concrete Cask and their characteristics are shown in Tables 2.1.1-1 (PWR) and 2.1.2-1 (BWR). The minimum initial enrichment limits are shown in Tables 2.1.1-3 and 2.1.2-3 for PWR and BWR fuel, respectively. The minimum enrichment limits exclude the loading of fuel assemblies enriched to less than 1.9 wt.% ^{235}U, including unenriched fuel assemblies, into the Transportable Storage Canister. The loading of unenriched fuel assemblies into the Transportable Storage Canister is not evaluated and is not authorized. However, BWR assemblies with unenriched axial blankets may be loaded into the canister. Any empty fuel rod position must be filled with a solid filler rod fabricated from either Zircaloy or Type 304 stainless steel. Fuel assembly parameter limits established for the design basis fuel analysis may be exceeded by site specific fuel assemblies. These site specific fuel assembies are separately evaluated, or shown to be bounded by the design basis fuel, as described in Section 2.1.3.

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2.1.3 Site Specific Spent Fuel

The UMS® Storage System design basis PWR fuel assemblies are described in Section 2.1.1. Four different assembly arrays: 14 x 14, 15 x 15, 16 x 16 and 17 x 17, produced by several different fuel vendors, were evaluated in the determination of the PWR design basis fuel.

The design basis BWR fuel assemblies are described in Section 2.1.2. Three different arrays: 7 x 7, 8 x 8 and 9 x 9, produced by several different fuel vendors were evaluated in the determination of the UMS® BWR design basis fuel.

This section describes site specific spent fuel, i.e., fuel assemblies that are configured differently or that have different fuel parameters, such as enrichment or burnup, than the design basis fuel assemblies. The site specific fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, testing programs intended to improve reactor operations or from the insertion of control components or other items within the fuel assembly.

A summary description of the site specific spent fuels is presented in Section 1.3.2. The site specific spent fuel configurations are either shown to be bounded by the design basis fuel analysis or are separately evaluated. Unless specifically excepted, site specific spent fuel must also meet the conditions specified for the design basis fuel presented in Section 1.3.1.

2.1.3.1 Maine Yankee Site Specific Spent Fuel

The Maine Yankee site specific spent fuel assemblies are categorized as intact (undamaged) or damaged as defined in Table 1-1. Generally, damaged fuel and certain undamaged fuel configurations are placed in a Maine Yankee fuel can for storage in the Transportable Storage Canister; however, under certain conditions, fuel classified as damaged may be loaded as intact fuel (Section 2.1.3.1.7).

The configurations of Maine Yankee site specific spent fuel that have been evaluated and found to be acceptable contents are summarized in Section 1.3.2.1, and include those standard fuel assembly configurations, which were modified by the installation or removal of fuel or non fuel-bearing components.

The three principal types of these modifications are:

- The removal of fuel rods without replacement.
- The replacement of removed fuel rods or burnable poison rods with rods of another material, such as stainless steel, or with fuel rods of a different enrichment.
- The insertion of control elements, non-fuel items including startup sources, or instrument or plug thimbles, in guide tube positions.

Site specific spent fuel also includes fuel assemblies that are uniquely designed to support reactor physics. These fuel assemblies include those that are variably enriched or that are variably enriched with annular axial blankets. Generally, these fuel assemblies (described in Sections 6.6.1.2.2 and 6.6.1.2.3) are bounded by the evaluation of the design basis fuel.

As described in Section 2.1.3.1.6, certain of the site specific spent fuel configurations, including damaged and consolidated fuel loaded in Maine Yankee fuel cans, must be preferentially loaded in corner positions of the fuel basket. In addition, certain of the site specific fuel has experienced burnup that exceeds the 45,000 MWD/MTU design basis burnup. The thermal evaluation of these fuel assemblies is presented in Section 4.5.1. The results of that evaluation show that a fuel assembly with a burnup between 45,000 and 50,000 MWD/MTU must be preferentially loaded in peripheral fuel positions in the basket.

2.1.3.1.1 Damaged Fuel Lattices

There are two lattices for damaged fuel rods in the current Maine Yankee fuel inventory, designated CF1 and CA3, that are loaded in Maine Yankee fuel cans. CF1 is a lattice having roughly the same dimensions as a standard fuel assembly. It is a 9 x 9 array of tubes, some of which contain damaged fuel rods. CA3 is a previously used fuel assembly lattice that has had all of the rods removed, and into which, damaged fuel rods have been inserted. The CF1 and CA3 lattices are placed in a Maine Yankee fuel can for storage. No credit is taken for the lattice structures in the criticality, structural, or thermal analysis.

2.1.3.1.2 Maine Yankee Consolidated Fuel

The Maine Yankee fuel inventory includes two consolidated fuel lattices, which house intact fuel rods taken from three fuel assemblies. Each lattice is a 17x17 array formed using stainless steel

grids and top and bottom stainless steel end fittings. Four solid stainless steel connector rods connect the end fittings. The top end fitting is designed so that the lattice can be handled by the standard fuel assembly lifting fixture (grapple). These lattices were not used in the reactor and the stainless steel hardware is not activated.

One of these lattices contains 283 fuel rods and 2 rod position vacancies. The other contains 172 fuel rods, with the 76 stainless steel dummy rods in the outer periphery of the lattice.

The consolidated fuel is placed in a Maine Yankee fuel can for storage. No credit is taken for the lattice structures in the criticality, structural, or thermal analysis.

2.1.3.1.3 Maine Yankee Spent Fuel with Inserted Integral Hardware or Non-Fuel Items

Certain Maine Yankee fuel assemblies have either a Control Element Assembly or an Instrument Thimble inserted in the fuel assembly. These components add to the gamma radiation source term of the standard fuel assembly.

A Maine Yankee Control Element Assembly (CEA) consists of five control rods mounted on a Type 304 stainless steel spider assembly. The five control rods are inserted in the fuel assembly guide tubes when the CEA is inserted in the fuel assembly. When fully inserted, the control element spider rests on the fuel assembly upper end fitting. The rods are fabricated from Inconel 625 or stainless steel and encapsulate B_4C as the primary neutron poison material. Fuel assemblies with a control element installed must be loaded into a Class 2 canister because of the additional height that the control element spider adds to the fuel assembly overall length.

Some standard fuel assemblies have an in-core instrument (ICI) thimble inserted in the center guide tube of the fuel assembly. The detector material and lead wire have been removed from the thimble assembly. The thimble top end and tube are primarily Zircaloy. When installed, the instrument thimble does not add to the overall fuel assembly length. Consequently, fuel assemblies with instrument thimbles are loaded in the Class 1 canister.

The non-fuel inventory includes a segment of an ICI instrument thimble approximately 24 inches long. This segment is loaded in the corner guide tube position of an intact fuel assembly. The fuel assembly with the ICI segment installed must have a CEA flow plug installed to close the top of the corner guide tube, capturing the segment between the CEA flow plug and the bottom

end plate of the fuel assembly. The ICI segment may be installed in a fuel assembly that also holds CEA finger tips in other corner guide tube positions. Because of the CEA fuel plug, the fuel assembly must be installed in a Class 2 canister.

The non-fuel inventory also includes five startup sources and one Boronometer source. One of the startup sources is unirradiated.

The Boronometer source contains 16 grams of plutonium and 8 grams of beryllium that are double encapsulated within a Type 304 stainless steel sealed tube. The tube is approximately 3.5 inches in length and 0.5 inches in diameter. The Boronometer source is loaded in the corner guide tube position of an intact fuel assembly. This fuel assembly must also have a CEA flow plug installed to close the top of the corner guide tube, capturing the Boronometer source between the CEA flow plug and the bottom end plate of the fuel assembly. The Boronometer source may not be installed in a fuel assembly that also holds a startup source, but may be installed in a fuel assembly that also holds the CEA finger tips and/or ICI segment.

The startup sources include three Pu-Be sources and two Sb-Be sources that are installed in the center guide tubes of fuel assemblies that subsequently must be loaded in one of the four corner fuel positions of the basket. Each source is designed to fit in the center guide tube of an assembly, and only one startup source may be loaded in any fuel assembly. All five of these startup sources contain Sb-Be pellets, which are 50% Be by volume. One of the three Pu-Be sources is unirradiated and evaluation of this source is based on a "fresh" source material assumption.

2.1.3.1.4 Maine Yankee Spent Fuel with Unique Design

Certain Maine Yankee fuel assemblies were uniquely designed to accommodate reactor physics. These assemblies incorporate variable radial enrichment and axial blankets.

Two batches of fuel used at Maine Yankee contain variably enriched fuel rods. The maximum fuel rod enrichment of one batch is 4.21 wt % ^{235}U with the variably enriched rods enriched to 3.5 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.99 wt % ^{235}U. For the other batch, the maximum fuel rod enrichment is 4.0 wt % ^{235}U, with the variably enriched rods enriched to 3.4 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.92 wt % ^{235}U.

One batch of variably enriched fuel also incorporates axial end blankets with fuel pellets that have a center hole, referred to as annular fuel pellets. Annular fuel pellets are used in the top and bottom 5% of the active fuel length of each fuel rod in this batch.

2.1.3.1.5 Maine Yankee Fuel Can

Fuel assemblies classified as damaged that exceed the limits for loading as intact fuel and certain undamaged fuel configurations are loaded in a Maine Yankee fuel can. The Maine Yankee fuel can is shown in Drawings 412-501 and 412-502 (Section 1.8). The fuel can may be loaded only in a corner position in the basket of a Class 1 canister. The fuel can analysis assumes the failure of 100 % of the fuel rods held in the fuel can.

The fuel can is sized to accommodate a fuel assembly and must be loaded in a corner position of the fuel basket. As shown in the drawings, the can is 162.8 inches in length and has an external square dimension of 8.62 inches and an internal square dimension of 8.52 inches. In the top 4.5 inches the external square dimension is 8.82 inches. The fuel can is closed on the bottom end by a 0.63-inch thick plate that is welded to the can shell. The plate has drilled holes in each corner to allow water to drain from the can. A screen covers the holes to preclude the release of gross particulates from the fuel can. A lid having an overall depth dimension of 2.38 inches closes the can. The lid is not secured to the can shell, but is held in place when the shield lid is installed in the canister. The lid also has four drilled and screened holes. The damaged fuel is inserted in the fuel can and the lid is installed. Slots in the can shell allow the loaded can to be lifted and installed in the basket. Alternately, the fuel can may be inserted in a basket corner position before the damaged fuel assembly is inserted in the fuel can. Since the fuel can lid is held in place by the canister shield lid, the fuel can may be used only in the Class 1 canister.

A Maine Yankee fuel can containing fuel debris with greater than 20 Curies of plutonium, requires double containment for transport conditions in accordance with 10 CFR 71.63 (b).

The Maine Yankee fuel can design and fabrication specification summary is provided in Table 2.1.3.1-2. The major physical design parameters of the Maine Yankee fuel can are provided in Table 2.1.3.1-3. The structural evaluation of the Maine Yankee site specific fuel configurations is provided in Section 3.6.1. As shown in Section 4.5.1, the maximum allowable heat load for the contents of a Maine Yankee fuel can is 0.958 kW.

2.1.3.1.6 Maine Yankee Site Specific Spent Fuel Preferential Loading

The estimated Maine Yankee site specific spent fuel inventory is shown in Table 2.1.3.1-1. (Note that the population of fuel in a given configuration may change based on future spent fuel inspection or survey.) As shown in this table, certain fuel configurations are preferentially loaded to take advantage of the design features of the Transportable Storage Canister and basket to allow the loading of fuel that does not specifically conform to the design basis spent fuel. The designated preferential loading positions are shown in Figure 2.1.3.1-1. The corner positions are designated by the letter "C." These positions are used primarily for the loading of fuel with missing fuel rods, fuel with fuel rods that have been replaced by rods of other material, for consolidated fuel lattices, and damaged fuel loaded in the Maine Yankee fuel can. The requirements for preferential loading schemes using the corner positions result primarily from shielding or criticality evaluations of the designated fuel configurations.

Maine Yankee consolidated fuel is loaded in a Maine Yankee fuel can and is, therefore, designated for a corner position. Preferential loading is also used for spent fuel having a burnup between 45,000 and 50,000 MWD/MTU. This fuel is assigned to peripheral locations designated by the letter "P" in Figure 2.1.3.1-1. The thermal analysis supporting the use of these locations for higher burnup fuel is presented in Section 4.5.1. As described in that section, the interior locations must be loaded with fuel that has lower burnup and/or longer cool times in order to maintain the design basis heat load and component temperature limits. Loading tables, which provide the limits for decay heat on a per assembly basis, are provided in Section 4.5.1.

High burnup fuel (45,000 – 50,000 MWD/MTU) may be loaded as intact fuel provided that for a given fuel assembly, the cladding oxide layer on no more than 1% of the fuel rods has a peak thickness greater than 80 microns and no more than 3% of the fuel rods have a peak oxide layer thickness greater than 70 microns. The high burnup fuel is classified as failed fuel if the cladding oxide layer criteria are not met, or if the oxide layer is detached or spalled from the cladding. Since the transportable storage canister is tested to be leak tight, no additional confinement analysis is required for the high burnup fuel.

Fuel assemblies with a control element inserted will be loaded in a Class 2 canister and basket for storage and transport due to the increased length of the assembly with the control element installed. However, these assemblies are not restricted as to loading position within the basket.

Fuel assemblies with a startup source in the center guide tube position or the Boronometer source in a corner guide tube position must be loaded in one of the basket corner positions. A fuel assembly may not hold more than one startup source and may not hold both a startup and a Boronometer source.

The loading position of fuel assemblies holding the CEA finger tips and/or the ICI segment in a fuel assembly corner guide tube position is not controlled; however, these fuel assemblies must have a CEA flow plug to ensure these items are captured within the guide tube(s).

2.1.3.1.7 Uncanned Damaged Fuel

The storage system has been evaluated to allow the storage of up to 24 fuel rods, in one or more fuel assemblies, classified as damaged (failed) that are not stored in the Maine Yankee fuel can. Subject to the preferential loading controls, this fuel may be stored in any basket periphery fuel position as intact fuel. The 24 fuel rods classified as damaged may be in a single fuel assembly, or be individual fuel rods in up to 12 fuel assemblies (the number of peripheral fuel positions). The damaged fuel classification may arise from the existence of greater than hairline cracks, pinhole leaks, or the cladding oxide layer thickness for high burnup fuel.

2.1.3.1.8 Maine Yankee High Burnup Fuel

There are ninety (90) Maine Yankee fuel assemblies that have achieved a burnup between 45,000 and 50,000 MWD/MTU. As described in Section 2.1.3.1.6, these fuel assemblies are preferentially loaded in peripheral locations in the basket. The high burnup assemblies are similar to the other Maine Yankee fuel planned to be placed in dry storage (i.e., those with burnup less than 45,000 MWD/MTU), but have design differences that support the high burnup objective.

The Combustion Engineering 14 x 14 high burnup fuel assemblies incorporate a lower (fuel rod) internal pressure than the UMS design basis fuel, which results in lower cladding stress throughout their reactor and storage life, and a greater cladding thickness. The greater cladding thickness, together with a larger fuel rod diameter, provide additional margin against regulatory limits. Some of the fuel assemblies have a "low tin" Zircaloy cladding, which results in lower hydrogen pick-up in the cladding and a lower cladding oxide layer thickness.

Publicly available DOE-sponsored research studies on high burnup fuel have measured irradiated Zircaloy material properties. These studies show that even at burnups over 50,000 MWD/MTU, Zircaloy cladding has adequate material strength and ductility to maintain fuel rod integrity throughout all conditions of storage. The technical details of the DOE sponsored research studies include hot cell examination of spent fuel from the Fort Calhoun [22] and Oconee [23] reactors.

The published reports conclude that there is an increase in the yield and ultimate strengths, and a decrease in ductility of the high burnup fuel rod Zircaloy cladding with an oxide layer thickness less than or equal to 80 microns. The Fort Calhoun and Oconee fuel rods examined had maximum burnup up to 55,700 and 54,800 MWD/MTU, respectively. Localized burnup of these fuel rods reached over 60,000 MWD/MTU. The burnups encompass the burnups of the Maine Yankee high burnup fuel. Tables 17, 18 and 19 of the Fort Calhoun report (DOE/ET/34030-11) demonstrate that, at the respective burnups, there is a significant increase in the yield and ultimate strengths of the Zircaloy cladding with a corresponding decrease in the material ductility (plastic strain). This is further confirmed in the Oconee fuel examination report (DOE/ET/34212-50) in Table 20. These studies show that the Zircaloy material property changes occur during the early stages of irradiation and do not change significantly during the higher burnup periods. The Fort Calhoun and Maine Yankee fuels are essentially identical and are fabricated by the same supplier (Combustion Engineering). Therefore, it is concluded that the Maine Yankee high burnup fuel (45,000 < Burnup < 50,000 MWD/MTU) Zircaloy cladding ultimate and yield strengths are greater than those of standard burnup fuel assemblies, while maintaining adequate material ductility to perform its design functions.

The Maine Yankee high burnup fuel assemblies were fabricated according to their respective fuel specifications without any discrepancies or deviations that affected cladding. Review of Plant Operating Data demonstrates that the fuel has not been subjected to any unanalyzed events that could potentially lead to excessive cladding stress.

Review of fuel inspection records and video tapes of the Maine Yankee high burnup fuel assemblies shows that the fuel is essentially identical to fuel that is burned less than 45,000 MWD/MTU, with no evidence of damage or excessive cladding oxidation.

The supporting data and information demonstrates that the physical and mechanical characteristics of the Maine Yankee high burnup fuel assemblies (45,000 < Burnup < 50,000 MWD/MTU) are essentially identical to those of the fuel assemblies with burnup less than 45,000 MWD/MTU.

Figure 2.1.3.1-1 Preferential Loading Diagram for Maine Yankee Site Specific Spent Fuel

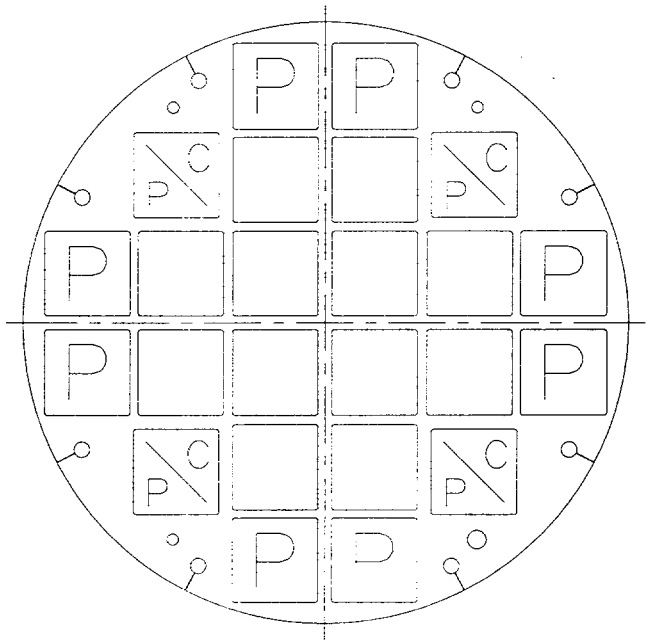

Table 2.1.3.1-1 Maine Yankee Site Specific Fuel Population

Site Specific Spent Fuel Configurations[1]	**Est. Number of Assemblies**[2]
Standard Fuel	1,434
Inserted Control Element Assembly (CEA)	168
Inserted In-Core Instrument (ICI) Thimble	138
Consolidated Fuel	2
Fuel Rod Replaced by Rod Enriched to 1.95 wt %	3
Fuel Rod Replaced by Stainless Steel Rod or Zircaloy Rod	18
Fuel Rods Removed	10
Variable Enrichment	72
Variable Enrichment and Axial Blanket	68
Burnable Poison Rod Replaced by Hollow Zircaloy Rod	80
Damaged Fuel in Maine Yankee Fuel Can	12
Burnup between 45,000 and 50,000 MWD/MTU	90
Maine Yankee Fuel Can	As Required
Uncanned Damaged Fuel	--
Inserted Startup Source	4
Inserted Boronometer Source	1
Inserted CEA Fingertip or ICI String Segment	1

1. The loading of the site specific fuel is controlled by the requirement of Section 12B2 of the Technical Specifications presented in Chapter 12.
2. The number of fuel assemblies in some categories may vary depending on future fuel inspections.

Table 2.1.3.1-2 Maine Yankee Fuel Can Design and Fabrication Specification Summary

Design

- The Maine Yankee Fuel Can shall be designed in accordance with ASME Code, Section III, Subsection NG except for: 1) the noted exceptions of table 12B3-1 for fuel basket structures; and 2) the Maine Yankee Fuel Can may deform under accident conditions of storage.
- The Maine Yankee Fuel Can will have screened vents in the lid and base plate. Stainless steel meshed screens (250x250) shall cover all openings.
- The Maine Yankee Fuel Can shall limit the release of material from damaged fuel assemblies and fuel debris to the canister cavity.
- The Maine Yankee Fuel Can lifting structure and lifting tool shall be designed with a minimum factor of safety of 3.0 on material yield strength.

Materials

- All material shall be in accordance with the referenced drawings and meet the applicable ASME Code sections.
- All structural materials are ASME SA 240, Type 304 stainless steel.

Welding

- All welds shall be in accordance with the referenced drawings.
- The final surface of all welds shall be liquid penetrant examined in accordance with ASME Code Section V, Article 6, with acceptance in accordance with ASME Code, Section NG-5350.

Fabrication

- All cutting, welding, and forming shall be in accordance with ASME Code Section III, NG-4000.

Acceptance Testing

- The Maine Yankee Fuel Can (first unit) and handling tool shall be load tested and visually inspected at the completion of fabrication.

Quality Assurance

- The Maine Yankee Fuel Can shall be constructed under a quality assurance program that meets 10 CFR 72 Subpart G. The quality assurance program must be accepted by NAC International and the licensee prior to initiation of the work.
- A Certificate of Conformance (or Compliance) shall be issued by the fabricator stating that the component meets the specifications and drawings.

Table 2.1.3.1-3 Major Physical Design Parameters of the Maine Yankee Fuel Can

Parameter	**Value**
Overall Length (in.)	162.8
Inside Cross Section (in.)	8.52 x 8.52
Outside Cross Section (in.) (1)	8.62 x 8.62
Can Wall Thickness	18 Gauge (0.048 in.)
Internal Cavity Length (in.)	160.0
Empty Weight (nominal) (lbs.)	130

Note (1) Outside cross section of Maine Yankee Fuel Can upper structure is 8.82 x 8.82 in. at top (4.5 in.) for lid engagement and fuel can lifting. This upper structure is located above the top weldment plate of the fuel basket assembly.

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3.6 Structural Evaluation of Site Specific Spent Fuel

This section presents the structural evaluation of fuel assemblies or configurations, which are unique to specific reactor sites or which differ from the UMS® Storage System design basis fuel. These site specific configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel that is classified as damaged. Damaged fuel includes fuel rods with cladding that exhibit defects greater than pinhole leaks or hairline cracks.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation.

3.6.1 Structural Evaluation of Maine Yankee Site Specific Spent Fuel for Normal Operating Conditions

This section describes the structural evaluation for site specific spent fuel configurations. As described in Sections 1.3.2.1 and 2.1.3.1, the inventory of site specific spent fuel configurations includes fuel classified as intact, intact with additional fuel and nonfuel-bearing hardware, consolidated fuel and fuel classified as damaged. Two categories of damaged fuel are considered:

- Up to 24 fuel rods in each canister classified as damaged that are not in Maine Yankee fuel cans.
- Damaged fuel that is separately containerized in Maine Yankee fuel cans.

These configurations are evaluated to ensure that they are bounded by the design basis fuel assembly analysis.

3.6.1.1 Maine Yankee Intact Spent Fuel

The description for Maine Yankee site specific fuel is in Section 1.3.2.1. The standard spent fuel assembly for the Maine Yankee site is the Combustion Engineering (CE) 14 x 14 fuel assembly.

Fuel of the same design has also been supplied by Westinghouse and by Exxon. The standard 14 x 14 fuel assemblies are included in the population of the design basis PWR fuel assemblies for the UMS® Storage System (see Table 2.1.1-1). The structural evaluation for the UMS® transport system loaded with the standard Maine Yankee fuels is bounded by the structural evaluations in Chapter 3 for normal conditions of storage and Chapter 11 for off-normal and accident conditions of storage.

With the Control Element Assembly (CEA) inserted, the weight of a standard CE 14 x 14 fuel assembly is 1,360 pounds. This weight is bounded by the weight of the design basis PWR fuel assembly (37,608/24 = 1,567 lbs) used in the structural evaluations (Table 3.2-1). The fuel configurations with removed fuel rods, with fuel rods replaced by solid stainless steel or Zircaloy rods, or with poison rods replaced by hollow Zircaloy rods, all weigh less than the standard CE 14 x 14 fuel assembly. The configuration with instrument thimbles installed in the center guide tube position weighs less than the standard assembly with the installed control element assembly. Consequently, this configuration is also bounded by the weight of the design basis fuel assembly. Since the weight of any of these fuel assembly configurations is bounded by the design basis fuel assembly weight, no additional analysis of these configurations is required.

The two consolidated fuel lattices are each constructed of 17 x 17 stainless steel fuel grids and stainless steel end fittings, which are connected by 4 stainless steel support rods. One of the consolidated fuel lattices has 283 fuel rods with 2 empty positions. The other has 172 fuel rods, with the remaining positions either empty or holding stainless steel rods. The calculated weight for the heaviest of the two consolidated fuel lattices is 2,100 pounds. Only one consolidated fuel lattice can be loaded into any one canister. The weight of the site specific 14 x 14 fuel assembly plus the CEA is approximately 1,360 lbs. Twenty-three (23) assemblies (at 1,360 lbs each) in addition to the consolidated fuel assembly (at approximately 2,100 lbs) would result in a total weight of 33,380 pounds.

Each canister may contain up to a total of 24 fuel rods in one or more fuel assemblies that are classified as failed and which are loaded into the canister as intact fuel. It is assumed that the fuel contents of the 24 fuel rods are allowed to escape the fuel cladding as "debris" and the debris is dispersed inside of the canister. Since the canister is in the vertical orientation for normal conditions of storage, the debris is considered to be located at the bottom of the canister. The total weight of the fuel debris from 24 fuel rods is less than 170 pounds. Therefore, the fuel debris is considered to have a negligible effect on the structural performance of the basket or canister.

Therefore, the design basis UMS® PWR fuel weight of 37,608 lbs bounds the site specific fuel and consolidated fuel by 12%. The evaluations for the Margin of Safety for the dead weight load of the fuel and the lifting evaluations in Section 3.4.4 bound the Margins of Safety for the Maine Yankee site specific fuel.

3.6.1.2 Maine Yankee Damaged Spent Fuel

The Maine Yankee fuel can, shown in Drawings 412-501 and 412-502, is provided to accommodate Maine Yankee damaged fuel. The fuel can fits within a standard PWR basket fuel tube. The primary function of the Maine Yankee fuel can is to confine the fuel material within the can to minimize the potential for dispersal of the fuel material into the canister cavity volume.

The Maine Yankee fuel can is designed to hold an intact fuel assembly, a damaged fuel assembly, a fuel assembly with a burnup between 45,000 and 50,000 MWD/MTU and having a cladding oxidation layer thickness greater than 80 microns, or consolidated fuel in the Maine Yankee fuel inventory.

The fuel can is a square cross-section tube made of Type 304 stainless steel with a total length of 162.8 inches. The can walls are 0.048-inch-thick sheet (18 gauge). The minimum internal width of the can is 8.52 inches. The bottom of the can is a 0.63-inch-thick plate. Four holes in the plates, screened with a Type 304 stainless steel wire screen (250 openings/inch x 250 openings/inch mesh), permit water to be drained from the can during loading operations. Since the bottom surface of the fuel can rests on the canister bottom plate, additional slots are machined in the fuel can (extending from the holes to the side of the bottom assembly) to allow the water to be drained from the can. At the top of the can, the wall thickness is increased to 0.15-inches to permit the can to be handled. Slots in the top assembly side plates allow the use of a handling tool to lift the can and contents. To confine the contents within the can, the top assembly consists of a 0.88-inch-thick plate with screened drain holes identical to those in the bottom plate. Once the can is loaded, the can and contents are inserted into the basket, where the can may be supported by the sides of the fuel assembly tube, which are backed by the structural support disks. Alternately, the empty fuel can may be placed in the basket prior to having the designated contents inserted in the fuel can.

In normal operation, the can is in a vertical position. The weight of the fuel can contents is transferred through the bottom plate of the can to the canister bottom plate, which is the identical load path for intact fuel. The only loading in the vertical direction is the weight of the can and the top assembly. The lifting of the can with its contents is also in the vertical direction.

Classical hand calculations are used to qualify the stresses in the Maine Yankee fuel can.

A conservative bounding temperature of 600°F is used for the evaluation of the fuel can for normal conditions of storage. A temperature of 300°F is used for the lifting components at the top of the fuel can and for the lifting tool.

Calculated stresses are compared to allowable stresses in accordance with ASME Code, Section III, Subsection NG. The ASME Code, Section III, Subsection NG allowable stresses used for stress analysis are:

Property	600°F	300°F
S_u	63.3 ksi	66.0 ksi
S_y	18.6 ksi	22.5 ksi
S_m	16.7 ksi	20.0 ksi
E	25.2×10^3 ksi	27.0×10^3 ksi

The Maine Yankee fuel can is evaluated for dead weight and handling loads for normal conditions of storage. Since the can is not restrained, it is free to expand. Therefore, the thermal stress is considered to be negligible.

The Maine Yankee fuel can lifting components and handling tools are designed with a safety factor of 3.0 on material yield strength.

3.6.1.2.1 Dead Weight and Handling Loading Evaluation

The weight of the Maine Yankee fuel can is 130 lbs. The maximum compressive stress acting in the tube of the fuel can is due to its own weight in addition to that of the top assembly. A 10% dynamic load factor is applied to the fuel can weight for an applied load of 143 pounds to account for loads due to handling. Based on the minimum cross sectional area of $(8.62)^2 - (8.52)^2 = 1.714$ in^2, the margin of safety at 300°F is:

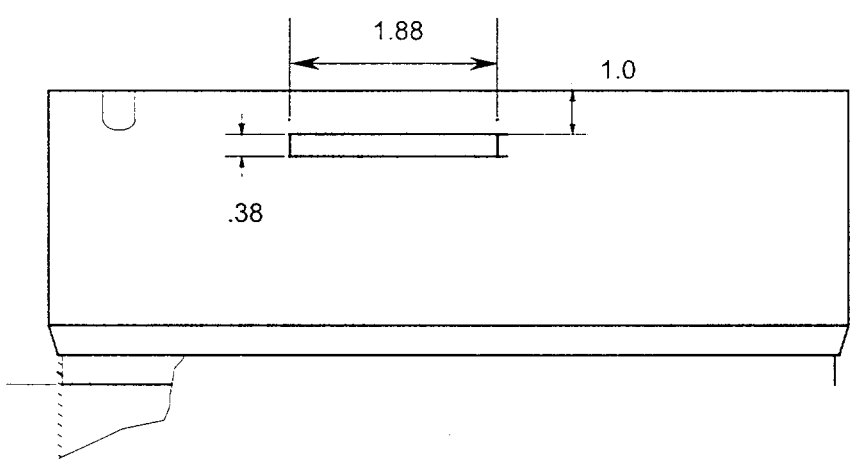
M.S. = 20,000/(143/1.714) – 1
M.S. = + LARGE

3.6.1.2.2 Lifting Evaluation

Based on the loaded weight of the fuel can, the lift evaluation does not require the use of the design criteria of ANSI N14.6 or NUREG-0612. However, for purposes of conservatism and good engineering practice, a factor of safety of three on material yield strength is used for the stress evaluations for the lift condition. Since a combined stress state results from the loading and the calculated stresses are compared to material yield strength, the Von Mises stress is computed.

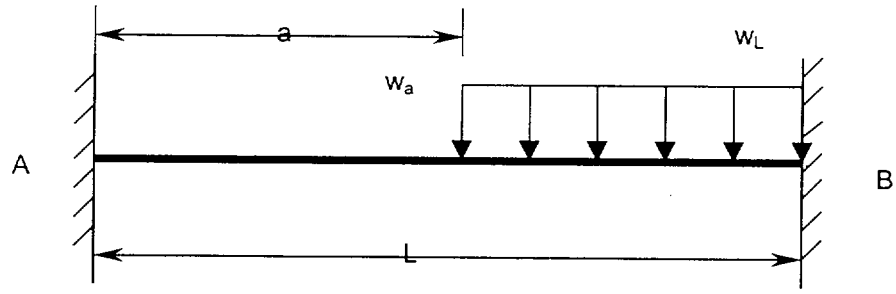
Side Plates

The side plates will be subjected to bending, shear and bearing stresses because of interaction with the lifting tool during handling operations. The lifting tool engages the 1.875-inch × 0.38-inch lifting slots with lugs that are 1-inch wide and lock into the four lifting slots. For this evaluation, the handling load is the weight of the consolidated fuel assembly (2,100 lbs design weight) plus the Maine Yankee fuel can weight (130 lbs), amplified by a dynamic load factor of 10%. Although the four slots are used to lift the can, the analysis assumes that the entire design load is shared by only two lift slots.



The stress in the side plate above the slot is determined by analyzing the section above the slot as a 0.15-inch-wide × 1.875-inch-long × 1.125-inch-deep beam that is fixed at both ends. The

lifting tool lug is 1-inch wide and engages the last 1 inch of the slot. The following figure represents the configuration to be evaluated:



where:

a = 0.875 in.
L = 1.875 in.
$w_a = w_L$ = (2,230 lbs/2)(1.10)/1.0 in. = 613.3 lbs/in, use 620 lbs/in.

Reactions and moments at the fixed ends of the beam are calculated per Roark's Formula, Table 3, Case 2d.

The reaction at the left end of the beam (R_A) is:

$$R_A = \frac{w_a}{2L^3}(L-a)^3(L+a)$$

$$= \frac{620}{2(1.875)^3}(1.875-0.875)^3(1.875+0.875) = 129.3 \text{ lbs}$$

The moment at the left end of the beam (M_A) is:

$$M_A = \frac{-w_a}{12L^2}(L-a)^3(L+3a)$$

$$= \frac{-620}{12(1.875)^2}(1.875-0.875)^3(1.875+3(0.875)) = -66.1 \text{ lbs}\cdot\text{in.}$$

The reaction at the right end of the beam (R_B) is:

$$R_B = w_a(L-a) - R_A = 620(1.875-0.875) - 164.2 = 490.7 \text{ lbs}$$

The moment at the right end of the beam (M_B) is:

$$M_B = R_A L + M_A - \frac{w_a}{2}(L-a)^2$$
$$= 129.3(1.875) + (-66.1) - \frac{620}{2}(1.875-0.875)^2 = -133.7 \text{ lbs}\cdot\text{in.}$$

The maximum bending stress (σ_b) in the side plate is:

$$\sigma_b = \frac{Mc}{I} = \frac{133.7(0.50)}{0.017} = 4{,}224 \text{ psi}$$

The maximum shear stress (τ) occurs at the right end of the slot:

$$\tau = \frac{R_B}{A} = \frac{490.7}{1.125(0.15)} = 2{,}908 \text{ psi}$$

The Von Mises stress (σ_{max}) is:

$$\sigma_{max} = \sqrt{\sigma_b^2 + 3\tau^2} = \sqrt{4{,}224^2 + 3(2{,}908)^2} = 6{,}573 \text{ psi}$$

The yield strength (S_y) for Type 304 stainless steel is 22,500 psi at 300°F. The factor of safety is calculated as:

$$FS = \frac{22{,}500}{6{,}573} = 3.4 > 3$$

The design condition requiring a safety factor of 3 on material yield strength is satisfied.

Tensile Stress

The tube body will be subjected to tensile loads during lifting operations. The load (P) includes the can contents (2,100 lbs design weight), the tube body weight (78.77 lbs), and the bottom

assembly weight (12.98 lbs) for a total of 2,191.8 pounds. A load of 2,200 lbs with a 10% dynamic load factor is used for the analysis.

The tensile stress (σ_t) is then:

$$\sigma_t = \frac{1.1P}{A} = \frac{1.1(2{,}200\ \text{lb})}{1.714\ \text{in}^2} = 1{,}412\ \text{psi}$$

where:

A = tube cross-section area = $8.62^2 - 8.52^2 = 1.714\ \text{in}^2$

The factor of safety (FS) based on the yield strength at 600°F (18,000 psi) is:

$$FS = \frac{18{,}600\ \text{psi}}{1{,}412} = 13.2 > 3$$

Weld Evaluation

The welds joining the tube body to the bottom weldment and to the side plates are full penetration welds (Type III, paragraph NG-3352.3). In accordance with NG-3352-1, the weld quality factor (n) for a Type III weld with visual surface inspection is 0.5.

The weld stress (σ_w) is:

$$\sigma_w = \frac{1.1(P)}{A} = \frac{1.1(2{,}200)}{1.714} = 1{,}412\ \text{psi}$$

where:

P = the combined weight of the tube body, bottom weldment, and can contents
A = cross sectional area of thinner member joined

The factor of safety (FS) is:

$$FS = \frac{n \cdot S_y}{\sigma_w} = \frac{0.5(18{,}600\ \text{psi})}{1{,}412\ \text{psi}} = +6.6 > 3$$

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4.5 Thermal Evaluation for Site Specific Spent Fuel

This section presents the thermal evaluation of fuel assemblies or configurations, which are unique to specific reactor sites or which differ from the UMS® Storage System design basis fuel. These site specific configurations result from conditions that occurred during reactor operations, participation in research and development programs, and from testing programs intended to improve reactor operations. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies, and fuel that is classified as damaged. Damaged fuel includes fuel rods with cladding that exhibit defects greater than pinhole leaks or hairline cracks.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable contents by specific evaluation.

4.5.1 Maine Yankee Site Specific Spent Fuel

The standard spent fuel assembly for the Maine Yankee site is the Combustion Engineering (CE) 14 x 14 fuel assembly. Fuel of the same design has also been supplied by Westinghouse and by Exxon. The standard 14 x 14 fuel assembly is included in the population of the design basis PWR fuel assemblies for the UMS® Storage System (See Table 2.1.1-1). The maximum decay heat for the standard Maine Yankee fuel is the design basis heat load for the PWR fuels (23 kW total, or 0.958 kW per assembly). This heat load is bounded by the thermal evaluations in Section 4.4 for the normal conditions of storage, Section 4.4.3.1 for less than design basis heat loads and Chapter 11 for off-normal and accident conditions.

Some Maine Yankee site specific fuel has a burnup greater than 45,000 MWD/MTU, but less than 50,000 MWD/MTU. This fuel is evaluated in Section 4.5.1.2. As shown in that section, loading of fuel assemblies in this burnup range is subject to preferential loading in designated basket positions in the Transportable Storage Canister and certain fuel assemblies in this burnup range must be loaded in a Maine Yankee fuel can.

The site specific fuels included in this evaluation are:

1. Consolidated fuel rod lattices consisting of a 17 x 17 lattice fabricated with 17 x 17 grids, 4 stainless steel support rods and stainless steel end fittings. One of these

lattices contains 283 fuel rods and 2 rod position vacancies. The other contains 172 fuel rods, with the remaining rod position locations either empty or containing stainless steel dummy rods.

2. Standard fuel assemblies with a Control Element Assembly (CEA) inserted in each one.
3. Standard fuel assemblies that have been modified by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods, or 1.95 wt % enriched fuel rods.
4. Standard fuel assemblies that have had the burnable poison rods removed and replaced with hollow Zircaloy tubes.
5. Standard fuel assemblies with in-core instrument thimbles stored in the center guide tube.
6. Standard fuel assemblies that are designed with variable enrichment (radial) and axial blankets.
7. Standard fuel assemblies that have some fuel rods removed.
8. Standard fuel assemblies that have damaged fuel rods.
9. Standard fuel assemblies that have some type of damage or physical alteration to the cage (fuel rods are not damaged).
10. Two (2) rod holders, designated CF1 and CA3. CF1 is a lattice having approximately the same dimensions as a standard fuel assembly. It is a 9 x 9 array of tubes, some of which contain damaged fuel rods. CA3 is a previously used fuel assembly lattice that has had all of the rods removed, and in which damaged fuel rods have been inserted.
11. Standard fuel assemblies that have damaged fuel rods stored in their guide tubes.
12. Fuel debris from 24 damaged (failed) fuel rods not in a Maine Yankee fuel can.
13. Standard fuel assemblies with inserted startup sources and other non-fuel items.

The Maine Yankee site specific fuels are also described in Section 1.3.2.1.

The thermal evaluations of these site specific fuels are provided in Section 4.5.1.1. Section 4.5.1.2 presents the evaluation of Maine Yankee fuel inventory that is not bounded by the evaluation performed in Section 4.4.7. This fuel may have higher burnup than the design basis fuel, have a higher decay heat on a per assembly basis, have a burnup/cool time condition that is outside of the cladding temperature evaluation presented in Section 4.4.7, or be subject to all of these differences.

4.5.1.1 Thermal Evaluation for Maine Yankee Site Specific Spent Fuel

The maximum heat load per assembly for site specific fuel considered in this section is limited to the design basis heat load (0.958 kW). The evaluation of fuel configurations having a greater heat load is presented in Section 4.5.1.2.

4.5.1.1.1 Consolidated Fuel

There are two (2) consolidated fuel lattices. One lattice contains 283 fuel rods and the other contains 172 fuel rods. Conservatively, only one consolidated fuel lattice is loaded in any Transportable Storage Canister.

The maximum decay heat of the consolidated fuel lattice having 283 fuel rods is 0.279 kW. This heat load is bounded by the design basis PWR fuel assembly, since it is less than one-third of the design basis heat load.

The second consolidated fuel lattice has 172 fuel rods with 76 stainless steel dummy rods at the outer periphery of the lattice. Due to the existence of the stainless steel rods, the effective thermal conductivities of this assembly may be slightly lower than those of the standard CE 14 x 14 fuel assembly. While the stainless steel rods provide better conductance in the axial direction, the radiation heat transfer is less effective at the surface of stainless steel rods, as compared to the standard fuel rods. The radiation is a function of surface emissivity and the emissivity for stainless steel (0.36) is less than one-half of that for Zircaloy (0.75). A parametric study is performed to demonstrate that the thermal performance of the UMS PWR basket loading configuration consisting of 23 standard CE 14 x 14 fuel assemblies and the consolidated fuel lattice with stainless rods is bounded by that of the configuration consisting of 24 standard CE 14 x 14 fuel assemblies. Two finite element models are used in the study: a two-dimensional fuel assembly model and a three-dimensional periodic canister internal model.

The two-dimensional model is used to determine the effective thermal conductivities of the consolidated fuel lattice with stainless steel rods. Considering the symmetry of the consolidated fuel, the finite element model represents a one-quarter section as shown in Figure 4.5.1.1-1. The methodology used in Section 4.4.1.5 for the two-dimensional fuel model for PWR fuel is employed in this model. The model includes the fuel pellets, cladding, helium between the fuel rods, and helium occupying the gap between the fuel pellets and cladding. In addition, the rods

at the two outer layers are modeled as solid stainless steel rods to represent the configuration of this consolidated fuel lattice. Modes of heat transfer modeled include conduction and radiation between individual rods for steady-state condition. ANSYS PLANE55 conduction elements and LINK31 radiation elements are used in the model. Radiation elements are defined between rods and from rods to the boundary of the model. The effective conductivity for the fuel is determined using the procedure described in Section 4.4.1.5.

The three-dimensional periodic canister internal model consists of a periodic section of the canister internals. The model contains one support disk with two heat transfer disks (half thickness) on its top and bottom, the fuel assemblies, the fuel tubes and the helium in the canister, as shown in Figure 4.5.1.1-2. The purpose of this model is to compare the maximum fuel cladding temperatures of the following cases:

1) Base Case: All 24 positions loaded with standard CE 14 x 14 fuel assemblies.
2) Case 2: 23 positions with standard fuel, with one consolidated fuel lattice in position 2.
3) Case 3: 23 positions with standard fuel, with one consolidated fuel lattice in position 3.
4) Case 4: 23 positions with standard fuel, with one consolidated fuel lattice in position 4.
5) Case 5: 23 positions with standard fuel, with one consolidated fuel lattice in position 5.

Positions 2, 3, 4, and 5 are shown in Figure 4.5.1.1-3. Based on symmetry, these locations represent all of the possible locations for consolidated fuel in the basket.

The fuel assemblies and fuel tubes are represented by homogeneous regions with effective thermal conductivities. The effective conductivities for the consolidated fuel are determined by the two-dimensional fuel assembly model discussed above. The effective conductivities for the CE 14 x 14 fuel assemblies are established based on the model described in Section 4.4.1.5. Effective properties for the fuel tubes are determined by the two-dimensional fuel tube model in Section 4.4.1.6. Volumetric heat generation corresponding to the design basis heat load of 0.958 kW per assembly is applied to the CE 14 x 14 fuel regions in the model. Similarly, a heat generation rate corresponding to 0.279 kW is applied to the consolidated fuel assembly region. The heat conduction in the axial direction is conservatively ignored by assuming that the top and

bottom surfaces of the model are adiabatic. A constant temperature of 400°F is applied to the outer surface of the model as boundary conditions. Note that the maximum canister temperature is 351°F for PWR configurations for the normal condition of storage (Table 4.1-4). Steady state thermal analysis is performed for all five cases and the calculated maximum fuel cladding temperatures in the model are:

	Base Case	Case 2	Case 3	Case 4	Case 5
Maximum Fuel Cladding Temperature (°F)	755	733	738	740	740

As shown, the maximum temperatures for Cases 2 through 5 are less than those of the Base Case. It is concluded that the thermal performance of the configuration consisting of 23 standard CE 14 x 14 fuel assemblies and one consolidated fuel lattice is bounded by that of the configuration consisting of 24 standard CE 14 x 14 fuel assemblies. This study shows that a consolidated fuel lattice can be located in any basket position. However, as shown in Table 12B2-6 in Chapter 12, the consolidated fuel assembly must be loaded in a corner position of the fuel basket (e.g., Position 5 shown in Figure 4.5.1.1-3).

4.5.1.1.2 Standard CE 14 x 14 Fuel Assemblies with Control Element Assemblies

A Control Element Assembly (CEA) consists of five solid B_4C rods encapsulated in stainless steel tubes. The B_4C material has a conductivity of 1.375 BTU/hr-in-°F. With the CEA inserted into the guide tubes of the CE 14 x 14 fuel assembly, the effective conductivity in the axial direction of the fuel assembly is increased because solid material replaces helium in the guide tubes. The change in the effective conductivity in the transverse direction of the fuel assembly is negligible since the CEA is inside of the guide tubes. Note that the total heat load, including the small amount of extra heat generated by the CEA, remains below the design basis heat load. Therefore, the thermal performance of the fuel assemblies with CEAs inserted is bounded by that of the standard fuel assemblies.

4.5.1.1.3 Modified Standard Fuel Assemblies

These assemblies include those standard fuel assemblies that have been modified by removing damaged fuel rods and replacing them with stainless steel dummy rods, solid zirconium rods or 1.95 wt % enriched fuel rods.

The maximum number of fuel rods replaced by stainless steel rods is six (6) per assembly, which is about 3% of the total number of fuel rods in each assembly (176). The conductivity of the stainless steel is similar to that of Zircaloy and better than that of the UO_2. The resultant increase in effective conductivity of the modified fuel assembly in the axial direction offsets the decrease in the effective conductivity in the transverse direction (due to slight reduction of radiation heat transfer at the surface of the stainless steel rods). The maximum number of fuel rods replaced by solid Zirconium rods is five (5) per assembly. Since the solid Zirconium rod has a higher conductivity than the fuel rod (UO_2 with Zircaloy clad), the effective conductivity of the repaired fuel assembly is increased. The thermal properties for the enriched fuel rod remain the same as for standard fuel rods, so there is no change in effective conductivity of the fuel assembly results from the use of fuel rods enriched to 1.95 wt % ^{235}U. These rods replace other fuel rods in the assembly after the first or second burnup cycles were completed. Therefore, these replacement fuel rods have been burned a minimum of one cycle less than the remainder of the assembly, producing a proportionally lower per rod heat load. The heat load (on a per rod basis) of the fuel rods in a standard assembly, bounds the heat load of the 1.95 wt % ^{235}U enriched fuel rods. Consequently, the loading of modified fuel assemblies is bounded by the thermal evaluation of the standard fuel assembly.

4.5.1.1.4 Use of Hollow Zircaloy Tubes

Certain standard fuel assemblies have had the burnable poison rods removed. These rods were replaced with hollow Zircaloy tubes.

There are 16 locations where burnable poison rods were removed and hollow Zircaloy tubes were installed in their place. Since the maximum heat load for these assemblies is 0.552 kW per assembly (less than two-thirds of the design basis heat load) and the number of hollow Zircaloy rods is only about one-tenth (16/176) of the total number of the fuel rods, the thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

4.5.1.1.5 Standard Fuel with In-core Instrument Thimbles

Certain fuel assemblies have in-core instrument thimbles stored within the center guide tube of each fuel assembly. Storing an in-core instrument thimble assembly in the center guide tube of a fuel assembly will slightly increase the axial conductance of the fuel assembly (helium replaced by solid material). Therefore, there is no negative impact on the thermal performance of the fuel

assembly with this configuration. The thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

4.5.1.1.6 Standard Fuel Assemblies with Variable Enrichment and Axial Blankets

The Maine Yankee variably enriched fuel assemblies are limited to two batches of fuel, which have a maximum burnup less than 30,000 MWD/MTU. The variably enriched rods in the fuel assemblies have enrichments greater than 3.4 wt % ^{235}U, except that the axial blankets on one batch are enriched to 2.6 wt % ^{235}U. As shown in Table 12B2-8, fuel at burnups less than or equal to 30,000 MWD/MTU with any enrichment greater than, or equal to, 1.9 wt % ^{235}U may be loaded with 5 years cool time.

The thermal conductivities of the fuel assemblies with variable enrichment (radial) and axial blankets are considered to be essentially the same as those of the standard fuel assemblies. Since the heat load per assembly is limited to the design basis heat load, there is no effect on the thermal performance of the system due to this loading configuration.

4.5.1.1.7 Standard Fuel Assemblies with Removed Fuel Rods

Except for assembly number EF0046, the maximum number of missing fuel rods from a standard fuel assembly is 14, or 8% (14/176) of the total number of rods in one fuel assembly. The maximum heat load for any one of these fuel assemblies is conservatively determined to be 0.63 kW. This heat load is 34% less than the design basis heat load of 0.958 kW. Fuel assembly EF0046 was used in the consolidated fuel demonstration program and has only 69 rods remaining in its lattice. This fuel assembly has a heat load of 70 watts, or 7% of the design basis heat load of 0.958 kW. Therefore, the thermal performance of fuel assemblies with removed fuel rods is bounded by that of the standard fuel assemblies.

4.5.1.1.8 Fuel Assemblies with Damaged Fuel Rods

Damaged fuel assemblies are standard fuel assemblies with fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. Fuel classified as damaged, which exceeds the assembly or canister limits described in Section 4.5.1.1.12, will be placed in a Maine Yankee fuel can. The primary function of the fuel can is to confine fuel material within the can and to facilitate handling and retrievability. The Maine Yankee fuel can is shown in Drawings

412-501 and 412-502. The placement of the loaded fuel cans is restricted by the operating procedures and/or Technical Specifications to loading into the four fuel tube positions at the periphery of the fuel basket as shown in Figure 12B2-1. The heat load for each damaged fuel assembly is limited to the design basis heat load 0.958 kW (23 kW/24).

A steady-state thermal analysis is performed using the three-dimensional canister model described in Section 4.4.1.2 simulating 100% failure of the fuel rods, fuel cladding, and guide tubes of the damaged fuel held in the Maine Yankee fuel can. The canister is assumed to contain 20 design basis PWR fuel assemblies and damaged fuel assemblies in fuel cans in each of the four corner positions.

Two debris compaction levels are considered for the 100% failure condition: (Case 1) 100% compaction of the fuel rod, fuel cladding, and guide tube debris resulting in a 52-inch debris level in the bottom of each fuel can, and (Case 2) 50% compaction of the fuel rod, fuel cladding, and guide tube debris resulting in a 104-inch debris level in the bottom of each fuel can. The entire heat generation rate for a single fuel assembly (i.e., 0.958 kW) is concentrated in the debris region with the remainder of the active fuel region having no heat generation rate applied. To ensure the analysis is bounding, the debris region is located at the lower part of the active fuel region in lieu of the bottom of the fuel can. This location is closer to the center of the basket where the maximum fuel cladding temperature occurs. The effective thermal conductivities for the design basis PWR fuel assembly (Section 4.4.1.5) are used for the debris region. This is conservative since the debris (100% failed rods) is expected to have higher density (better conduction) and more surface area (better radiation) than an intact fuel assembly. In addition, the thermal conductivity of helium is used for the remainder of the active fuel length. Boundary conditions corresponding to the normal condition of storage are used at the outer surface of the canister model (see Section 4.4.1.2). A steady-state thermal analysis is performed The results of the thermal analyses performed for 100% fuel rod, fuel cladding, and guide tube failure are:

	Maximum Temperature (°F)			
Description	**Fuel Cladding**	**Damaged Fuel**	**Support Disk**	**Heat Transfer Disk**
Case 1 (100% Compaction)	654	672	598	594
Case 2 (50% Compaction)	674	594	620	616
Design Basis PWR Fuel	670	N/A	615	612
Allowable	716	N/A	650	650

As demonstrated, the extreme case of 100% fuel rod, fuel cladding, and guide tube failure with 50% compaction of the debris results in temperatures that are less than 1% higher than those calculated for the design basis PWR fuel. The maximum temperatures for the fuel cladding, damaged fuel assembly, support disks, and heat transfer disks remain within the allowable temperature range for both 100% failure cases. Additionally, the temperatures used in the structural analyses of the fuel basket envelope those calculated for both 100% failure cases.

Additionally, the above analysis has been repeated to consider a maximum heat load of 1.05 kW/assembly (maximum heat load for the 50,000 MWD/MTU fuel, see Section 4.5.1.2.1) in the Maine Yankee fuel cans. To maintain the 23 kW total heat load per canister, the model considers a heat load of 1.05 kW/assembly in the four Maine Yankee fuel cans and 0.94 kW/assembly in the rest of the 20 basket locations. The analysis results indicate that the maximum temperatures for the fuel cladding and basket components are slightly lower than those for the case with a heat load of 0.958 kW in the damaged fuel can, as presented above. The maximum fuel cladding temperature is 650°F (< 654°F) and 672°F (< 674°F) for 100% and 50% compaction ratio cases, respectively. Therefore, the case with 1.05 kW/assembly in the Maine Yankee fuel can is bounded by the case with 0.958 kW/assembly in the fuel cans.

4.5.1.1.9 Standard Fuel Assemblies with Damaged Lattice

Certain standard fuel assemblies may have damage or physical alteration to the lattice or cage that holds the fuel rods, but not exhibit damage to the fuel rods. Fuel assemblies with lattice damage are evaluated in Section 11.2.16. The structural analysis demonstrates that these assemblies retain their configuration in the design basis accident events and loading conditions.

The effective thermal conductivity for the fuel assembly used in the thermal analyses in Section 4.4 is determined by the two-dimensional fuel model (Section 4.4.1.5). The model conservatively ignores the conductance of the steel cage of the fuel assembly. Therefore, damage or physical alteration to the cage has no effect on the thermal conductivity of the fuel assembly used in the thermal models. The thermal performance of these fuel assemblies is bounded by that of the standard fuel assemblies.

4.5.1.1.10 Damaged Fuel Rod Holders

The Maine Yankee site specific fuel inventory includes two damaged fuel rod holders designated CF1 and CA3. CF1 is a 9 x 9 array of tubes having roughly the same dimensions as a fuel assembly. Some of the tubes hold damaged fuel rods. CA3 is a previously used fuel assembly cage (i.e., a fuel assembly with all of the fuel rods removed), into which damaged fuel rods have been inserted.

Similar to the fuel assemblies that have damaged fuel rods, the damaged fuel rod holders will be placed in Maine Yankee fuel cans and their location in the basket is restricted to one of the four corner fuel tube positions of the basket. The decay heat generated by the fuel in each of these rod holders is less than one-fourth of the design basis heat load of 0.958 kW. Therefore, the thermal performance of the damaged fuel rod holders is bounded by that of the standard fuel assemblies.

4.5.1.1.11 Assemblies with Damaged Fuel Rods Inserted in Guide Tubes

Similar to fuel assemblies that have damaged fuel rods, fuel assemblies that have up to two damaged fuel rods or poison rods stored in each guide tube are placed in Maine Yankee fuel cans and their loading positions are restricted to the four corner fuel tubes in the basket. The rods inserted in the guide tubes can not be from a different fuel assembly (i.e., any rod in a guide tube originally occupied a rod position in the same fuel assembly). Storing fuel rods in the guide tubes of a fuel assembly slightly increases the axial conductance of the fuel assembly (helium replaced by solid material). The design basis heat load bounds the heat load for these assemblies. Therefore, the thermal performance of fuel assemblies with rods inserted in the guide tubes is bounded by that of the standard fuel assemblies.

4.5.1.1.12 Fuel Debris from the Failure of 24 Fuel Rods not in a Maine Yankee Fuel Can

This evaluation assumes that the fuel material from 24 Maine Yankee fuel rods (from fuel assemblies not in Maine Yankee fuel cans) escapes from the fuel rods and is dispersed in the canister as fuel debris. During normal storage conditions, the debris is considered to be located at the bottom of the canister. The maximum total heat load of 24 fuel rods is 0.131 kW (23 kW/24 assemblies × 24 fuel rods/176 fuel rods per assembly = 0.131 kW). Two steady-state analyses are performed using the three-dimensional canister model (PWR) described in Section 4.4.1.2 (Figure 4.4.1.2-1) to determine the effect of the debris on the fuel cladding (Case 1) and

bottom weldment (Case 2). Both cases consider that the additional heat due to the dispersed fuel material is concentrated at the bottom region of a center-most (hottest region) fuel assembly. A debris compaction length of 22 inches is considered in the analysis based on the volume of 24 fuel rods, an available cross-sectional area of a fuel assembly and a 50% compaction ratio of the debris.

To obtain a bounding maximum fuel cladding temperature, Case 1 considers a combination of the fuel debris at the bottom of the active fuel region of the center-most fuel assembly and the worst case condition described in Section 4.5.1.1.8 (Fuel Assemblies with Damaged Fuel Rods) simulating 100% failure of the fuel rods of the damaged fuel held in all four Maine Yankee fuel cans, as shown in Figure 4.5.1.1-5. Case 2 does not consider the combination of fuel debris and the damaged fuel with 100% failure of the fuel rods, since the main purpose of Case 2 is to obtain the maximum temperature difference (ΔT) between center and edge of the bottom weldment. Furthermore, the fuel debris region (22-inches high) in the thermal model for Case 2 is considered to start from the canister bottom plate as opposed to starting from the bottom of the active fuel region. Note that the model is a half-symmetry model. Therefore, applying the additional heat to one central fuel assembly actually simulates the fuel debris in two central fuel assembly locations (total additional heat load = 0.113 kW × 2 = 0.226 kW), which is conservative. The results of the analyses are:

Configuration	Maximum Temperature (°F)				
	Fuel Cladding	Heat Transfer Disk	Support Disk	Bottom Weldment	
				Max.	ΔT
Case 1: Additional Fuel Debris at the bottom of one of the center fuel assembly locations plus Damaged Fuel at the four basket corner locations	675	617	620	170	58
Case 2: Additional Fuel Debris at the bottom of one of the center fuel assembly locations	672	613	617	188	77
Design Basis PWR Fuel	670	612	615	151	40
Allowable Temperature	716	650	650	800	N/A

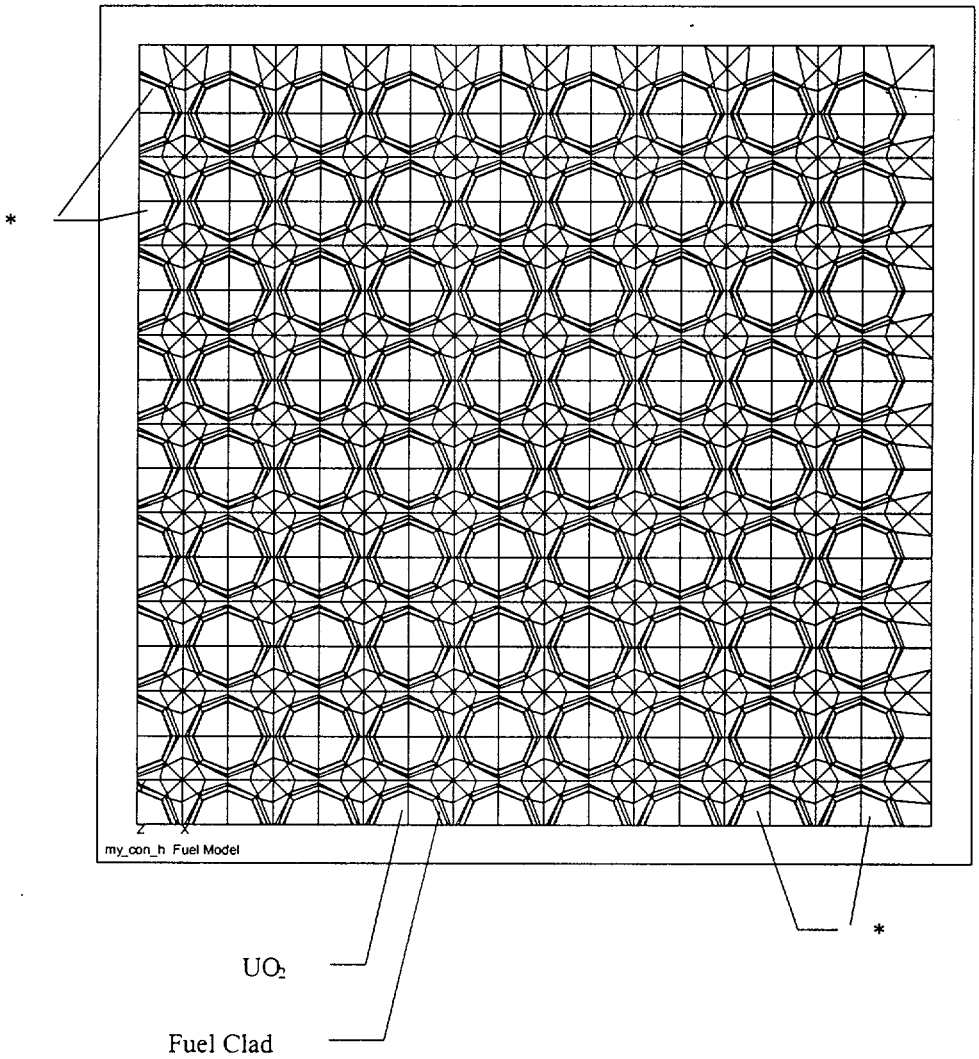
As shown, the maximum temperatures for the fuel cladding, heat transfer disk and support disk for both cases remain well below their allowable temperatures. The maximum temperature and the temperature difference between center and edge (ΔT) for the bottom weldment both increased

by 37°F (Case 2). However, both the maximum temperature (188°F) and the temperature difference (77°F) for the bottom weldment remain bounded by the temperatures used in the structural analysis as shown in Section 3.4.4.1.9 (maximum temperature = 250°F and temperture difference = 80°F).

4.5.1.1.13 Standard Fuel Assemblies with Inserted Startup Sources and Other Non-Fuel Items

Five Control Element Assembly (CEA) fingertips and a neutron source associated with the Boronometer may be placed into the guide tubes of a fuel assembly. In addition, four used startup neutron sources and one unirradiated source will be loaded into separate fuel assemblies. With the CEA fingertips and the neutron sources inserted into the guide tubes of the fuel assemblies, the effective conductivity in the axial direction of the fuel assembly is increased because solid material replaces helium in the guide tubes. The change in the effective conductivity in the transverse direction of the fuel assembly is negligible, since the non-fuel items are inside of the guide tubes. Note that the total heat load of the fuel assembly, including the small amount of extra heat generated by the CEA fingertips and the neutron sources, remains below the design basis heat load. Therefore, the thermal performance of the fuel assemblies with CEA fingertips and neutron sources inserted is bounded by that of the standard fuel assemblies.

Figure 4.5.1.1-1 Quarter Symmetry Model for Maine Yankee Consolidated Fuel



* Two outer layers (rows) of rods are modeled as stainless steel

Figure 4.5.1.1-2 Maine Yankee Three-Dimensional Periodic Canister Internal Model

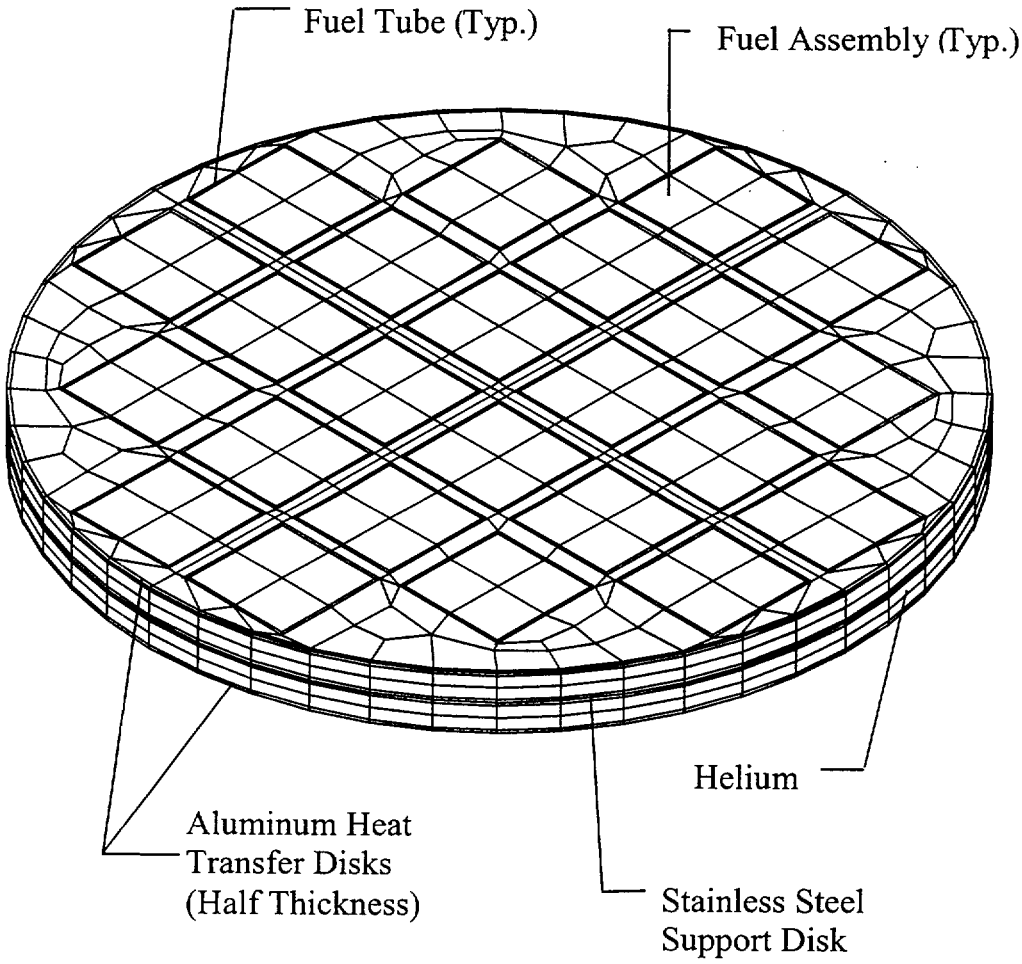


Figure 4.5.1.1-3 Evaluated Locations for the Maine Yankee Consolidated Fuel Lattice in the PWR Fuel Basket

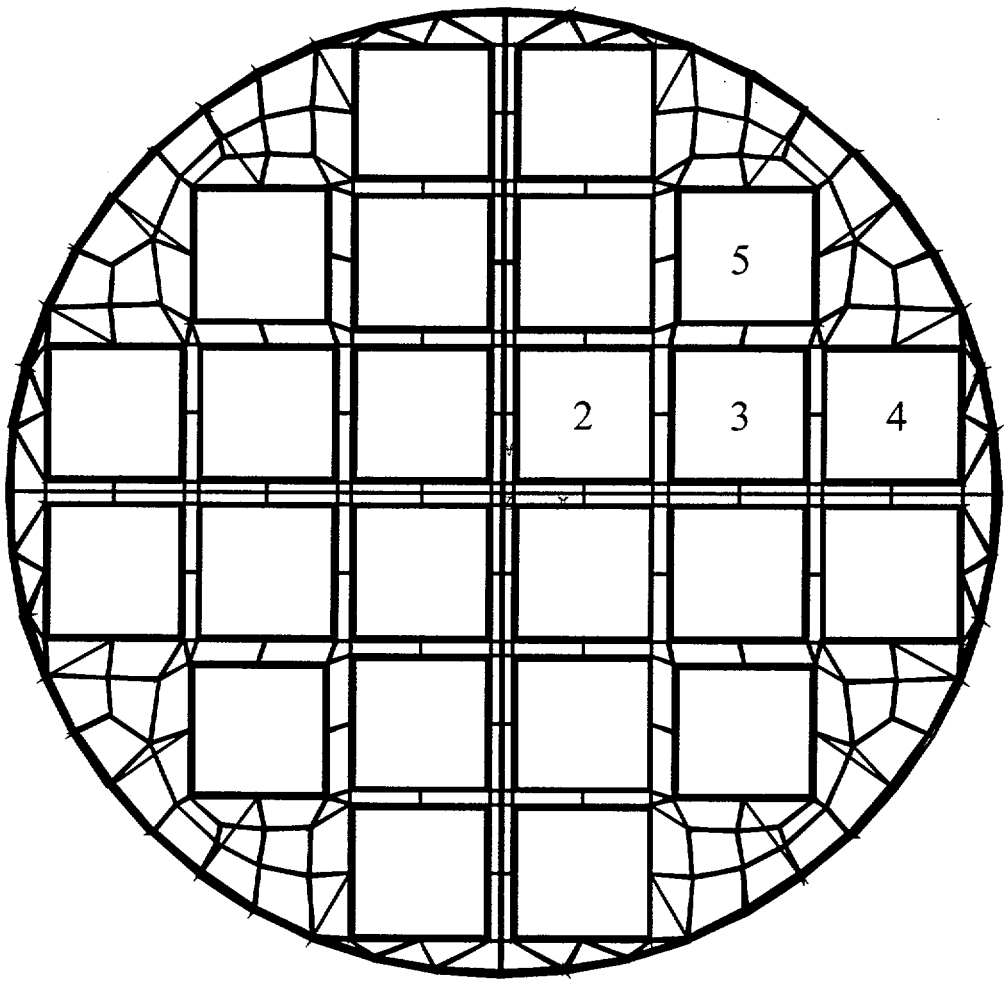
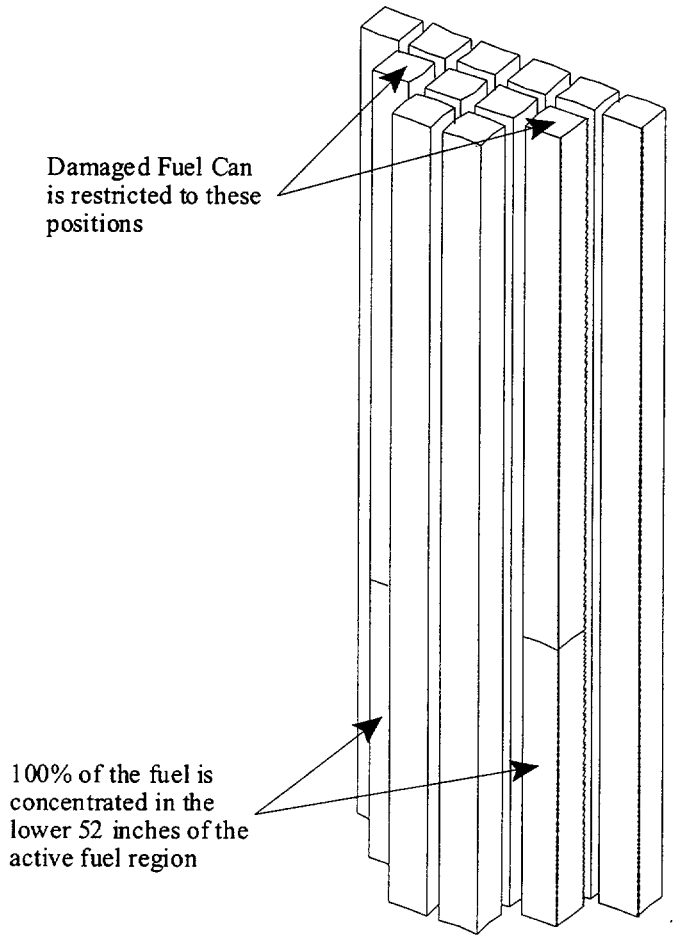
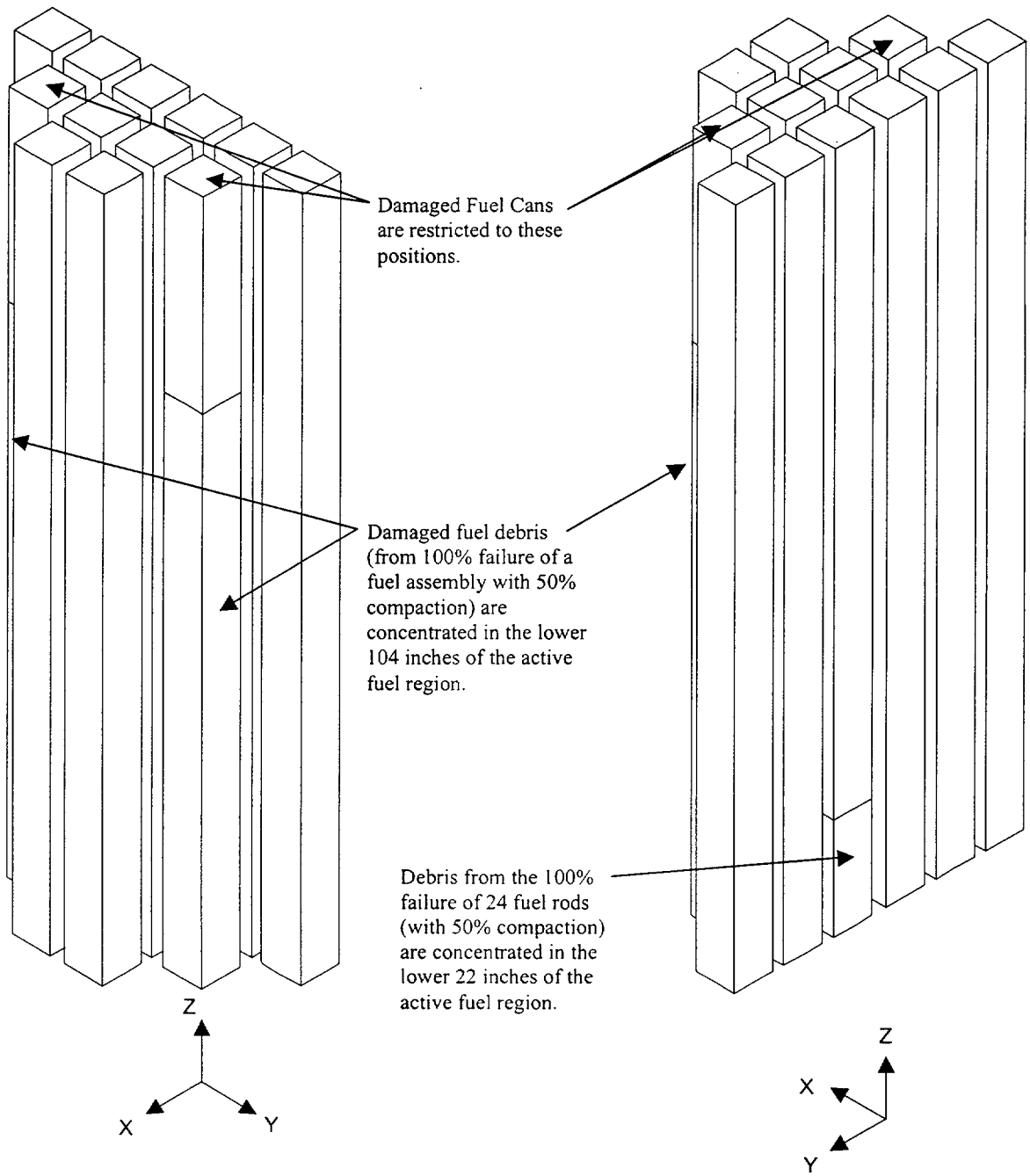


Figure 4.5.1.1-4 Active Fuel Region in the Three-Dimensional Canister Model



Note: Finite element mesh not shown for clarity.

Figure 4.5.1.1-5 Fuel Debris and Damaged Fuel Regions in the Three-Dimensional Canister Model



4.5.1.2 Maximum Allowable Heat Loads for Maine Yankee Site Specific Spent Fuel

This section includes evaluations for the Maine Yankee fuel inventory that is not bounded by the evaluation performed in Section 4.4.7. This fuel may have higher burnup than the design basis fuel, have a higher decay heat on a per assembly basis, have a burnup/cool time condition that is outside of the cladding temperature evaluation presented in Section 4.4.7, or be subject to all of these differences.

Maximum allowable clad temperatures and decay heats are evaluated for:

1. Fuel with burnup in excess of 45,000 MWD/MTU (maximum 50,000 MWD/MTU),
2. Preferential loading patterns with hotter fuel on the periphery of the basket, and
3. Preferential loading with fuel exceeding design basis heat load (0.958 kW) per assembly on the basket periphery.

As shown in Section 4.4.7, the standard CE 14 x 14 fuel assembly has a significantly lower cladding stress level than the equivalent burnup Westinghouse 14 x 14 assembly. It is, therefore, conservative to apply the characteristics of the design basis assembly to the CE 14 x 14 Maine Yankee fuel assemblies. (Note that the Westinghouse 14 x 14 assembly evaluated in Section 4.4.7 is the fuel assembly used in Westinghouse reactors, but it is not the Westinghouse 14 x 14 assembly built for use in the CE reactors, such as the Maine Yankee reactor.)

The maximum allowable decay heat, listed either on a per canister or per assembly basis, is combined with dose rate limits in Chapter 5 to establish cool time limits as a function of burnup and initial enrichment. Cool time limits are shown in Tables 5.6.1-10 for Maine Yankee fuel assemblies without installed control components, and in Table 5.6.1-12 for fuel assemblies with installed control components.

High burnup fuel (45,000 – 50,000 MWD/MTU) may be loaded as intact fuel provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns. The high burnup fuel must be loaded as failed fuel (i.e., in a Maine Yankee fuel can), if these criteria are not met, or if the cladding oxide layer is detached or spalled from the cladding. Since the transportable storage canister is tested to be leak tight, no additional confinement analysis is required for the high burnup fuel.

4.5.1.2.1 Maximum Allowable Temperature and Decay Heat for 50,000 MWD/MTU Fuel

To evaluate higher burnup fuel, cladding oxidation layer thickness and fission gas release fractions are established. Maine Yankee reports that for high burnup fuel rods (i.e., rod peak burnup up to 55,000 MWD/MTU), ABB/Combustion Engineering Incorporated imposes a cladding oxide layer thickness of 120 microns as an operational limit and reports that the maximum gas release fraction (fuel pellet to rod plenum in intact fuel rods) is less than 3% [36]. Therefore, the allowable cladding temperature calculations employ a cladding oxide layer thickness of 0.012 cm (120 microns). This is conservative with respect to the 80 micron cladding oxide layer thickness considered for high burnup fuel that is loaded as intact fuel. A 12% release fraction, established for standard PWR fuel burned up to 45,000 MWD/MTU, is conservatively applied to higher burnup PWR fuel.

Using the evaluation method presented in Section 4.4.7 and a cladding oxidation layer thickness of 0.012 cm, the cladding stress levels for the 50,000 MWD/MTU burnup PWR assembly (maximum stress) are determined and listed in Table 4.5.1.2-1. The data is plotted against the generic allowable temperature curves in Figure 4.5.1.2-2. Included in Figure 4.5.1.2-2 are the 35,000 MWD/MTU to 45,000 MWD/MTU limit lines developed in Section 4.4.7. The intercept of the 50,000 MWD/MTU results in the limiting cladding temperatures shown in Table 4.5.1.2-2, which considers the 1% creep strain limit. The resulting maximum allowable heat load per canister for fuel assemblies with burnup of 50,000 MWD/MTU is listed in Table 4.5.1.2-3.

4.5.1.2.2 Preferential Loading with Hotter Fuel on the Periphery of the Basket

The design basis heat load for the UMS thermal analysis is 23 kW uniformly distributed throughout the basket (0.958 kW per assembly). This heat load applies to the basket structural components at any initial fuel loading time. Further reduction in heat load is required for the Maine Yankee fuel assemblies that fall outside the bounds of the requirement of maximum heat load as shown in Tables 4.4.7-8 and 4.5.1.2-3. These assemblies include:

1. Fuel assemblies (with specific burnup and cool time) that may exceed the maximum allowable decay heat dictated by their cladding temperature allowable (exceeding the limits as shown in Tables 4.4.7-8 and 4.5.1.2-3), if loaded uniformly (all 24 fuel assemblies with the same burnup and cool time, i.e., the same decay heat).

2. Fuel assemblies that are expected to exceed the design basis heat load of 0.958 kW per assembly (maximum heat per assembly less than 1.05 kW).

To ensure that these fuel assemblies do not exceed their allowable cladding temperatures, a loading pattern is considered that places higher heat load assemblies at the periphery of the basket (Positions "A" in Figure 4.5.1.2-1) and compensates by placing lower heat load assemblies in the basket interior positions (Positions "B" in Figure 4.5.1.2-1). There are 12 interior basket locations and 12 peripheral basket locations in the UMS PWR basket design. The maximum total basket heat loads indicated in Tables 4.4.7-8 and 4.5.1.2-3 are maintained for these peripheral loading scenarios.

Two preferential loading scenarios are evaluated. The first approach limits any assembly to the 0.958 kW design basis heat load limit (23 kW divided by 24 assemblies), while the second approach increases the per assembly heat load limit to 1.05 kW for assemblies in the basket peripheral locations. The split approach allows maximum flexibility at fuel loading.

In order to load the preferential pattern, the fuel cladding maximum temperature must be maintained below the allowable temperatures for peripheral and interior assemblies. The requirement of maximum total heat load per basket, as shown in Tables 4.4.7-8 and 4.5.1.2-3, must also be met.

4.5.1.2.2.1 Peripheral Assemblies Limited to a Decay Heat Load of 0.958 kW per Assembly

With a basket heat load of 23 kW, uniformly loaded, the maximum cladding temperature of a peripheral assembly location was determined to be 566°F (297°C) based on the thermal analysis using the three-dimensional canister model as presented in Section 4.4.1.2. While any basket location is restricted to a heat load of 0.958 kW, any non-uniform loading with a total basket heat load less than 23 kW will result in a peripheral assembly cladding temperature less than 297°C. This temperature is well below the lowest maximum allowable clad temperature of 313°C indicated in Table 4.5.1.2-2 (which was already reduced to 95% of the actual allowable of 329°C). Fuel assemblies at a maximum heat load of 0.958 kW may, therefore, be loaded into the peripheral basket location at any cool time, provided interior assemblies meet the restrictions outlined below.

Decay Heat Limit on Fuel Assemblies Loaded into Basket Interior Positions

Interior fuel assembly decay heat loads must be reduced from those in a uniform loading configuration, see Table 4.4.7-8 and Table 4.5.1.2-3, to allow loading of the higher heat load assemblies in the peripheral locations. A parametric study is performed using the

three-dimensional periodic model as described in Section 4.5.1.1 (Figure 4.5.1.1-2) to demonstrate that placing a higher heat load in the peripheral locations does not result in heating of the fuel assemblies in the interior locations beyond that found in the uniform heat loading case. The side surface of the model is assumed to have a uniform temperature of 350°F.

Two cases are considered (total heat load per cask = 20 kW for both cases):

1. Uniform loading: Heat load = 0.833 (20/24) kW per assembly for all 24 assemblies
2. Non-uniform loading:
   Heat load = 0.958 (23/24) kW per assembly for 12 Peripheral assemblies
   Heat load = 0.708 (17/24) kW per assembly for 12 Interior assemblies

The analysis results (maximum temperatures) are:

	Case 1 Uniform Loading (°F)	Case 2 Non-Uniform Loading (°F)
Fuel (Location 1)	675	648
Fuel (Locations 2 & 4)	632	611
Fuel (Location 5)	577	588
Fuel (Locations 3 & 6)	563	576
Basket	611	592

Locations are shown in Figure 4.5.1.2-1.

The maximum fuel cladding temperature for Case 2 (non-uniform loading pattern) is well below that for Case 1 (uniform loading pattern). The comparison shows that placing hotter fuel in the peripheral locations of the basket and cooler fuel in the interior locations (while maintaining the same total heat load per basket) reduces the maximum fuel cladding temperature (which occurs in the interior assembly), as well as the maximum basket temperature.

Because the basket interior temperatures decrease for non-uniform loading, it is conservative to determine the maximum allowable heat load for the interior assemblies based on the values (total allowed heat load) shown in Tables 4.4.7-8 and 4.5.1.2-3, and the heat load for the fuel assemblies in 12 peripheral locations (12 x 0.958 kW). For example, the 10-year cooled, 45,000 MWD/MTU fuel in a uniform loading pattern, is restricted to a basket average heat load of 19.5 kW per Table 4.4.7-8. Placing 12 fuel assemblies at 23/24 (0.958) kW into the basket periphery

requires the interior assemblies to be reduced to 0.667 kW per assembly to retain the 19.5 kW basket total heat load. Table 4.5.1.2-4 contains the matrix of maximum allowable heat loads per assembly as a function of burnup and cool time for interior assemblies for the configuration with the peripheral assemblies having a maximum heat load of 0.958 kW per assembly.

4.5.1.2.2.2 Peripheral Assemblies Limited to a Decay Heat Load of 1.05 kW per Assembly

The Maine Yankee fuel inventory includes fuel assemblies that will exceed the initial per assembly heat load of 0.958 kW at a loading prior to August 2002. To enable loading of these assemblies into the storage cask, higher peripheral heat load is evaluated. The maximum heat load for peripheral assemblies is set at 1.05 kW.

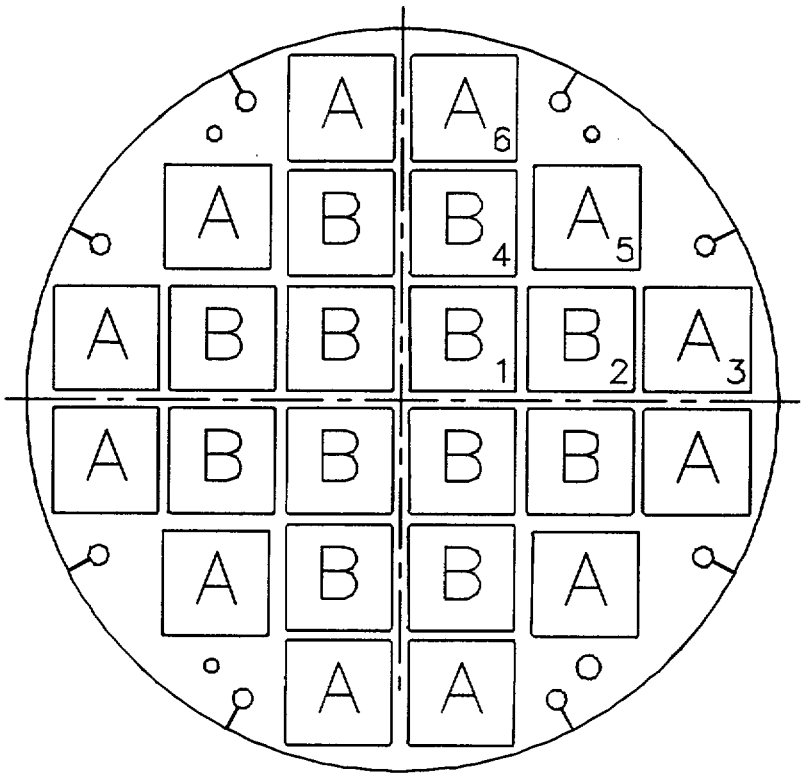
The maximum basket heat load for this configuration is restricted to 23 kW. Given the higher than design basis heat load in peripheral basket locations, an evaluation is performed to assure that maximum cladding allowable temperatures are not exceeded.

Based on the parametric study (uniform versus non-uniform analysis) of the 20 kW basket, a 15% redistribution of heat load resulted in a maximum increase of 13°F (576-563=13) in a peripheral basket location. Changing the basket peripheral location heat load from 0.958 kW maximum to 1.05 kW is a less than 10% redistribution for the 23 kW maximum basket heat load. The highest temperature of a peripheral basket location may, therefore, be estimated by adding 13°F to 566°F (maximum temperature in peripheral assemblies for the 23 kW basket). The 579°F (304°C) is less than the lowest maximum allowable cladding temperature of 313°C indicated in Table 4.5.1.2-2 (which was already reduced to 95% of the actual allowable of 329°C). Fuel assemblies at a maximum heat load of 1.05 kW may, therefore, be loaded into the peripheral basket location at any cool time, provided interior assemblies meet the restrictions outlined below.

Decay Heat Limit on Fuel Assemblies Loaded into Basket Interior Positions

Basket interior assemblies heat load limits are based on the same method used for the configuration with 0.958 kW assemblies in peripheral locations, with the exception that each peripheral fuel assembly is assigned a maximum decay heat of 1.05 kW. The higher peripheral heat load in turn will reduce the allowable heat load in the interior locations. Table 4.5.1.2-5 contains the maximum allowable decay heats for basket interior fuel assemblies with an assembly heat load of 1.05 kW for peripheral locations.

Figure 4.5.1.2-1 Canister Basket Preferential Loading Plan



"A" indicates peripheral locations.

"B" indicates interior locations.

Numbered locations indicate positions where maximum fuel temperatures are presented.

Figure 4.5.1.2-2 Maximum Allowable Cladding Temperature at Initial Storage versus Cladding Stress (50,000 MWD/MTU)

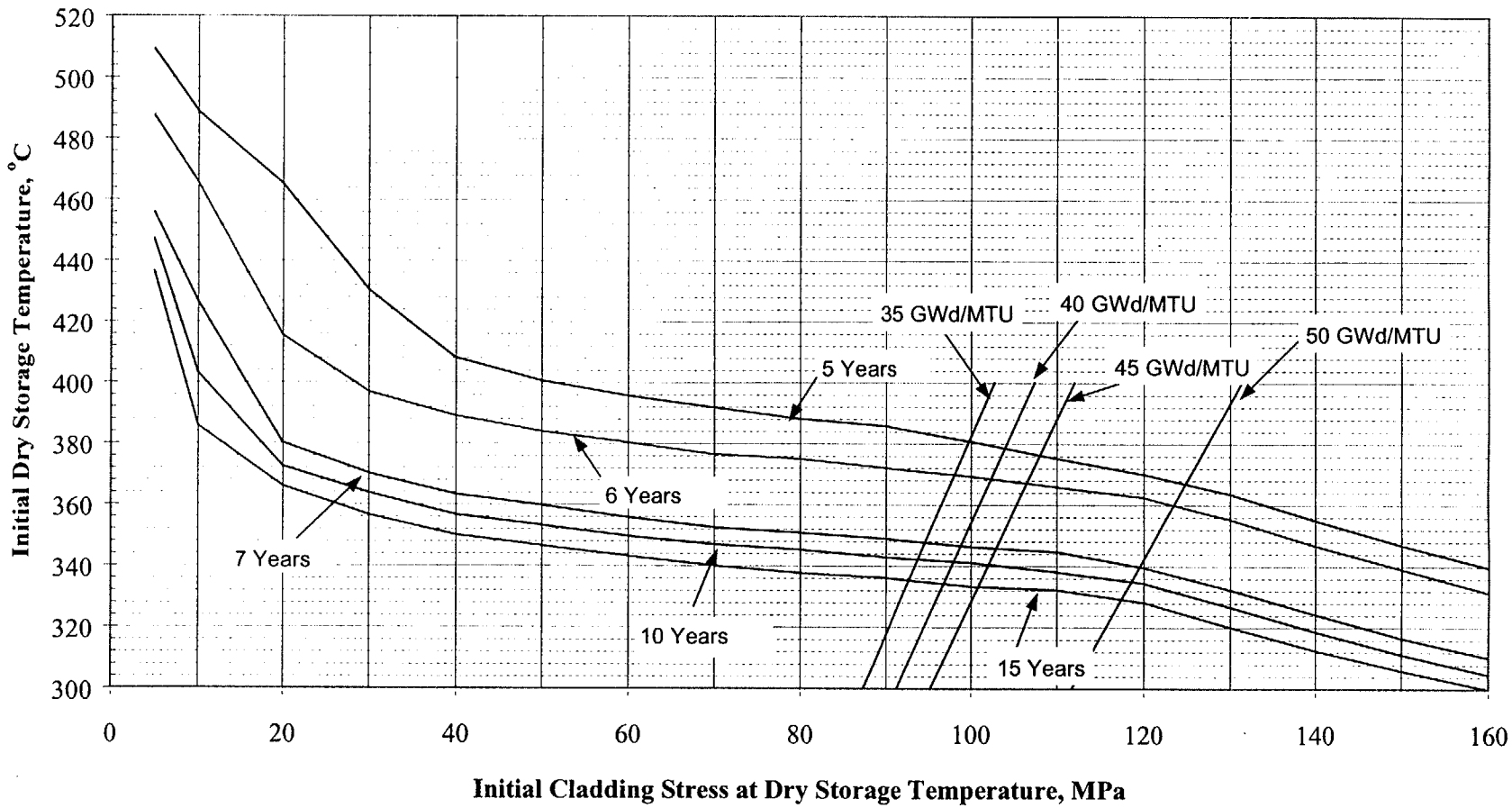


Table 4.5.1.2-1 Cladding Stress for 50,000 MWD/MTU Burnup Fuel

Clad Maximum Temperature	**300°C**	**400°C**
Stress (MPa)	111.7	131.4

Table 4.5.1.2-2 Maximum Allowable Cladding Temperature for 50,000 MWD/MTU Burnup Fuel

Cool Time	**Maximum Allowable Cladding Temperature**	**Cladding Temperature Adjusted to 95% of Maximum**
5 yr	368°C	350°C
6 yr	360°C	342°C
7 yr	340°C	323°C
10 yr	335°C	318°C
15 yr	329°C	313°C

Table 4.5.1.2-3 Maximum Allowable Canister Heat Load for 50,000 MWD/MTU Burnup Fuel

Cool Time	**Maximum Allowable Heat Load**
5 yr	22.1 kW
6 yr	21.2 kW
7 yr	19.5 kW
10 yr	19.1 kW
15 yr	18.7 kW

Table 4.5.1.2-4 Heat Load for Interior Assemblies for the Configuration with 0.958 kW Assemblies in Peripheral Locations

Heat Load Limit (kW)[1]				
Interior Assembly	**Burnup (MWD/MTU)**			
	35,000	**40,000**	**45,000**	**50,000**
Cool Time (years)	---	---	---	---
5	0.958	0.958	0.958	0.883
6	0.908	0.883	0.867	0.808
7	0.725	0.717	0.708	0.667
10	0.683	0.675	0.667	0.633
15	0.633	0.625	0.617	0.600

1. Decay heat per assembly, based on twelve (12) 0.958 kW assemblies in peripheral locations.

Table 4.5.1.2-5 Heat Load Limit for Interior Assemblies for the Configuration with 1.05 kW Assemblies in Peripheral Locations

Heat Load Limit (kW)[1]				
Interior Assembly	**Burnup (MWD/MTU)**			
	35,000	**40,000**	**45,000**	**50,000**
Cool Time (years)	---	---	---	---
5	0.867	0.867	0.867	0.792
6	0.817	0.792	0.775	0.717
7	0.633	0.625	0.617	0.575
10	0.592	0.583	0.575	0.542
15	0.542	0.533	0.525	0.508

1. Decay heat per assembly, based on twelve (12) 1.05 kW assemblies in peripheral locations.

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Table 5.3-5 Transfer Cask Material Densities

Material	Mixture ID	SCL Name	Density [g/cm³]	27N-18G Library Nuclide	Density [a/barn-cm]
Carbon and Low-Alloy Steel	11	CARBONSTEEL	7.8212	CARBON-12 IRON	3.9250E-03 8.3498E-02
Stainless Steel	12	SS304	7.9200	CHROMIUM(SS304) MANGANESE IRON(SS304) NICKEL(SS304)	1.7429E-02 1.7363E-03 5.9358E-02 7.7207E-03
Lead	13	PB	11.3440	LEAD	3.2969E-02
NS-4-FR	14	H B-10 B-11 C N O AL	1.63	HYDROGEN BORON-10 BORON-11 CARBON-12 NITROGEN-14 OXYGEN-16 ALUMINUM	5.8540E-02 8.5530E-05 3.4220E-04 2.2640E-02 1.3940E-03 2.6090E-02 7.7630E-03
Aluminum	17	AL	2.7020	ALUMINUM	6.0307E-02
Concrete	18	REG-CONCRETE	2.2426	HYDROGEN OXYGEN-16 SODIUM-23 ALUMINUM SILICON CALCIUM IRON	1.3401E-02 4.4931E-02 1.7036E-03 1.7018E-03 1.6205E-02 1.4826E-03 3.3857E-04
Canister Void (Dry Conditions)	19	N	VF=1.0E-6	NITROGEN-14	4.3006E-08
Canister Water (Wet Conditions)	19	H2O	0.9982	HYDROGEN OXYGEN-16	6.6769E-02 3.3385E-02

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5.6.1 Shielding Evaluation for Maine Yankee Site Specific Spent Fuel

This analysis considers both fuel assembly sources and sources from activated non-fuel material such as control element assemblies (CEA), in-core instrument (ICI) thimbles, and fuel assemblies containing activated stainless steel replacement (SSR) rods and other non-fuel material, including neutron sources. It considers the consolidated fuel, damaged fuel, and fuel debris present in the Maine Yankee spent fuel inventory, in addition to those fuel assemblies having a burnup between 45,000 and 50,000 MWD/MTU.

The Maine Yankee spent fuel inventory also contains fuel assemblies with hollow zirconium rods, removed fuel rods, axial blankets, poison rods, variable radial enrichment, and low enriched substitute rods. These components do not result in additional sources to be considered in shielding evaluations and are, therefore, enveloped by the standard fuel assembly evaluation. For shielding considerations of the variable radial enrichment assemblies, the planar-average enrichment is employed in determining minimum cool times. As described in Section 6.6.1.2.2, fuel assemblies with variable radial enrichment incorporate fuel rods that are enriched to one of two levels of enrichment. Fuel assemblies that also incorporate axial blankets are described in Section 6.6.1.2.3. Axial blankets consist of annular fuel pellets enriched to 2.6 wt % ^{235}U, used in the top and bottom 5% (≈ 7 inches) of the active fuel length. The remaining active fuel length of the fuel rod is enriched to one of two levels of enrichment incorporated in the fuel design.

5.6.1.1 Fuel Source Term Description

Maine Yankee utilized 14 x 14 array size fuel based on designs provided by Combustion Engineering, Westinghouse, and Exxon Nuclear. The previously analyzed Combustion Engineering CE 14 x 14 standard fuel design is selected as the design basis for this analysis because its uranium loading is the highest of the three vendor fuel types, based on a 0.3765-inch nominal fuel pellet diameter, a 137-inch active fuel length, and a 95% theoretical fuel density. This results in a fuel mass of 0.4037 MTU. This exceeds the maximum reported Maine Yankee fuel mass of 0.397 MTU and, therefore, produces bounding source terms. The SAS2H model of the CE 14 x 14 assembly (shown in Figure 5.6.1-1) at a nominal burnup of 40,000 MWD/MTU and initial enrichment of 3.7 wt %, is based on data provided in Table 2.1.1-1.

Source terms for various combinations of burnup and initial enrichment are computed by adjusting the SAS2H BURN parameter to model the desired burnup and specifying the initial enrichment in the Material Information Processor input for UO_2.

5.6.1.1.1 Control Element Assemblies (CEA)

For the CEA evaluation, the assumptions are:

1. The irradiated portion of the CEA assembly is limited to the CEA tips since during normal operation the elements are retracted from the core and only the tips are subject to significant neutron flux.
2. The CEA tips are defined as that portion present in the "Gas Plenum" neutron source region in the Characteristics Database (CDB) [10].
3. Material subject to activation in the CEA tips is limited to stainless steel, Inconel and Ag-In-Cd in the tip of the CEA absorber rods. Stainless steel and Inconel is assumed to have a concentration of 1.2 g/kg ^{59}Co. The CDB indicates that a total of 2.495 kg/CEA of this material is present in the Gas Plenum region of the core during operation. The Ag-In-Cd alloy present in the gas plenum region during core operation is approximately 80% silver and weighs 2.767 kg/CEA.
4. The irradiated CEA material is assumed to be present in the lower 8 inches of the active fuel region when inserted in the assembly. The location of the CEA source is based on the relative length of the fuel assembly and CEA rods and the insertion depth of the CEA spider into the top end-fitting.
5. The decay heat generated in the most limiting CEA at 5 years cool time is 2.16 W/kg of activated steel and inconel, and 3.11 W/kg of activated Ag-In-Cd. Although longer cool times are considered in this analysis for the fuel source term, this decay heat generation rate is conservatively used for all longer CEA cool times. For a cask fully loaded with fuel assemblies containing design basis CEAs, the additional heat generation due to the CEAs amounts to (2.16 W/kg x 2.495 kg/CEA + 3.11 x 2.767 kg/CEA)(24 CEA/cask) = 336 W/cask, which is conservatively rounded to 350 W/cask.

Since the activated portion of the CEA is present only in the lower 8 inches of the active fuel, an adjustment to the one-dimensional dose rate limit is derived based on detailed three-dimensional results obtained for the CE 14 x 14 fuel with and without a CEA present.

Table 5.6.1-1 shows the activation history for CEAs employed at Maine Yankee. Based on this data, individual source term calculations are performed for each CEA group, and a single

5.6.1.4.1.1 Establishment of Limiting Values

Since the additional activated material in the CEA analysis is assumed present in the lower 8 inches of the active fuel source region, the one-dimensional dose methodology is not appropriate to address the additional source term due to its small axial extent. The one-dimensional analysis is based on the response from the full-length fuel region source. To account for the additional source, the one-dimensional normal conditions dose rate limit is adjusted by an amount that ensures that the contribution from the additional activated material is bounded.

By adjusting the one-dimensional dose rate limit, we require the fuel to cool longer to a point where the decrease in fuel region dose rate matches the increased dose rate due to the additional CEA material. Hence, it is necessary to determine the amount by which the dose rate increases as a result of the added material. A one-dimensional calculation of this additional dose rate is not reasonable due to the small axial extent of the CEA source. One-dimensional buckling corrections are inaccurate for a cylindrical source where the ratio of height to diameter of the source is less than unity, as is the case here.

Instead, the additional contribution to dose rate due to the activated material is computed by a detailed three-dimensional shielding model. The model is based on the three-dimensional models described in Section 5.3. However, the fuel is modeled in a Class 2 canister since that canister will be used to store/transfer CEA-bearing assemblies.

The three-dimensional shielding evaluation is conducted for the CE 14 x 14 fuel at a burnup of 40,000 MWD/MTU and initial enrichment of 3.7 wt %. According to the cool time analysis conducted for PWR fuels in Table 5.6.1-10, this fuel will require 5 years cool time before it is acceptable for transfer or storage in the UMS Vertical Concrete Cask. Hence, the 5-year cooled CE 14 x 14 at 40,000 MWD/MTU and 3.7 wt % initial enrichment provides the base case for the dose rate limit adjustment calculation.

Additional three-dimensional models are defined based on the base case fuel configuration in a Class 2 canister and either containing a design basis CEA assumed to be cooled for 5, 10, 15, or 20 years or containing no CEA at all (No CEA case below).

5.6.1.4.1.2 Three-Dimensional Model Results

Table 5.6.1-11 gives the three-dimensional UMS® Vertical Concrete Cask and transfer cask bottom model results for each case. Only the bottom model is considered because the top model is not sensitive to changes in the CEA description. The "Delta" shown in the table is the difference between the base case maximum (from Table 5.5-3 for the storage cask) dose rate and the value computed for each remaining case. This quantity is directly applied to the one-dimensional design basis normal conditions dose rate limit, as specified in Table 5.6.1-11 for the storage cask to determine a modified limiting value applicable to each CEA decay case. The resulting dose rate limits are shown in the "Limit" column of the table.

Note that direct application of the "Delta" to the one-dimensional dose rate limit is somewhat conservative. The three-dimensional maximum dose rate results are significantly higher than the one-dimensional results, hence a given difference between three-dimensional results represents a larger percentage of the corresponding one-dimensional results.

Also note that the dose rate delta for the "No-CEA" case in Table 5.6.1-11 is zero. Unlike the UMS transport cask, where a spacer positions the canister in the cask, the UMS transfer and storage casks are extended to accommodate the longer Class 2 canister. These cask extensions maintain the spacing of the fuel assembly with respect to the points of minimum shielding in the bottom cask model, and thereby result in identical cask bottom half dose rates for fuel assemblies in Class 1 and Class 2 canisters.

5.6.1.4.1.3 Decay Heat Limits

As discussed in Section 5.6.1.1.1, the additional decay heat associated with a full cask of CEAs is conservatively taken as 0.35 kW/cask. This additional heat load is accounted for by reducing the fuel assembly decay heat limit that is dependent on cool time.

5.6.1.4.1.4 Loading Table Analysis

With the adjusted one-dimensional dose and heat generation rate limits established above, the loading table analysis proceeds following the methodology developed in Section 5.5. Each combination of initial enrichment and burnup is analyzed to determine the minimum required cool time in order for an assembly to either 1) contain a design basis CEA cooled 5, 10, 15, or 20

loading pattern, permitting 1.05 kW per peripheral assembly reduces the minimum cool time based on thermal constraints to 6 years. The storage cask dose rate constraint is satisfied for the preferentially loaded assembles after 5 years cooling. Recognizing that only two of the assemblies in the Maine Yankee spent fuel inventory, R439 and R444, require peripheral loading, the transfer cask dose rate limit is not applied for these two assemblies. Since the dose rate comparisons are made on the basis of an assumed fuel cask of assemblies, the transfer cask dose rate limit is unnecessarily restrictive.

5.6.1.4.4 Consolidated Fuel

There are two consolidated fuel lattices intended for storage (and transfer) in the Universal Storage Cask. The lattices house fuel rods taken from assemblies as shown in Table 5.6.1-6. This fuel has decayed for over twenty years and does not represent a significant shielding issue.

A limiting cool time analysis is conducted by identifying a fuel assembly description analyzed in the loading table analysis that bounds the parameters of the fuel rods in the consolidated fuel lattices. The parameters of those fuel rods are shown in Table 5.6.1-15. The CE 14 x 14 fuel at 30,000 MWD/MTU and 1.9 wt % enrichment represents a bounding assembly type since it has a significantly higher burnup and a lower enrichment than the original assemblies. This fuel requires six years cool time before it can be loaded in the storage or transfer cask as shown in Table 5.6.1-10. The consolidated fuel has been cooled for at least 24 years. For container CN-1 lattice, one can immediately conclude that dose rates are bounded by the limiting fuel.

However, the CN-10 lattice contains significantly more fuel rods than an intact assembly. Neglecting the mitigating effects of additional self-shielding, this situation is addressed by comparing the radiation source strength of the limiting fuel at six and 24 years cool time. Conservatively assuming that all fuel rods present in CN-10 are at the limiting conditions of 30,000 MWD/MTU and 1.9 wt %, the ratio of the source rate in the CN-10 to the source rate in the limiting fuel assembly is shown to be less than one for each source type in Table 5.6.1-16. For each source type, the ratio is computed as:

Ratio = (Num Rods in CN-10)(Source Rate at 24 Yr) / (Num Rods in F/A)(Source Rate at 6 Yr)

Hence, CN-10 is also bounded by the limiting case as of January 1, 2001.

5.6.1.4.5 Damaged Fuel and Fuel Debris

The Maine Yankee spent fuel inventory contains damaged fuel rods and fuel debris. Damaged fuel rods and fuel debris will be placed into a screened Maine Yankee fuel can prior to loading in the UMS basket. Damaged rods and fuel debris in Maine Yankee fuel cans are restricted to loading into one of the four corner basket locations. The damaged fuel mass can not exceed the fuel mass of 100% of an intact fuel assembly. Damaged fuel rods may be loaded in the can with intact rods.

To approximate the effect of collapsed fuel inside the Maine Yankee fuel can, a three-dimensional shielding analysis was performed doubling the source magnitude and material density in the four corner basket locations. Conservatively, the screened can itself is not included in the shielding model. As expected, the increased self-shielding of the collapsed fuel material minimizes the dose rate increase resulting from the source term density doubling. Based on a cask average surface dose rate of less than 40 mrem/hr under normal operating conditions, no significant increases in personnel exposures are expected as a result of the collapsed fuel material.

Where no collapse of the fuel rods occurs, the analysis presented for the intact fuel assemblies bounds that of the damaged fuel rods. Since the additional shielding provided by the screened canister is not being credited by this approach, the actual expected dose rates will be lower for the transportable storage canisters loaded with damaged fuel. For cases in which the Maine Yankee fuel can holds fuel rods from multiple assemblies, the minimum cool time for the rods containing the most restrictive enrichment and burnup combination is applied to the contents of the entire can.

Fuel debris must be placed into a rod structure prior to loading into the screened canister. Once the fuel debris is configured in a rod structure it can be treated from a shielding perspective identical to the damaged fuel rods.

Fuel debris from high burnup rods not loaded into a damaged fuel can may fall to the bottom of the canister. Based on the source doubling and material density doubling analysis described above, no significant increases in dose rates or personnel exposures are expected as a result of uncanned fuel debris.

5.6.1.4.6 Additional Non-fuel and Neutron Source Material

The additional non-fuel material consists of:

1. Three plutonium-beryllium (Pu-Be) neutron sources, two irradiated and one unirradiated.
2. Two antimony-beryllium (Sb-Be) neutron sources, both irradiated.
3. Control element assembly (CEA) fingertips.
4. ICI string segment.
5. Boronometer Pu-Be source.

The five neutron sources will be inserted into the center guide tubes of five different assemblies and loaded into Class 1 canisters. These five assemblies will be loaded in five different canisters. This requirement is conservative since the shielding evaluation shows that only the irradiated Pu-Be sources must be placed in different canisters and that the remaining sources may be loaded in any remaining corner positions of the canister. The CEA fingertips, ICI string segment, and Boronometer source may be inserted into one or more assemblies and loaded into a Class 2 canister to accommodate a CEA flow plug to close the guide tubes with the added hardware. These fuel assemblies must be loaded in corner positions in the fuel basket.

The characterization of the additional non-fuel hardware is provided in Tables 5.6.1-17 and 5.6.1-18. The data is divided into two separate categories:

1. Non-neutron producing radiation sources – this category includes the CEA fingertips, ICI string, and the Sb-Be neutron sources (the neutron production rate of these is negligible).
2. Neutron producing radiation sources – this category includes the two irradiated and one unirradiated Pu-Be neutron sources and the Boronometer Pu-Be neutron source.

The masses of ^{238}Pu and ^{239}Pu given for the unirradiated Pu-Be source are used in conjunction with the delivery date of May 1972 to generate source terms.

The neutron sources have an additional source component due to the irradiation of the stainless steel rod encasing the source. The quantity of irradiated steel is taken as 10 lbs. (4.54 kg) for this evaluation.

From the waste characterization, it is apparent that the Sb-Be sources already include the contribution of irradiated stainless steel. Therefore, only the Pu-Be irradiated stainless steel

requires activation. The hardware source spectra for the irradiated Pu-Be sources are based on the Maine Yankee exposure history shown in Table 5.6.1-4. The combined Pu-Be assembly hardware irradiation for Cycles 1-13 is shown in Table 5.6.1-19 at a cool time of five years from 1/1/1997.

The waste characterization sources given in Tables 5.6.1-17 and 5.6.1-18 are used to generate source terms using ORIGEN-S [9]. For the non-neutron producing sources, the total curie content is assigned to ^{60}Co to provide bounding source terms. Also, only one Sb-Be spectrum is produced, based on the higher curie content source. For the neutron producing sources, the given curie contents are used for the Boronometer source and irradiated sources, whereas the plutonium masses are used for the unirradiated Pu-Be source.

Based on the loading plan, there are two areas of application of both spectra and dose rates. The CEA finger tips, ICI string segment, and the Boronometer source will be loaded into one assembly. Therefore, the gamma spectra of these items are summed and only one gamma and neutron spectrum are used to calculate the dose rates due to this loaded assembly. If these items are loaded into separate fuel assemblies, the source term is lower. Each of the five neutron sources will be loaded into a separate assembly, and the spectra are presented accordingly. The single assembly spectra for the inserted hardware items are presented in Table 5.6.1-20. The startup source spectra is presented in Table 5.6.1-21.

Dose rates are calculated by simply groupwise multiplying the spectra and CE 14 x 14 dose rate response functions and adjusting by a factor of 24/(10E+10 x 5.6193E+06) to remove the volume component and the calculation scaling factor. Dose rates are presented in Tables 5.6.1-22 through 5.6.1-24 and show the minimal dose rate contribution due to the inclusion of the additional non-fuel material.

Figure 5.6.1-1 SAS2H Model Input File – CE 14 x 14

```
=SAS2H       PARM=(HALT03,SKIPSHIPDATA)
CE 14 x 14 3.7 W/O U235, 45000 MWD/MTU 12.0-22.0 YEAR COOLING
27GROUPNDF4 LATTICECELL
UO2         1 0.950 900 92235 3.7 92238 96.3 END
ZIRCALLOY  2 1.0 620 END
H2O         3 DEN=0.725 1.0 580 END
ARBM-BORMOD 0.725 1 1.0 0 5000 100 3 550.0E-6 580 END
END COMP
SQUAREPITCH 1.4732 0.9563 1 3 1.1176 2 0.9754 0 END
NPIN=176 FUEL=347.98 NCYC=3 NLIB=1 PRIN=6 LIGH=5
INPL=1 NUMH=20 NUMI=0 ORTU=0.5588 SRTU=0.49285 END
POWER=13.065 BURN=463.5350 DOWN=60.0 END
POWER=13.065 BURN=463.5350 DOWN=60.0 END
POWER=13.065 BURN=463.5350 DOWN=1461.00 END
```

Table 5.6.1-1 Maine Yankee CEA Exposure History by Group

CEA Group	First Cycle	Last Cycle	Maximum Exposure (MWD/MTU)	Number of Cycles	Exposure Per Cycle (MWD/MTU)	Cool Time as of 1/1/2001 (y)
A1-A8	7	15	60239	9	6693	4
B1-B5	9	15	48909	7	6987	4
C1-C11, C13-C15	10	15	44315	6	7386	4
D1-D15	11	15	35283	5	7057	4
E1-E17, GN, *78, 101, 102, 138-153	12	15	29367	4	7342	4
F1,F2	13	15	18663	3	6221	4
4A	12	12	9786	1	9786	8
C12	10	12	24309	3	8103	8
NA	1	11	75444	11	6859	10
1-69	1	8	53258	8	6657	15

Note: The asterisk is added to CEA 78* to distinguish it from the original CEA 78.

Table 5.6.1-2 Maine Yankee CEA Hardware Spectra - 5, 10, 15 and 20 Years Cool Time

Energy Group	5 yr (γ / sec)	10 yr (γ / sec)	15 yr (γ / sec)	20 yr (γ / sec)
1	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
2	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
3	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
4	0.0000E+00	0.0000E+00	0.0000E+00	0.0000E+00
5	1.3479E-04	4.4697E-06	1.4822E-07	4.9154E-09
6	7.1467E+06	2.6384E+06	1.3598E+06	7.0431E+05
7	4.0337E+09	1.6979E+09	8.7691E+08	4.5422E+08
8	3.7246E+10	2.3434E+08	1.4804E+06	1.5188E+04
9	1.8642E+14	7.1649E+13	3.6955E+13	1.9142E+13
10	4.8840E+14	2.5265E+14	1.3086E+14	6.7790E+13
11	1.3804E+14	9.4554E+11	4.7779E+10	3.7897E+10
12	1.1469E+15	9.3808E+14	9.1172E+14	8.8714E+14
13	4.3885E+14	4.2316E+14	4.1174E+14	4.0065E+14
14	9.1526E+11	5.5505E+11	5.2913E+11	5.0949E+11
15	1.2039E+12	8.4093E+11	8.0140E+11	7.6939E+11
16	3.8479E+12	2.9855E+12	2.7489E+12	2.5803E+12
17	5.1828E+13	4.4134E+13	4.2118E+13	4.0659E+13
18	3.4899E+14	2.7741E+14	2.6393E+14	2.5520E+14
Steel/Inc Source Rate	6.3886E+14	3.2951E+14	1.7066E+14	8.8413E+13
Ag-In-Cd Source Rate	2.1666E+15	1.6829E+15	1.6308E+15	1.5861E+15
Total Source Rate	2.8055E+15	2.0124E+15	1.8014E+15	1.6745E+15
SFA	5.6110E+15	4.0249E+15	3.6029E+15	3.3490E+15

Table 5.6.1-3 Maine Yankee ICI Thimble Exposure History and Source Rate by Group

Group	Quantity	Cycles Exposed	Number of Cycles	Total Source [γ/sec]
A	41	1, 1A, 2	3	9.1881E+11
B	1	1	1	2.3775E+11
C	2	1, 1A	2	3.6244E+11
D	1	1A, 2	2	6.8106E+11
E	3	2	1	5.5637E+11
F	15	3 thru 11, 13	10	1.1695E+13
G	12	3 thru 11, 14	10	1.2126E+13
H	12	3 thru 11, 15	10	1.1454E+13
I	3	3 thru 9,14,15	9	1.1309E+13
J	2	10 thru 15	6	1.4940E+13
K	1	10 thru 12	3	6.1296E+12
L	25	12 thru 15	4	1.1491E+13
M	17	12	1	2.6801E+12
N	3	13 thru 15	3	8.8105E+12

Table 5.6.1-4 Maine Yankee Core Exposure History by Cycle of Operation

Cycle	Discharge Date	Cycle Burnup [MWD/MTU]	Core Average Enrichment [wt %]
1	6/29/74	10367	2.44
1A	5/2/75	4492	2.30
2	4/9/77	17365	2.45
3	7/14/78	11105	2.59
4	1/11/80	10500	2.84
5	5/8/81	10799	2.98
6	9/24/82	11585	3.01
7	3/31/84	12483	3.10
8	8/17/85	12504	3.20
9	3/28/87	14424	3.29
10	10/15/88	12675	3.36
11	4/7/90	13786	3.50
12	2/14/92	15364	3.62
13	7/30/93	13668	3.68
14	1/14/95	13075	3.75
15	12/6/96	7859	3.76

Table 5.6.1-5 Burnup of Maine Yankee Fuel Assemblies with Stainless Steel Replacement Rods

Assembly Number	1st Cycle	2nd Cycle	3rd Cycle	1st Cycle Burnup[1]	2nd Cycle Burnup[1]	3rd Cycle Burnup[1]	Number of SSR Rods
N420	9	10	11	16,428	13,467	11,893	3
N842	9	10	-	18,420	13,885	0	1
N868	9	10	11	18,622	13,386	4,919	1
R032	12	13	14	16,464	15,386	12,168	1
R439	12	13	14	20,371	14,779	11,685	1
R444	12	13	14	20,371	14,779	11,685	4
U01	15	-	-	7,339	0	0	1
U05	15	-	-	7,339	0	0	1
U16	15	-	-	10,598	0	0	1
U37	15	-	-	9,005	0	0	1
U51	15	-	-	8,288	0	0	1
U60	15	-	-	8,288	0	0	6

1. MWD/MTU.

Table 5.6.1-6 Contents of Maine Yankee Consolidated Fuel Lattices CN-1 and CN-10

Consolidated Fuel Lattice	Original Fuel Assembly	Number of Rods	Actual Burnup [MWD/MTU]	Initial Enrichment [wt %]
CN-1	EF0039	172	5150	1.929
CN-10	EF0045	176	17150	1.953
	EF0046	107	17150	1.953

Table 5.6.1-7 Maine Yankee CE 14 x 14 Homogenized Fuel Region Isotopic Composition

Isotope	CE 14 x 14 [atom/b-cm]
ALUMINUM	2.05114E-03
BORON-10	1.90898E-04
BORON-11	7.68387E-04
CARBON-12	2.39821E-04
CHROMIUM(SS304)	7.19369E-04
IRON(SS304)	2.4501E-03
MANGANESE	7.16674E-05
NICKEL(SS304)	3.18674E-04
OXYGEN-16	8.72597E-03
URANIUM-234	2.39964E-07
URANIUM-235	3.14135E-05
URANIUM-238	4.33133E-03
ZIRCALLOY	3.06324E-03

Table 5.6.1-8 Isotopic Compositions of Maine Yankee CE 14 x 14 Fuel Assembly Non-Fuel Source Regions

Isotope	Upper Plenum [atom/b-cm]	Upper End Fit [atom/b-cm]	Lower End Fit [atom/b-cm]
CHROMIUM(SS304)	1.59190E-03	1.89910E-03	3.08125E-03
MANGANESE	1.58594E-04	1.89199E-04	3.06971E-04
IRON(SS304)	5.42166E-03	6.46791E-03	1.04941E-02
NICKEL(SS304)	7.05196E-04	8.41284E-04	1.36497E-03
ZIRCALLOY	3.22036E-03	–	–

Table 5.6.1-9 Isotopic Compositions of Maine Yankee CE 14 x 14 Canister Annular Region Materials (One-Dimensional Analysis Only)

Isotope	Fuel Annulus [atom/b-cm]	Upper Plenum Annulus [atom/b-cm]	Upper End Fit Annulus [atom/b-cm]	Lower End Fit Annulus [atom/b-cm]
ALUMINUM	5.96817E-03	–	–	–
CHROMIUM(SS304)	1.77895E-03	9.31065E-04	2.53529E-03	4.13797E-03
MANGANESE	1.77228E-04	9.27577E-05	2.52579E-04	4.12247E-04
IRON(SS304)	6.05870E-03	3.1710E-03	8.63463E-03	1.40930E-02
NICKEL(SS304)	7.88057E-04	4.12453E-04	1.12311E-03	1.83308E-03

Table 5.6.1-10 Loading Table for Maine Yankee CE 14 x 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable

Enrichment	Burnup ≤ 30 GWD/MTU - Minimum Cool Time [years] for[1]				
	Standard[2]	Pref (0.958i)	Pref (0.958p)	Pref (1.05i)	Pref (1.05p)
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time [years] for				
	Standard[2]	Pref (0.958i)	Pref (0.958p)	Pref (1.05i)	Pref (1.05p)
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time [years] for				
	Standard[2]	Pref (0.958i)	Pref (0.958p)	Pref (1.05i)	Pref (1.05p)
1.9 ≤ E < 2.1	7	7	6	15	5
2.1 ≤ E < 2.3	6	6	6	15	5
2.3 ≤ E < 2.5	6	6	5	14	5
2.5 ≤ E < 2.7	5	5	5	14	5
2.7 ≤ E < 2.9	5	5	5	14	5
2.9 ≤ E < 3.1	5	5	5	6	5
3.1 ≤ E < 3.3	5	5	5	6	5
3.3 ≤ E < 3.5	5	5	5	6	5
3.5 ≤ E < 3.7	5	5	5	6	5
3.7 ≤ E ≤ 4.2	5	5	5	6	5

1. Cool times for preferential loading of fuel assemblies with a decay heat of either 0.958 or 1.05 kw per assembly, loaded in either interior (i) or periphery (p) basket positions.
2. Fuel assemblies with cool times from 5 to 7 years must be preferentially loaded based on cool time, with fuel with the shortest cool time in the basket interior, in accordance with Section B2.1.2 in Chapter 12.

Table 5.6.1-10 Loading Table for Maine Yankee CE 14 x 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable (Continued)

Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time [years] for[1]				
	Standard[2]	**Pref(0.958i)**	**Pref(0.958p)**	**Pref(1.05i)**	**Pref(1.05p)**
1.9 ≤ E < 2.1	11	20	7	Not Allowed	6
2.1 ≤ E < 2.3	9	15	7	Not Allowed	6
2.3 ≤ E < 2.5	8	15	6	Not Allowed	6
2.5 ≤ E < 2.7	8	15	6	Not Allowed	6
2.7 ≤ E < 2.9	8	14	6	Not Allowed	6
2.9 ≤ E < 3.1	8	14	6	Not Allowed	6
3.1 ≤ E < 3.3	7	14	6	Not Allowed	5
3.3 ≤ E < 3.5	6	14	6	Not Allowed	5
3.5 ≤ E < 3.7	6	13	6	Not Allowed	5
3.7 ≤ E ≤ 4.2	6	13	6	Not Allowed	5
Enrichment	**45 < Burnup ≤ 50 GWD/MTU - Minimum Cool Time [years] for**				
	Standard	**Pref(0.958i)**	**Pref(0.958p)**	**Pref(1.05i)**	**Pref(1.05p)**
1.9 ≤ E < 2.1	Not Allowed	Not Allowed	8	Not Allowed	7
2.1 ≤ E < 2.3	Not Allowed	Not Allowed	8	Not Allowed	7
2.3 ≤ E < 2.5	Not Allowed	Not Allowed	8	Not Allowed	7
2.5 ≤ E < 2.7	Not Allowed	Not Allowed	8	Not Allowed	7
2.7 ≤ E < 2.9	Not Allowed	Not Allowed	8	Not Allowed	7
2.9 ≤ E < 3.1	Not Allowed	Not Allowed	8	Not Allowed	7
3.1 ≤ E < 3.3	Not Allowed	Not Allowed	7	Not Allowed	7
3.3 ≤ E < 3.5	Not Allowed	Not Allowed	7	Not Allowed	6
3.5 ≤ E < 3.7	Not Allowed	Not Allowed	7	Not Allowed	6
3.7 ≤ E ≤ 4.2	Not Allowed	Not Allowed	7	Not Allowed	6

1. Cool times for preferential loading of fuel assemblies with a decay heat of either 0.958 or 1.05 kw per assembly, loaded in either interior (i) or periphery (p) basket positions.
2. Fuel assemblies with cool times from 5 to 7 years must be preferentially loaded based on cool time, with fuel with the shortest cool time in the basket interior, in accordance with Section B2.1.2 in Chapter 12.

Table 5.6.1-11 Three-Dimensional Shielding Analysis Results for Various Maine Yankee CEA Configurations Establishing One-Dimensional Dose Rate Limits for Loading Table Analysis

CEA Cool Time [years]	Dose Rate [mrem/hr]	FSD	Delta [mrem/hr]	Limit [mrem/hr]
Class 1 Result	32.0	0.85%	-	34.2
NoCEA	32.0	0.85%	-0.0	34.2
05y	43.8	0.59%	-11.8	22.4
10y	33.1	0.69%	-1.1	33.1
15y	32.0	0.85%	-0.0	34.2
20y	32.0	0.85%	-0.0	34.2

Table 5.6.1-12 Loading Table for Maine Yankee CE 14 x 14 Fuel Containing CEA Cooled to Indicated Time

	≤ 30 GWD/MTU Burnup - Minimum Cool Time in Years for					
Enrichment	**No CEA (Class 1)**		**CEA 5 Year**	**10 Year CEA**	**15 Year CEA**	**20 Year CEA**
1.9 ≤ E < 2.1	5		5	5	5	5
2.1 ≤ E < 2.3	5		5	5	5	5
2.3 ≤ E < 2.5	5		5	5	5	5
2.5 ≤ E < 2.7	5		5	5	5	5
2.7 ≤ E < 2.9	5		5	5	5	5
2.9 ≤ E < 3.1	5		5	5	5	5
3.1 ≤ E < 3.3	5		5	5	5	5
3.3 ≤ E < 3.5	5		5	5	5	5
3.5 ≤ E < 3.7	5		5	5	5	5
3.7 ≤ E ≤ 4.2	5		5	5	5	5
	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time in Years for					
Enrichment	**No CEA (Class 1)**		**5 Year CEA**	**10 Year CEA**	**15 Year CEA**	**20 Year CEA**
1.9 ≤ E < 2.1	5		5	5	5	5
2.1 ≤ E < 2.3	5		5	5	5	5
2.3 ≤ E < 2.5	5		5	5	5	5
2.5 ≤ E < 2.7	5		5	5	5	5
2.7 ≤ E < 2.9	5		5	5	5	5
2.9 ≤ E < 3.1	5		5	5	5	5
3.1 ≤ E < 3.3	5		5	5	5	5
3.3 ≤ E < 3.5	5		5	5	5	5
3.5 ≤ E < 3.7	5		5	5	5	5
3.7 ≤ E ≤ 4.2	5		5	5	5	5
	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time in Years for					
Enrichment	**No CEA (Class 1)**		**5 Year CEA**	**10 Year CEA**	**15 Year CEA**	**20 Year CEA**
1.9 ≤ E < 2.1	7		7	7	7	7
2.1 ≤ E < 2.3	6		6	6	6	6
2.3 ≤ E < 2.5	6		6	6	6	6
2.5 ≤ E < 2.7	5		5	5	5	5
2.7 ≤ E < 2.9	5		5	5	5	5
2.9 ≤ E < 3.1	5		5	5	5	5
3.1 ≤ E < 3.3	5		5	5	5	5
3.3 ≤ E < 3.5	5		5	5	5	5
3.5 ≤ E < 3.7	5		5	5	5	5
3.7 ≤ E ≤ 4.2	5		5	5	5	5
	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time in Years for					
Enrichment	**No CEA (Class 1)**		**5 Year CEA**	**10 Year CEA**	**15 Year CEA**	**20 Year CEA**
1.9 ≤ E < 2.1	11		11	11	11	11
2.1 ≤ E < 2.3	9		9	9	9	9
2.3 ≤ E < 2.5	8		8	8	8	8
2.5 ≤ E < 2.7	8		8	8	8	8
2.7 ≤ E < 2.9	8		8	8	8	8
2.9 ≤ E < 3.1	8		8	8	8	8
3.1 ≤ E < 3.3	7		7	8	8	8
3.3 ≤ E < 3.5	6		6	7	7	7
3.5 ≤ E < 3.7	6		6	6	6	6
3.7 ≤ E ≤ 4.2	6		6	6	6	6

Note: The NoCEA (Class 2) column is provided for comparison. Fuel assemblies without a CEA insert may not be loaded in a Class 2 canister.

Table 5.6.1-13 Establishment of Dose Rate Limit for Maine Yankee ICI Thimble Analysis

Case	Top Model	
	Rate (mrem/hr)	FSD
No ICI Thimble	33.3	1.4%
4 Year Cooled ICI Thimble	33.3	1.4%
Delta	0.0	

Table 5.6.1-14 Required Cool Time for Maine Yankee Fuel Assemblies with Activated Stainless Steel Replacement Rods

Assembly Number	Burnup [MWD/MTU]	Enrichment [wt %]	SSR Source [g/s/assy]	Cool Time [years]	Earliest Loadable	Loading Configuration
N420	45,000	3.3	2.1602E+13	6	Jan 2001	Standard
N842	35,000	3.3	3.1396E+12	5	Jan 2001	Standard
N868	40,000	3.3	5.2444E+12	5	Jan 2001	Standard
R032	45,000	3.5	1.4550E+13	6	Jan 2002	Standard
R439	50,000	3.5	1.3998E+13	10	Jan 2006	Standard
R444	50,000	3.5	5.5993E+13	10	Jan 2006	Standard
R439	50,000	3.5	1.3998E+13	6	Jan 2002	Pref(1.050)
R444	50,000	3.5	5.5993E+13	6	Jan 2002	Pref(1.050)

Table 5.6.1-15 Maine Yankee Consolidated Fuel Model Parameters

Lattice	Assy	Num Rods	Actual Burnup [MWD/MTU]	Actual Enrichment [wt %]	Modeled Burnup [MWD/MTU]	Modeled Enrichment [wt %]	Required Cool Time [y]	Cool Time 1/1/01 [y]
CN-1	EF0039	172	5150	1.929	30000	1.9	6	26
CN-10	EF0045	176	17150	1.953	30000	1.9	6	24
	EF0046	107	17150	1.953	30000	1.9	6	24

Table 5.6.1-16 Maine Yankee Source Rate Analysis for CN-10 Consolidated Fuel Lattice

Cool Time [years]	Num Rods Present	Decay Heat [kW/cask]	Fuel Neutron [n/s/assy]	Fuel Gamma [g/sec/assy]	Fuel Hardware [g/sec/assy]
6	176	13.9	1.63E+08	3.16E+15	9.28E+12
24	283	7.42	8.41E+07	1.28E+15	8.67E+11
Src Ratio 24/6		0.86	0.83	0.65	0.15

Table 5.6.1-17 Additional Maine Yankee Non-Fuel Hardware Characterization – Non-Neutron Sources

Non Fuel Material	Waste Volume [ft^3]	Total Curies	Co-60 Curies
Sb-Be Source 1H1	0.020	4.15E+02	2.22E+02
Sb-Be Source 6H4	0.020	4.32E+02	2.31E+02
CEA Tips	0.100	1.06E+02	8.90E+01
ICI	0.007	2.82E+01	1.76E+01

Table 5.6.1-18 Additional Maine Yankee Non-Fuel Hardware Characterization – Neutron Sources

Non Fuel Material	Pu-238 grams	Pu-238 Curies	Pu-239 grams	Pu-239 Curies
Boronometer Source	-	323	-	1.18
Pu-Be Unirradiated Source	1.16	-	0.24	-
Pu-Be Irradiated Sources	1.16	5.10E-02	0.24	5.88E-05

Table 5.6.1-19 Pu-Be Assembly Hardware Spectra (Cycles 1-13) – 5 Year Cool Time from 1/1/1997

Group	Pu-Be SS Hardware [g/sec]
1	0.0000E+00
2	0.0000E+00
3	0.0000E+00
4	0.0000E+00
5	1.8059E-15
6	3.5714E+05
7	2.3032E+08
8	8.9078E-03
9	9.7053E+12
10	3.4367E+13
11	1.2604E+10
12	4.0605E+07
13	1.1692E+08
14	1.8500E+09
15	1.4100E+09
16	2.8397E+10
17	1.1771E+11
18	5.9808E+11
TOTAL	8.4119E+12

Table 5.6.1-20 Additional Maine Yankee Non-Fuel Hardware – HW Assembly Spectra (Class 2 Canister) – 5 Year Cool Time from 1/1/1997

Group	ICI Segment [g/sec]	CEA Tips [g/sec]	Boronometer [g/sec]	Total Gamma [g/sec]	Boronometer [n/sec]
1	0.0000E+00	0.0000E+00	3.7397E+01	3.7397E+01	9.657E+02
2	0.0000E+00	0.0000E+00	1.8333E+02	1.8333E+02	6.471E+04
3	0.0000E+00	0.0000E+00	9.8804E+02	9.8804E+02	1.645E+05
4	0.0000E+00	0.0000E+00	2.6109E+03	2.6109E+03	4.776E+04
5	0.0000E+00	0.0000E+00	8.2579E+03	8.2579E+03	3.228E+04
6	5.6364E+04	1.4995E+04	9.7070E+03	8.1066E+04	1.679E+04
7	3.6350E+07	9.6704E+06	1.7507E+04	4.6038E+07	3.023E+03
8	0.0000E+00	0.0000E+00	3.0335E+04	3.0335E+04	-
9	1.5317E+12	4.0749E+11	2.9757E-03	1.9392E+12	-
10	5.4239E+12	1.4430E+12	1.8213E+05	6.8669E+12	-
11	2.4164E+08	6.4285E+07	7.9962E+05	3.0672E+08	-
12	6.4084E+06	1.7049E+06	6.1465E+06	1.4260E+07	-
13	1.8453E+07	4.9092E+06	6.9158E+05	2.4054E+07	-
14	2.9197E+08	7.7675E+07	2.1223E+06	3.7177E+08	-
15	2.2253E+08	5.9201E+07	7.3513E+05	2.8247E+08	-
16	4.4816E+09	1.1923E+09	4.1772E+08	6.0916E+09	-
17	1.8576E+10	4.9418E+09	5.9152E+08	2.4109E+10	-
18	9.3171E+10	2.4787E+10	6.2953E+11	7.4749E+11	-
Total	7.0726E+12	1.8816E+12	6.3055E+11	9.5848E+12	3.30E+05

Table 5.6.1-21 Additional Maine Yankee Non-Fuel Hardware – Source Assembly Spectra – 5 Year Cool Time from 1/1/1997

	Sb-Be Source	Pu-Be Unirradiated Source		Pu-Be Irradiated Source			
Group	Gamma [g/sec]	Gamma [g/sec]	Neutron [n/sec]	Gamma [g/sec]	Hw Gamma [g/sec]	Total Gamma [g/sec]	Neutron [n/sec]
1	0.0000E+00	1.8438E+00	4.7620E+01	5.9037E-03	0.0000E+00	5.9037E-03	1.5250E-01
2	0.0000E+00	9.0379E+00	3.1850E+03	2.8938E-02	0.0000E+00	2.8938E-02	1.0200E+01
3	0.0000E+00	4.8704E+01	8.0950E+03	1.5595E-01	0.0000E+00	1.5595E-01	2.5920E+01
4	0.0000E+00	1.2868E+02	2.3510E+03	4.1204E-01	0.0000E+00	4.1204E-01	7.5290E+00
5	0.0000E+00	4.0697E+02	1.5900E+03	1.3030E+00	1.8059E-15	1.3030E+00	5.0900E+00
6	2.2971E+05	4.7836E+02	8.2740E+02	1.5315E+00	3.5714E+05	3.5714E+05	2.6490E+00
7	1.4814E+08	8.6530E+02	1.4900E+02	2.7621E+00	2.3032E+08	2.3032E+08	4.7700E-01
8	0.0000E+00	1.5016E+03	-	4.7854E+00	8.9078E-03	4.7943E+00	-
9	6.2425E+12	4.2159E+00	-	4.6985E-07	9.7053E+12	9.7053E+12	-
10	2.2105E+13	8.9859E+03	-	2.8745E+01	3.4367E+13	3.4367E+13	-
11	9.8479E+08	3.9420E+04	-	1.2621E+02	1.2604E+10	1.2604E+10	-
12	2.6117E+07	3.0176E+05	-	9.6649E+02	4.0605E+07	4.0606E+07	-
13	7.5204E+07	8.7531E+03	-	3.4464E+01	1.1692E+08	1.1692E+08	-
14	1.1899E+09	2.6915E+04	-	1.0614E+02	1.8500E+09	1.8500E+09	-
15	9.0690E+08	2.5370E+04	-	8.3993E+01	1.4100E+09	1.4100E+09	-
16	1.8265E+10	2.0487E+07	-	6.5574E+04	2.8397E+10	2.8397E+10	-
17	7.5705E+10	2.8935E+07	-	9.2577E+04	1.1771E+11	1.1771E+11	-
18	3.7972E+11	3.1017E+10	-	9.9310E+07	5.9808E+11	5.9818E+11	-
Total	2.8825E+13	3.1067E+10	1.625E+04	9.9470E+07	4.4833E+13	4.4833E+13	5.202E+01

Table 5.6.1-22 Additional Maine Yankee Non-Fuel Hardware – Hardware Assembly Dose Rates (Class 2) – 5 Years Cooled from 1/1/1997

	Storage - Surface Dose		Transfer – Surface Dose	
Group	**Gamma Dose [mrem/hr]**	**Neutron Dose [mrem/hr]**	**Gamma Dose [mrem/hr]**	**Neutron Dose [mrem/hr]**
1	3.66E-10	5.96E-07	1.51E-10	6.90E-05
2	1.41E-09	2.26E-05	8.97E-10	3.05E-03
3	4.92E-09	4.98E-05	5.00E-09	7.26E-03
4	7.10E-09	1.13E-05	1.20E-08	1.48E-03
5	1.08E-08	6.68E-06	2.99E-08	7.42E-04
6	4.21E-08	3.29E-06	1.91E-07	2.72E-04
7	9.96E-06	4.45E-07	6.12E-05	1.36E-05
8	2.24E-09	-	1.72E-08	-
9	4.59E-02	-	3.77E-01	-
10	3.49E-02	-	2.24E-01	-
11	2.31E-07	-	6.42E-07	-
12	1.82E-09	-	1.02E-09	-
13	2.68E-10	-	9.13E-13	-
14	9.84E-11	-	3.12E-19	-
15	2.65E-12	-	1.49E-40	-
16	1.11E-14	-	0.00E+00	-
17	1.91E-41	-	0.00E+00	-
18	0.00E+00	-	0.00E+00	-
Total	8.09E-02	9.48E-05	6.01E-01	1.29E-02

Table 5.6.1-23 Additional Maine Yankee Non-Fuel Hardware – Storage Cask Source Assembly Surface Dose Rates – 5 Years Cooled from 1/1/1997

	Sb-Be Source Dose	Pu-Be Unirradiated Source Dose		Pu-Be Irradiated Source Dose	
Group	Gamma [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]
1	0.00E+00	1.81E-11	2.94E-08	5.78E-14	9.41E-11
2	0.00E+00	6.93E-11	1.11E-06	2.22E-13	3.57E-09
3	0.00E+00	2.42E-10	2.45E-06	7.76E-13	7.85E-09
4	0.00E+00	3.50E-10	5.57E-07	1.12E-12	1.78E-09
5	0.00E+00	5.31E-10	3.29E-07	1.70E-12	1.05E-09
6	1.19E-07	2.49E-10	1.62E-07	1.86E-07	5.19E-10
7	3.21E-05	1.87E-10	2.19E-08	4.99E-05	7.02E-11
8	0.00E+00	1.11E-10	-	3.53E-13	-
9	1.48E-01	9.99E-14	-	2.30E-01	-
10	1.12E-01	4.57E-11	-	1.75E-01	-
11	7.41E-07	2.97E-11	-	9.48E-06	-
12	3.34E-09	3.86E-11	-	5.19E-09	-
13	8.37E-10	9.74E-14	-	1.30E-09	-
14	3.15E-10	7.13E-15	-	4.90E-10	-
15	8.52E-12	2.38E-16	-	1.32E-11	-
16	3.34E-14	3.74E-17	-	5.19E-14	-
17	5.99E-41	2.29E-44	-	9.31E-41	-
18	0.00E+00	0.00E+00	-	0.00E+00	-
Total	2.60E-01	1.87E-09	4.67E-06	4.05E-01	1.49E-08

Table 5.6.1-24 Additional Maine Yankee Non-Fuel Hardware – Transfer Cask Source Assembly Surface Dose Rates – 5 Years Cooled from 1/1/1997

	Sb-Be Source Dose	Pu-Be Unirradiated Source Dose		Pu-Be Irradiated Source Dose	
Group	Gamma [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]	Gamma [mrem/hr]	Neutron [mrem/hr]
1	0.00E+00	7.43E-12	3.40E-06	2.38E-14	1.09E-08
2	0.00E+00	4.42E-11	1.50E-04	1.42E-13	4.81E-07
3	0.00E+00	2.46E-10	3.57E-04	7.89E-13	1.14E-06
4	0.00E+00	5.90E-10	7.29E-05	1.89E-12	2.33E-07
5	0.00E+00	1.47E-09	3.65E-05	4.72E-12	1.17E-07
6	5.40E-07	1.12E-09	1.34E-05	8.40E-07	4.30E-08
7	1.97E-04	1.15E-09	6.69E-07	3.06E-04	2.14E-09
8	0.00E+00	8.53E-10	-	2.72E-12	-
9	1.21E+00	8.20E-13	-	1.89E+00	-
10	7.21E-01	2.93E-10	-	1.12E+00	-
11	2.06E-06	8.25E-11	-	2.64E-05	-
12	1.86E-09	2.15E-11	-	2.89E-09	-
13	2.85E-12	3.32E-16	-	4.44E-12	-
14	9.99E-19	2.26E-23	-	1.55E-18	-
15	4.77E-40	1.33E-44	-	7.42E-40	-
16	0.00E+00	0.00E+00	-	0.00E+00	-
17	0.00E+00	0.00E+00	-	0.00E+00	-
18	0.00E+00	0.00E+00	-	0.00E+00	-
Total	1.94E+00	5.89E-09	6.34E-04	3.01E+00	2.03E-06

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6.5 Critical Benchmark Experiments

This section provides the validation of the CSAS25 criticality analysis sequence contained in Version 4.3 and Version 4.4 of the SCALE package. This validation is required by the criticality safety standards ANSI/ANS-8.1 [11]. The section describes the method, computer program and cross-section libraries used, experimental data, areas of applicability, and bias and margins of safety.

ANSI/ANS-8.17 [12] prescribes the criterion to establish subcriticality safety margins. This criterion is as follows:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad (1)$$

where:

k_s = calculated allowable maximum multiplication factor, k_{eff}, of system being evaluated for all normal or credible abnormal conditions or events.

k_c = mean k_{eff} that results from calculation of benchmark criticality experiments using particular calculational method. If calculated k_{eff} values for criticality experiments exhibit trend with parameter, then k_c shall be determined by extrapolation based on best fit to calculated values. Criticality experiments used as benchmarks in computing k_c should have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of system being evaluated.

Δk_s = allowance for

a. statistical or convergence uncertainties, or both, in computation of k_s,
b. material and fabrication tolerances, and
c. geometric or material representations used in computational method.

Δk_c = margin for uncertainty in k_c which includes allowance for

a. uncertainties in critical experiments,
b. statistical or convergence uncertainties, or both, in computation of k_c,
c. uncertainties resulting from extrapolation of k_c outside range of experimental data, and

d. uncertainties resulting from limitations in geometrical or material representations used in computational method.

Δk_m = arbitrary margin to ensure subcriticality of k_s.

The various uncertainties are combined statistically if they are independent. Correlated uncertainties are combined by addition.

Equation 1 can be rewritten as:

$$k_s \leq 1 - \Delta k_m - \Delta k_s - (1-k_c) - \Delta k_c \qquad (2)$$

Noting that the NRC requires a 5% subcriticality margin ($\Delta k_m = 0.05$) and the definition of the bias ($\beta = 1-k_c$), the equation 2 can then be written as:

$$k_s \leq 0.95 - \Delta k_s - \beta - \Delta\beta \qquad (3)$$

where $\Delta\beta = \Delta k_c$. Thus, the k_s (the maximum allowable value for k_{eff}) must be below 0.95 minus the bias, uncertainties in the bias, and uncertainties in the system being analyzed (i.e., Monte Carlo, mechanical, and modeling). This is an upper safety limit criteria often used in the DOE criticality safety community.

Alternatively, equation 3 can be rewritten applying the bias and uncertainties to the k_{eff} of the system being analyzed as:

$$k_s \equiv k_{eff} + \Delta k_s + \beta + \Delta\beta \leq 0.95 \qquad (4)$$

In Equation 4, k_{eff} replaces k_s, and k_s has been redefined as the effective multiplication factor of the system being analyzed, including the method bias and all uncertainties. This is a maximum calculated k_{eff} criteria often used in light water reactor spent fuel storage and transport analyses.

Both β and $\Delta\beta$ are evaluated below for KENO-Va with the 27-group ENDF/B-IV library for use in criticality evaluations of light water reactor fuel in storage and transport casks.

6.5.1 SCALE 4.3 Benchmark Experiments and Applicability

The criticality safety method is CSAS embedded in SCALE version 4.3 for the PC. CSAS includes the SCALE Material Information Processor, BONAMI-S, NITAWL-S, and KENO-Va. The Material Information Processor generates number densities for standard compositions, prepares geometry data for resonance self-shielding, and creates data input files for the cross-section processing codes. The BONAMI-S and NITAWL-S codes are used to prepare a resonance-corrected cross-section library in AMPX working format. The KENO-Va code uses Monte Carlo techniques to calculate the model k_{eff}. The 27-group ENDF/B-IV neutron cross-section library is used in this validation.

6.5.1.1 Description of Experiments

The 63 critical experiments selected are as follows: nine B&W 2.46 wt % ^{235}U fuel storage [13], ten PNL 4.31 wt % ^{235}U lattice [14], twenty-one PNL 2.35 and 4.31 wt % ^{235}U with metal reflectors (Bierman, April 1979 and August 1981) [15, 16], twelve PNL flux trap [14, 17] and eleven VCML 4.74 wt % ^{235}U experiments, some involving moderator density variations [18]. These experiments span a range of fuel enrichments, fuel rod pitches, neutron absorber sheet characteristics, shielding materials and geometries that are typical of light water reactor fuel in a cask.

To achieve accurate results, three-dimensional models, as close to the actual experiment as possible, are used to evaluate the experiments. Stochastic Monte Carlo error is kept within ±0.1% by executing at least 1,000 neutrons/generation for more than 400 generations.

6.5.1.2 Applicability of Experiments

All of the experiments chosen in this validation are applicable to either PWR or BWR fuel. Fuel enrichments have covered a range from 2.35 up to 4.74 wt % ^{235}U, typical of light water reactor fuel presently used. The experiment fuel rod and pitch characteristics are within the range of standard PWR or BWR fuel rods (i.e., pellet OD from 0.78 to 1.2 cm, rod OD from 0.95 to 1.88 cm, and pitch from 1.26 to 1.87 cm). This is particularly true of the VCML (PWR rod type) and B&W experiments (BWR rod type). The H/U volume ratios of the experimental fuel arrays are within the range of PWR fuel assemblies (1.6 to 2.32) and BWR fuel assemblies (1.6 to 1.9). Experiments covered the geometry and neutron absorber sheet arrangements typical of NAC basket designs. Flux trap gap spacings of 3.81 cm such as those in the NAC-STC and UMS® PWR

baskets and gap spacings as low 1.91 cm as in the NAC-MPC were included. ^{10}B neutron absorber loadings, also typical of NAC basket designs (0.005 to 0.025g $^{10}B/cm^2$), were included as well. The experiments addressed the influence of water and metal reflector regions, including steel and lead, that would be present in storage and transport cask shielding.

Confidence in predicting criticality, including bias and uncertainty, has been demonstrated for light water reactor fuel with enrichments up to 4.74 wt % ^{235}U and results indicate confidence well above 5 wt % ^{235}U. Confidence in predicting criticality has been demonstrated for storage and transport arrays in which critical controls consist of flux trap or single neutron absorber sheets or simple spacing. Confidence in predicting criticality has been demonstrated for light water reactor fuel storage and transport arrays next to water and metal reflector regions.

6.5.1.3 Results of Benchmark Calculations

The k-effective results for the experiments are shown in Table 6.5.1-1 and a frequency plot is provided in Figure 6.5.1-1. Five sets of cases are presented: Set 1, B&W; Set 2, PNL lattice; Set 3, PNL reflector; Set 4, PNL flux trap, and Set 5,VCML critical experiments. Sixty-three results are reported.

The overall average and standard deviation of the 63 cases is 0.9948±0.0044. The average Monte Carlo error (statistical convergence) is ±0.0012 for the 63 cases. This uncertainty component is statistically subtracted from the uncertainties, because it is previously included in the standard deviation. The KENO-Va models are three-dimensional, fully explicit representations (no homogenization) of the experimental geometry. Therefore, the uncertainty resulting from limitations of geometrical modeling is taken to be 0.0. The experiments modeled cover the range of fuel types, enrichments, neutron absorber configurations, neutron absorber B^{10} loading, and metal reflector effects so that no extrapolations are necessary outside the range of data, and the uncertainty resulting from extrapolation is also taken to be 0.0.

On the basis of the reported experimental error for the B&W cases, the reported error of the critical size number of rods for the PNL cases and the reported error for the critical height in the VCML cases, the experimental error is conservatively taken to be ±0.001. Criticality can then be represented as 1.000±0.001. This uncertainty component is statistically added to the sum of the other uncertainties, because the bias is the difference between two random variates (i.e., criticality and code prediction, and the uncertainty in the difference between two random variables is the statistical sum [(rms)] of their individual uncertainties).

Thus, the bias or average difference between code calculated and the critical condition is β=1-0.9948 = 0.0052. The uncertainty in the bias, accounting for the statistical convergence (Monte Carlo error) and the uncertainty in criticality is $(0.0044^2 - 0.0012^2 + 0.0010^2)^{1/2} = 0.0043$. For 63 samples of criticality, the 95/95 one-side tolerance factor is 2.012 [19]. The result is a 95/95 one-sided uncertainty in the bias of $\Delta\beta$=2.012x0.0043=0.0087. Equation 3 now becomes:

$$k_{eff} + \Delta k_s + 0.0052 + 0.0087 \leq 0.95 \qquad (5)$$

where Δk_s becomes the uncertainty in k_s resulting from Monte Carlo error, mechanical and material tolerances, and geometric or material representations. If the nominal representation of the system is evaluated for k_s, then the mechanical and material perturbations can be evaluated independently and can be combined statistically as the root sum of squares. If the worst-case mechanical and material tolerances are used to calculate k_s (e. g., 75% of boron loading and most reactive positioning of fuel or basket components), then Δk_s becomes 0.0 and the Monte Carlo error, σ_{mc}, can be combined statistically, because it is independent, with the uncertainty in the bias as:

$$k_{eff} + 0.0052 + \sqrt{0.0087^2 + (2\sigma)^2} \leq 0.95 \qquad (6)$$

6.5.1.4 Trends

Scatter plots of k_{eff} versus wt % ^{235}U, rod pitch, H/U volume ratio, average neutron group causing fission, ^{10}B loading for flux trap cases, and flux trap gap thickness are shown in Figures 6.5.1-2 through 6.5.1-7. Included in these scatter plots are linear regression lines with a corresponding correlation coefficient (r) to statistically indicate any trend or lack thereof. In particular, the correlation coefficient is a measure of the linear relationship between k_{eff} and a critical experiment parameter. If r is +1, a perfect linear relationship with a positive slope is indicated, and if r is -1, a perfect linear relationship with a negative slope is indicated. When r is 0, no linear relationship is indicated.

The largest correlation coefficient indicated in the plots is 0.3608 (k_{eff} versus enrichment) and the lowest is 0.0693 (k_{eff} versus ^{10}B loading in flux trap experiments). On the basis of the correlation coefficients, no statistically significant trends exist over the range of variables studied. Most importantly, no trend is shown with flux trap gap spacing and/or ^{10}B loading. This is the major criticality control feature of the UMS® Storage System basket.

6.5.1.5 Comparison of NAC Method to NUREG/CR-6361 – SCALE 4.3

NUREG/CR-6361, "Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages" (NUREG), provides a guide to LWR criticality benchmark calculations and the determination of bias and subcritical limits in critical safety evaluations. In Section 2 of the NUREG, a series of LWR critical experiments are described in sufficient detail for independent modeling. In Section 3, the critical experiments are modeled, and the results (k_{eff} values) are presented. The method utilized in the NUREG is KENO-Va with the 44 group ENDF/B-V cross section library embedded in SCALE 4.3. Inputs are provided in Appendix A of the NUREG. In Section 4, a guide for the determination of bias and subcritical safety limits is provided based on ANSI/ANS-8.17 and statistical analysis of the trending in the bias. Finally, guidelines for experiment selection and applicability are presented in Section 5. The approach outlined in Section 4 of the NUREG is described in detail below and is compared to the NAC approach presented in Sections 6.5.1, 6.5.1.1 and 6.5.1.2.

NAC has performed an extensive LWR critical benchmarking as documented in Sections 6.5.1.1 and 6.5.1.2. The method used in NAC benchmarking/validation included the CSAS25 (KENO-Va) criticality analysis sequence, with the 27-group ENDF/B-IV library, contained in SCALE 4.3. Trending in k_{eff} was evaluated for the following independent variables: wt % ^{235}U, rod pitch, H/U volume ratio, average neutron group causing fission, ^{10}B loading for flux trap cases, and flux trap gap thickness. No statistically significant trends were found, and a constant bias with associated uncertainty was determined for criticality evaluation.

Both the NUREG/CR-6361 and the NAC approach to criticality evaluation start with ANSI/ANS-8.17 criticality safety criterion. This criterion is as follows:

$$k_s \leq k_c - \Delta k_s - \Delta k_c - \Delta k_m \qquad (1)$$

where:

k_s = calculated allowable maximum multiplication factor, k_{eff}, of the system being evaluated for all normal or credible abnormal conditions or events.

k_c = mean k_{eff} that results from a calculation of benchmark criticality experiments using a particular calculation method. If the calculated k_{eff} values for the criticality experiments exhibit a trend with an independent parameter, then k_c shall be determined by extrapolation based on best fit to calculated values. Criticality experiments used as benchmarks in computing k_c should have physical compositions, configurations, and nuclear characteristics (including reflectors) similar to those of the system being evaluated.

$\Delta k_s =$ allowance for:

a) statistical or convergence uncertainties, or both, in computation of k_s,
b) material and fabrication tolerances, and
c) geometric or material representations used in computational method.

$\Delta k_c =$ margin for uncertainty in k_c which includes allowance for:

a) uncertainties in critical experiments,
b) statistical or convergence uncertainties, or both, in computation of k_c,
c) uncertainties resulting from extrapolation of k_c outside range of experimental data, and
d) uncertainties resulting from limitations in geometrical or material representations used in the computational method.

$\Delta k_m =$ arbitrary administrative margin to ensure subcriticality of k_s

The various uncertainties are combined statistically if they are independent. Correlated uncertainties are combined by addition.

Equation 1 can be rewritten as:

$$k_s \leq 1 - \Delta k_m - \Delta k_s - (1 - k_c) - \Delta k_c \qquad (2)$$

Noting that the definition of the bias is $\beta = 1 - k_c$, Equation 2 can be written as:

$$k_s + \Delta k_s \leq 1 - \Delta k_m - \beta - \Delta\beta \qquad (3)$$

where $\Delta\beta = \Delta k_c$. Thus, the maximum allowable value for k_{eff} plus uncertainties in the system being analyzed must be below 1 minus an administrative margin (typically 0.05), which includes the bias and the uncertainty in the bias. This can also be written as:

$$k_s + \Delta k_s \leq \text{Upper Subcritical Limit (USL)} \qquad (4)$$

where:

$$\text{USL} \equiv 1 - \Delta k_m - \beta - \Delta\beta \qquad (5)$$

This is the Upper Subcritical Limit criterion as described in Section 4 of NUREG/CR-6361. Two methods are prescribed for the statistical determination of the USL: Confidence Band with

Administrative Margin (USL-1) and Single Sided Uniform with Close Approach (USL-2). In the first method, $\Delta k_m = 0.05$ and a lower confidence band (usually 95%) is specified based on a linear regression of k_{eff} as a function of some system parameter. In the second method, the arbitrary administrative margin is set to zero and a uniform lower tolerance band is determined based on a linear regression. The second method provides a criticality safety margin that is generally less than 0.05. In cases where there are a limited number of data points, this method may indicate the need for a larger administrative margin. In both cases, all of the significant system parameters need to be studied to determine the strongest correlation.

In the analyses presented in Section 6.5.1.2, the bias and uncertainties are applied directly to the estimate of the system k_{eff}. Noting that the NRC requires a 5% subcriticality margin ($\Delta k_m = 0.05$), Equation 3 can be rewritten applying the bias and uncertainty in the bias to the k_{eff} of the system being analyzed as:

$$k_s + \Delta k_s + \beta + \Delta\beta \leq 0.95 \qquad (6)$$

In Equation 6, the method bias and all uncertainties are added to k_s. This is the maximum k_{eff} criterion defined in Section 6.5.1.2.

To this point, both the USL criterion and maximum k_{eff} criterion are equivalent. The effects of trending in the bias or the uncertainty in the bias can be directly incorporated into either Equation 5 or Equation 6. Trending is established by performing a regression analysis of k_{eff} as a function of the principle system variables such as: enrichment, rod pitch, H to U ratio, average group of fission, ^{10}B absorber loading and flux trap gap spacing. Usually, simple linear regression is performed, and the line with the greatest correlation is used to functionalize β. This approach is recommended in NUREG/CR-6361. However, if no strong correlation can be determined, then a constant bias adjustment can be made. This is typically done with a one-side tolerance factor that guarantees 95% confidence in the uncertainty in the bias. This is the approach taken in the UMS criticality analysis.

Both NUREG/CR-6361 and the NAC evaluation perform regression analysis on key system parameters. For all of the major system parameters, the evaluation found no strong correlation. This is based on the observation that the correlation coefficients are all much less than ± 1. Thus a constant bias with a 95/95 confidence factor is applied to the system k_{eff}. NAC's statistical analysis of the k_{eff} results produced a bias of 0.0052 and a 95/95 uncertainty of 0.0087. Adding the two together and subtracting from 0.95 yields an effective constant USL of 0.9361.

To assure compliance with NUREG/CR-6361, an upper safety limit is generated using USLSTATS and is compared to the constant NAC bias and bias uncertainty used in Section 6.5.1.2.

To evaluate the relative importance of the trend analysis to the upper subcritical limits, correlation coefficients are required for all independent parameters. Table 6.5.1-2 contains the correlation coefficient, R, for each linear fit of k_{eff} versus experimental parameter (data is extracted from Figures 6.5.1-2 through 6.5.1-7 by taking the square root of the R^2 value). Based on the highest correlation coefficient and the method presented in NUREG/CR-6361, a USL is established based on the variation of k_{eff} with enrichment. Note that even the enrichment function shows a low statistical correlation coefficient (an |R| equal or near 1 would indicate a good fit). The output generated by USLSTATS is shown in Figure 6.5.1-8.

The NAC applied USL of 0.9361 bounds the calculated upper subcritical limits for all enrichment values above 3.0 wt % ^{235}U. Since the maximum reactivities in the UMS® are calculated at enrichments well above this level, the existing bias bounds the NUREG calculated USL. The parameters of the most reactive cask configuration are presented in Table 6.5.1-3. The most reactive UMS® configuration is the PWR basket configuration with Westinghouse 17 x 17 OFA fuel assemblies.

Figure 6.5.1-1 KENO-Va Validation—27-Group Library Results: Frequency Distribution of k_{eff} Values

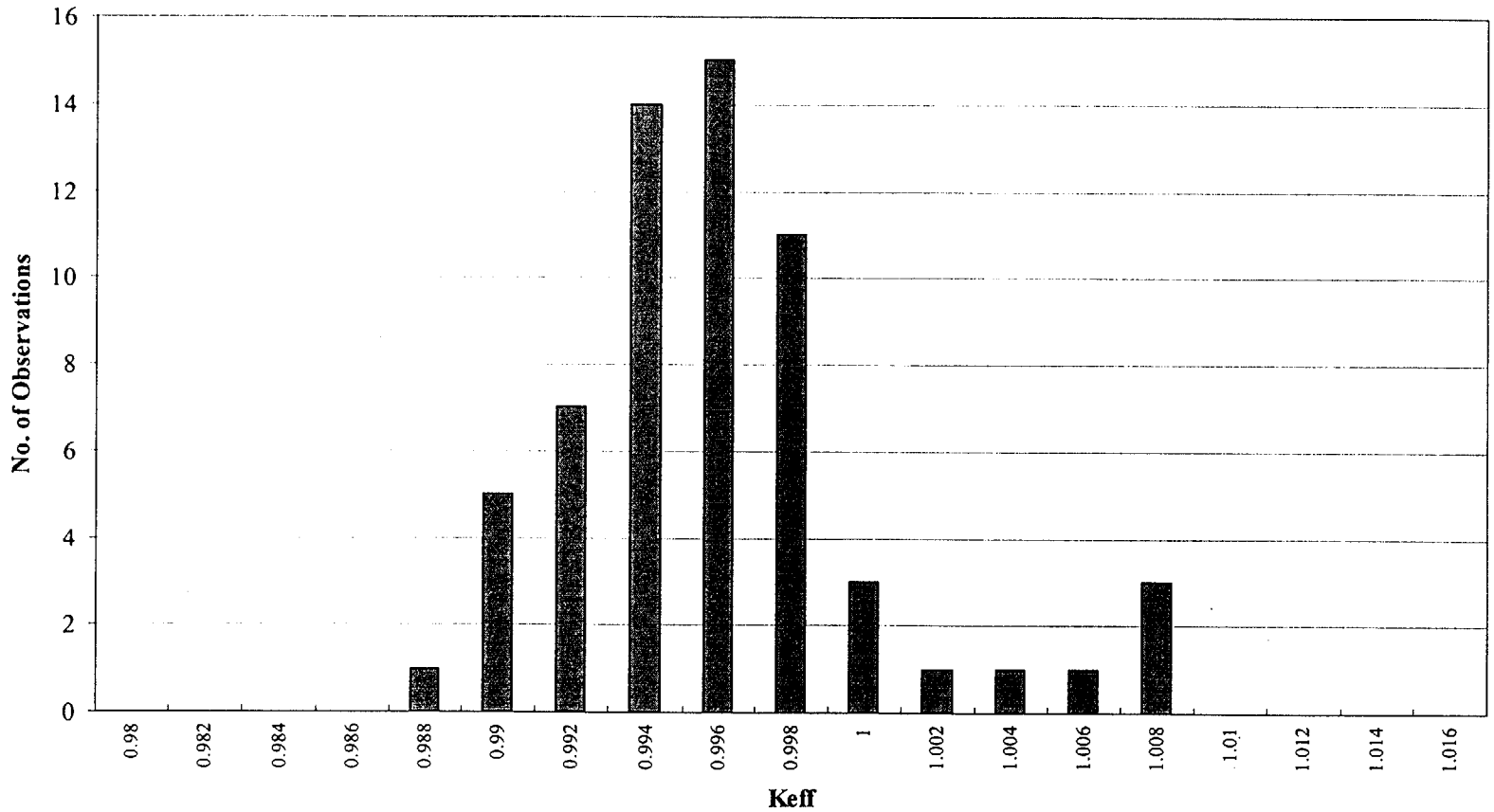

Figure 6.5.1-2 KENO-Va Validation—27-Group Library Results: k_{eff} versus Enrichment

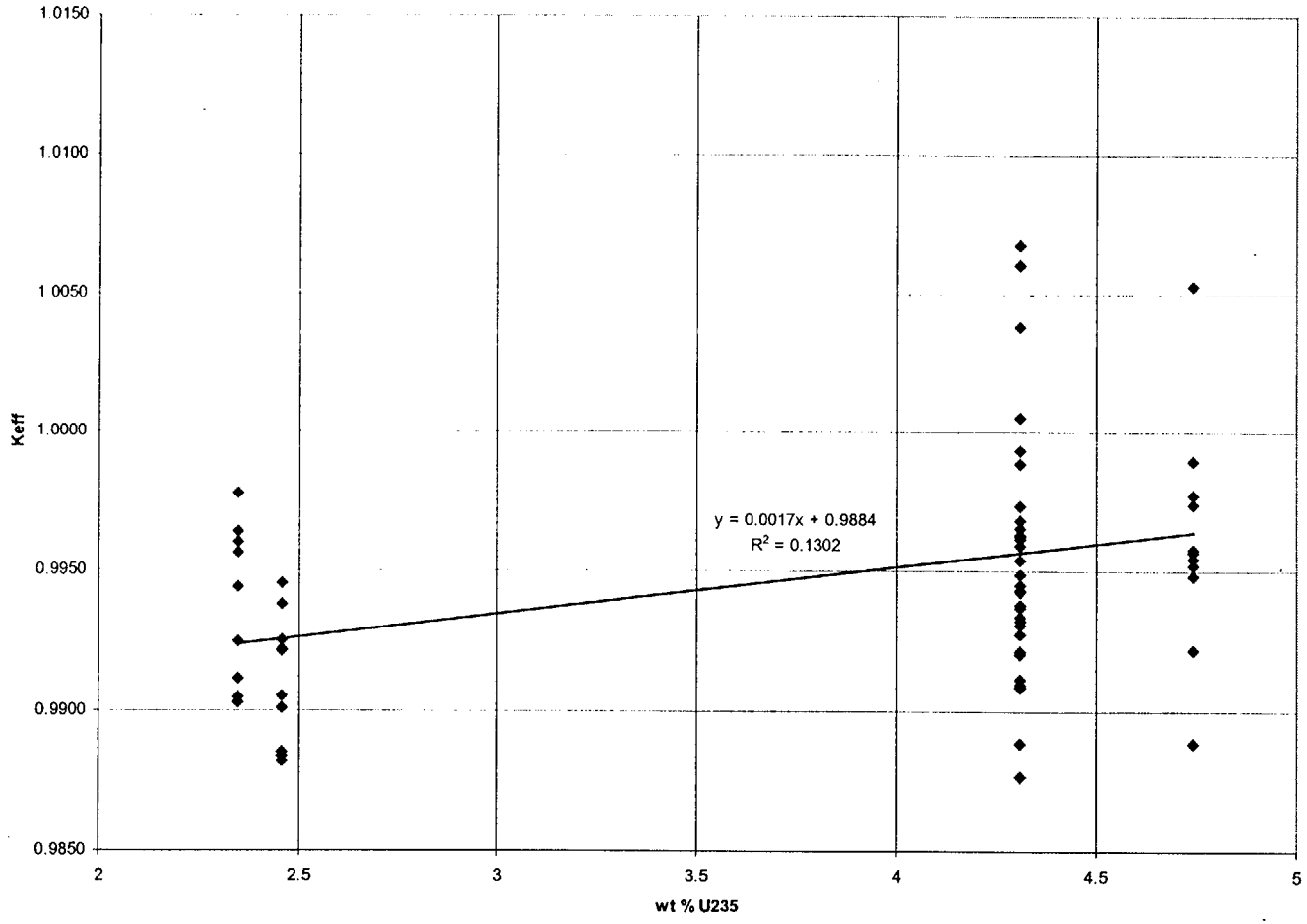

Figure 6.5.1-3 KENO-Va Validation—27-Group Library Results: k_{eff} versus Rod Pitch

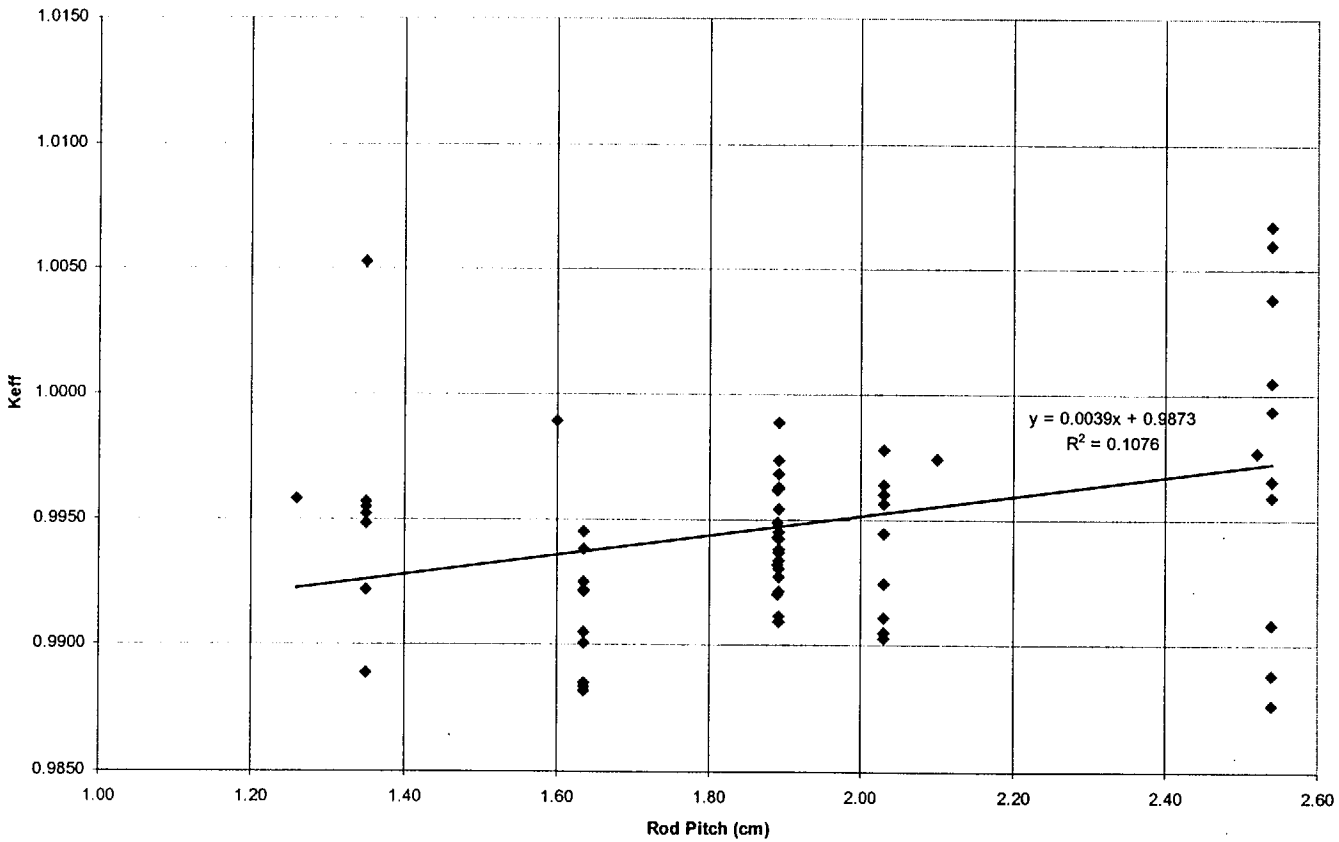

Figure 6.5.1-4 KENO-Va Validation—27-Group Library Results: k_{eff} versus H/U Volume Ratio

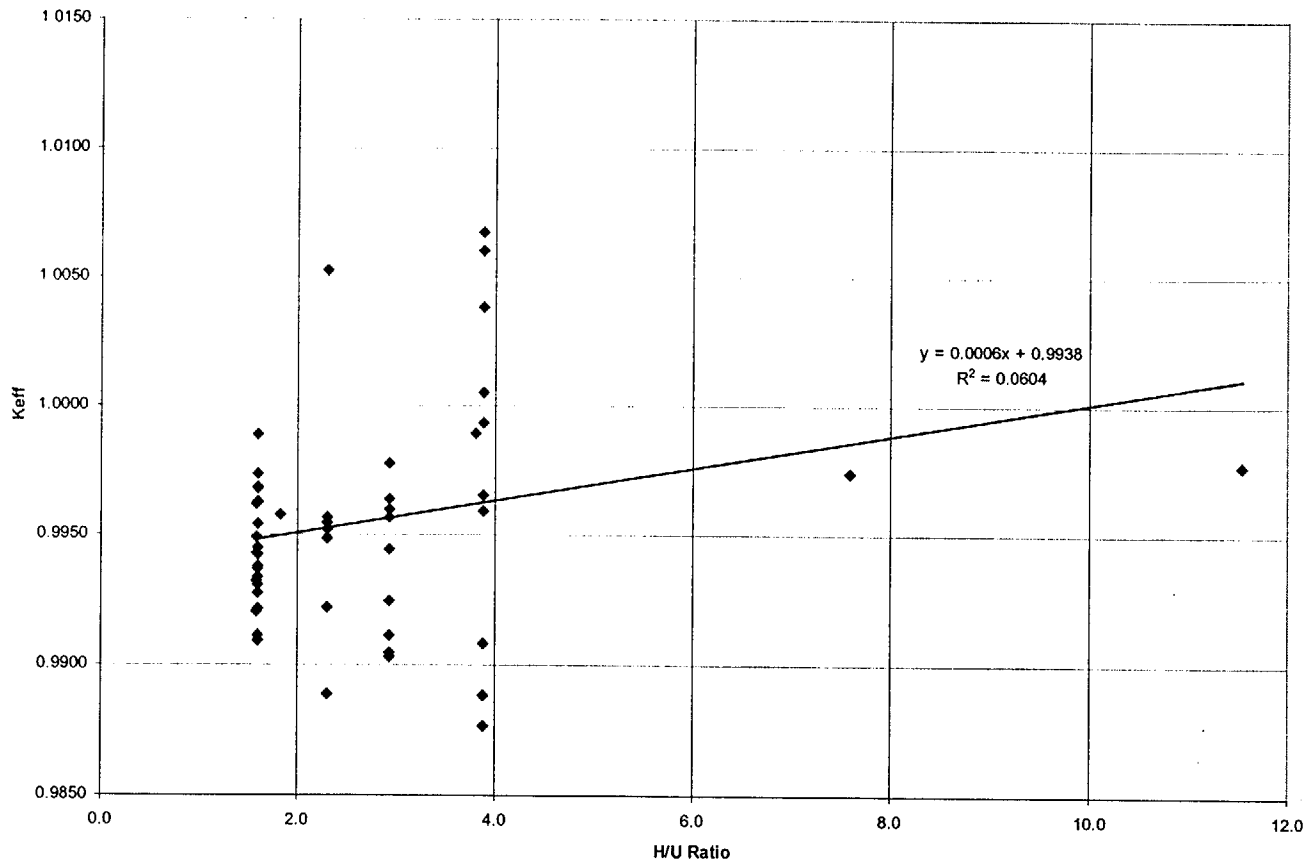

Figure 6.5.1-5 KENO-Va Validation—27-Group Library Results: k_{eff} versus Average Group of Fission

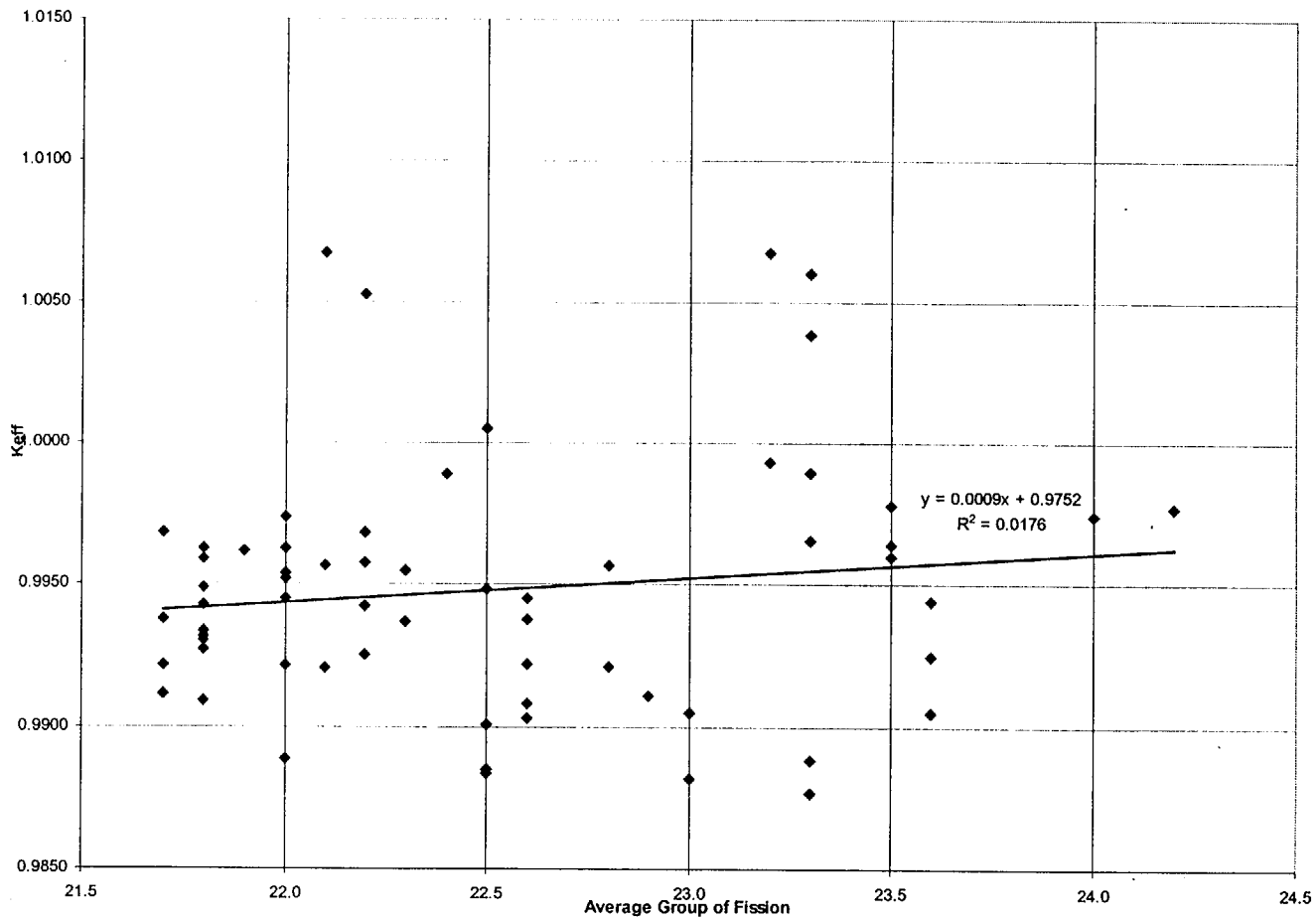

Figure 6.5.1-6 KENO-Va Validation—27-Group Library Results: k_{eff} versus B^{10} Loading for Flux Trap Criticals

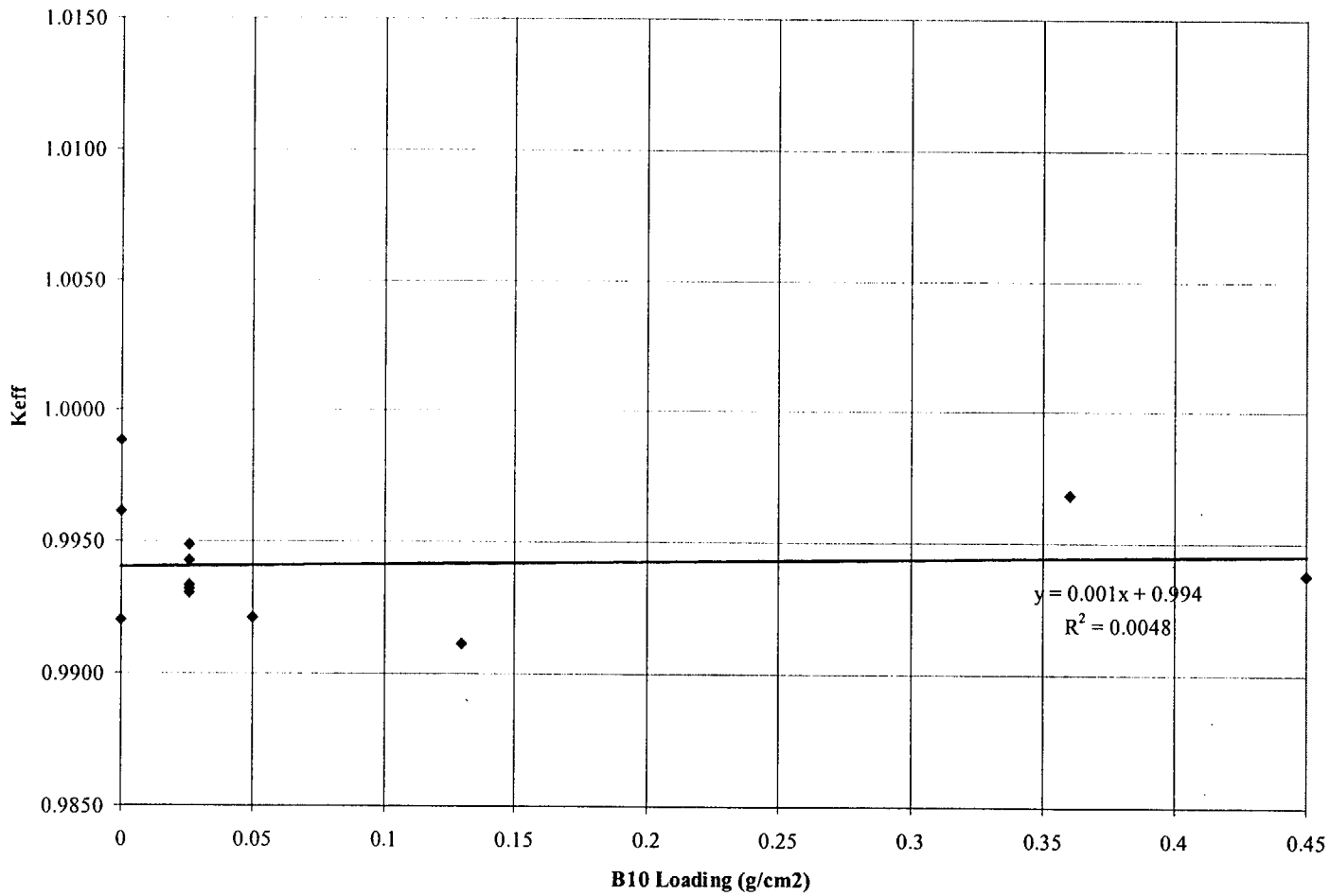

Figure 6.5.1-7 KENO-Va Validation—27-Group Library Results: k_{eff} versus Flux Trap Critical Gap Thickness

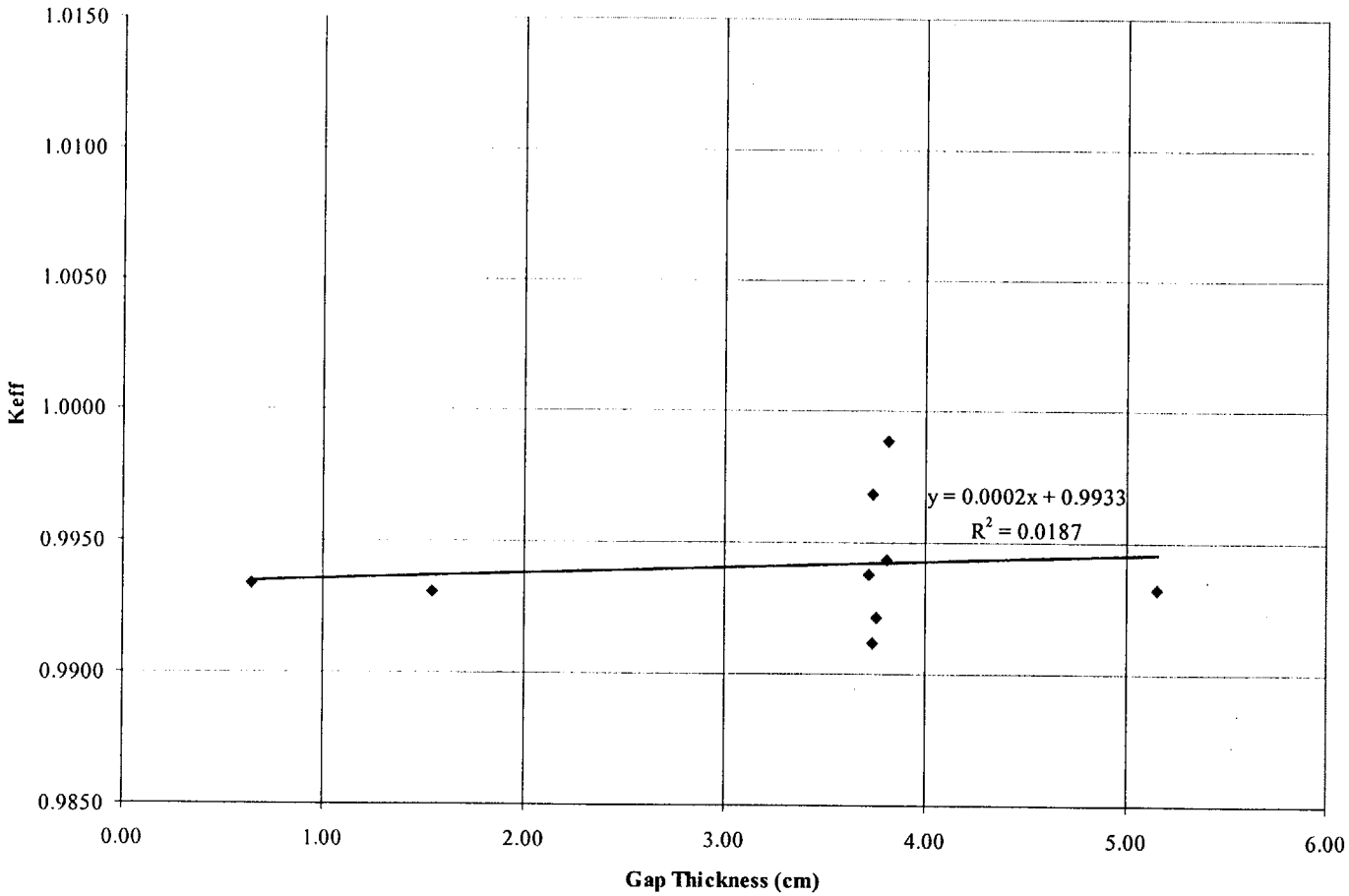

Figure 6.5.1-8 USLSTATS Output for Fuel Enrichment Study

```
uslstats: a utility to calculate upper subcritical
limits for criticality safety applications

*****************************************************************************
Version 1.3.4, February 12, 1998
Oak Ridge National Laboratory
*****************************************************************************

Input to statistical treatment from file:EN_KEFF.TXT

Title: 63 LWR CRITICAL EXPERIMENT KEFF VS ENRICHMENT

Proportion of the population = .995
Confidence of fit            = .950
Confidence on proportion     = .950
Number of observations       =  63
Minimum value of closed band =   0.00
Maximum value of closed band =   0.00
Administrative margin        =   0.05
```

independent variable - x	dependent variable - y	deviation in y	independent variable - x	dependent variable - y	deviation in y
2.35000E+00	9.96400E-01	1.00000E-03	4.31000E+00	9.96500E-01	1.10000E-03
2.35000E+00	9.94400E-01	1.00000E-03	4.31000E+00	1.00680E+00	2.10000E-03
2.35000E+00	9.90500E-01	1.00000E-03	4.31000E+00	1.00380E+00	1.20000E-03
2.35000E+00	9.96000E-01	1.10000E-03	4.31000E+00	9.88900E-01	1.10000E-03
2.35000E+00	9.97800E-01	1.00000E-03	4.31000E+00	9.95900E-01	1.10000E-03
2.35000E+00	9.92500E-01	1.00000E-03	4.31000E+00	1.00670E+00	1.00000E-03
2.35000E+00	9.90300E-01	9.00000E-04	4.31000E+00	1.00050E+00	1.10000E-03
2.35000E+00	9.95700E-01	1.00000E-03	4.31000E+00	9.90800E-01	1.10000E-03
2.35000E+00	9.91100E-01	1.00000E-03	4.31000E+00	9.98900E-01	1.20000E-03
2.46000E+00	9.92100E-01	1.10000E-03	4.31000E+00	9.92100E-01	1.20000E-03
2.46000E+00	9.92500E-01	9.00000E-04	4.31000E+00	9.91100E-01	1.20000E-03
2.46000E+00	9.93800E-01	9.00000E-04	4.31000E+00	9.96800E-01	1.30000E-03
2.46000E+00	9.90500E-01	1.00000E-03	4.31000E+00	9.93800E-01	1.20000E-03
2.46000E+00	9.88200E-01	1.00000E-03	4.31000E+00	9.93400E-01	1.00000E-03
2.46000E+00	9.94500E-01	1.00000E-03	4.31000E+00	9.93100E-01	1.00000E-03
2.46000E+00	9.92200E-01	1.00000E-03	4.31000E+00	9.94300E-01	1.00000E-03
2.46000E+00	9.88500E-01	1.00000E-03	4.31000E+00	9.93200E-01	1.00000E-03
2.46000E+00	9.88400E-01	1.00000E-03	4.31000E+00	9.94900E-01	1.00000E-03
2.46000E+00	9.90100E-01	9.00000E-04	4.31000E+00	9.92000E-01	1.00000E-03
4.31000E+00	9.95400E-01	1.40000E-03	4.31000E+00	9.96200E-01	1.00000E-03
4.31000E+00	9.94500E-01	1.30000E-03	4.74000E+00	9.92200E-01	1.30000E-03
4.31000E+00	9.97400E-01	1.30000E-03	4.74000E+00	9.88900E-01	1.30000E-03
4.31000E+00	9.96300E-01	1.30000E-03	4.74000E+00	9.95700E-01	1.30000E-03
4.31000E+00	9.92700E-01	1.20000E-03	4.74000E+00	1.00530E+00	1.10000E-03
4.31000E+00	9.90900E-01	1.20000E-03	4.74000E+00	9.95500E-01	1.20000E-03
4.31000E+00	9.96200E-01	1.20000E-03	4.74000E+00	9.94800E-01	1.30000E-03
4.31000E+00	9.93700E-01	1.30000E-03	4.74000E+00	9.95800E-01	1.20000E-03
4.31000E+00	9.94200E-01	1.20000E-03	4.74000E+00	9.95200E-01	1.20000E-03
4.31000E+00	9.96800E-01	1.20000E-03	4.74000E+00	9.98900E-01	1.30000E-03
4.31000E+00	9.87700E-01	2.30000E-03	4.74000E+00	9.97400E-01	1.20000E-03
4.31000E+00	9.99300E-01	1.20000E-03	4.74000E+00	9.97700E-01	1.10000E-03
4.31000E+00	1.00600E+00	2.20000E-03			

```
chi =  2.1587 (upper bound = 9.49). The data tests normal.

Output from statistical treatment

63 LWR CRITICAL EXPERIMENT KEFF VS ENRICHMENT

Number of data points (n)                          63
Linear regression, k(X)                            0.9884 + ( 1.6748E-03)*X
Confidence on fit (1-gamma) [input]                95.0%
Confidence on proportion (alpha) [input]           95.0%
Proportion of population falling above
lower tolerance interval (rho) [input]             99.5%
Minimum value of X                                    2.3500
Maximum value of X                                    4.7400
Average value of X                                    3.81143
```

Figure 6.5.1-8 USLSTATS Output for Fuel Enrichment Study (Continued)

```
Average value of k                                  0.99482
Minimum value of k                                  0.98770
Variance of fit, s(k,X)^2                          1.6973E-05
Within variance, s(w)^2                            1.4306E-06
Pooled variance, s(p)^2                            1.8404E-05
Pooled std. deviation, s(p)                        4.2900E-03
C(alpha,rho)*s(p)                                  1.5488E-02
student-t @ (n-2,1-gamma)                          1.67078E+00
Confidence band width, W                           7.3606E-03
Minimum margin of subcriticality, C*s(p)-W         8.1273E-03

Upper subcritical limits: (  2.35000 <= X <=   4.74000)
***** *********** *******

USL Method 1 (Confidence Band with
Administrative Margin)            USL1 = 0.9311 + ( 1.6748E-03)*X

USL Method 2 (Single-Sided Uniform
Width Closed Interval Approach)   USL2 = 0.9729 + ( 1.6748E-03)*X


USLs Evaluated Over Range of Parameter X:
**** ********* **** ***** ** ********* **

X:   2.35   2.69   3.03   3.37   3.72   4.06   4.40   4.74
-------------------------------------------------------
USL-1:  0.9350 0.9356 0.9362 0.9367 0.9373 0.9379 0.9384 0.9390
USL-2:  0.9769 0.9775 0.9780 0.9786 0.9792 0.9797 0.9803 0.9809
-------------------------------------------------------

*****************************************************************************
Thus spake USLSTATS
Finis.
```

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics

Criticals	Configura-tion	wt % ^{235}U	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
Set 1										Gap				
B&W-I	Cylindrical	2.46	1.636	1.03	1.206	1.6	0	na	na	0	na	22.8	0.9921	0.0011
B&W-II	3X3-14X14	2.46	1.636	1.03	1.206	1.6	1037	na	na	0	na	22.2	0.9925	0.0009
B&W-III	3X3-14X14	2.46	1.636	1.03	1.206	1.6	764	na	na	1.636	0.9982	22.6	0.9938	0.0009
B&W-IX	3X3-14X14	2.46	1.636	1.03	1.206	1.6	0	na	na	6.543	0.9982	23	0.9905	0.0010
B&W-X	3X3-14X14	2.46	1.636	1.03	1.206	1.6	143	na	na	4.907	0.9982	23	0.9882	0.0010
B&W-XI	3X3-14X14	2.46	1.636	1.03	1.206	1.6	514	Steel	0	1.636	0.9982	22.6	0.9945	0.0010
B&W-XIII	3X3-14X14	2.46	1.636	1.03	1.206	1.6	15	B-Al	0.0052	1.636	0.9982	22.6	0.9922	0.0010
B&W-XIV	3X3-14X14	2.46	1.636	1.03	1.206	1.6	92	B-Al	0.0040	1.636	0.9982	22.5	0.9885	0.0010
B&W-XVII	3X3-14X14	2.46	1.636	1.03	1.206	1.6	487	B-Al	0.0008	1.636	0.9982	22.5	0.9884	0.0010
B&W-XIX	3X3-14X14	2.46	1.636	1.03	1.206	1.6	634	B-Al	0.0003	1.636	0.9982	22.5	0.9901	0.0009
												Average	0.9911	0.0023

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (continued)

Criticals Set 2	Configuration	wt % ^{235}U	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
PNL-043	17X13 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9954	0.0014
PNL-044	16X14 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9945	0.0013
PNL-045	14X16 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9974	0.0013
PNL-046	12x19 Lattice	4.31	1.892	1.415	1.265	1.6	0	na	na	na	na	22.0	0.9963	0.0013
PNL-087	4 11X14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.066	2.83	0.9982	21.8	0.9927	0.0012
PNL-079	4 11X14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.030	2.83	0.9982	21.8	0.9909	0.0012
PNL-093	4 11X14 Arrays	4.31	1.892	1.415	1.265	1.6	0	BORAL	0.026	2.83	0.9982	21.8	0.9962	0.0012
PNL-115	4 9X12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Aluminum	0	2.83	0.9982	22.3	0.9937	0.0013
PNL-064	4 9X12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Steel (.302)	0	2.83	0.9982	22.2	0.9942	0.0012
PNL-071	4 9X12 Arrays	4.31	1.892	1.415	1.265	1.6	0	Steel (.485)	0	2.83	0.9982	22.2	0.9968	0.0012
												Average	0.9948	0.0020

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (continued)

Criticals Set 3	Configura-tion	wt % ^{235}U	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap Cluster (cm)	Gap Wall/ Cluster (cm)	Ave. Group Fission	k_{eff}	σ
PNL-STA	3X1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	10.65	0.00	23.5	0.9964	0.0010
PNL-STB	3X1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.20	1.32	23.6	0.9944	0.0010
PNL-STC	3X1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	10.36	2.62	23.6	0.9905	0.0010
PNL-PBA	3X1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.84	0.00	23.5	0.9960	0.0011
PNL-PBB	3X1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.72	0.66	23.5	0.9978	0.0010
PNL_PBC	3X1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.25	2.62	23.6	0.9925	0.0010
PNL-DUA	3X1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.83	0.00	22.6	0.9903	0.0009
PNL-DUB	3X1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	14.11	1.96	22.8	0.9957	0.0010
PNL-DUC	3X1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.70	2.62	22.9	0.9911	0.0010

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (continued)

Criticals	Configura-tion	wt % ^{235}U	itch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap (cm)	Ave. Group Fission	k_{eff}	σ
Set 3 (Contd.)										Cluster	Wall/ Cluster			
PNL-H20	3X1 H2O Refl	4.31	2.54	1.265	1.415	3.9	0	na	na	8.24	inf	23.3	0.9877	0.0023
PNL-ST0	3X1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	12.89	0	23.2	0.9993	0.0012
PNL-ST1	3X1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	14.12	1.32	23.3	1.0060	0.0022
PNL-ST26	3X1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	12.44	2.62	23.3	0.9965	0.0011
PNL-PB0	3X1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	20.62	0	23.2	1.0068	0.0021
PNL-PB13	3X1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	19.04	1.32	23.3	1.0038	0.0012
PNL-PB5	3X1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	10.3	5.41	23.3	0.9889	0.0011
PNL-DU0	3X1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	15.38	0	21.8	0.9959	0.0011
PNL-DU13	3X1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	19.04	1.32	22.1	1.0067	0.0010
PNL-DU39	3X1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	18.05	3.91	22.5	1.0005	0.0011
PNL-DU54	3X1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	13.49	5.41	22.6	0.9908	0.0011
												Average	0.9964	0.0060

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (continued)

Criticals	Configura-tion	wt % U^{235}	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
Set 4														
PNL-229	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Aluminum	0	3.81	0.9982	22.4	0.9989	0.0012
PNL-230	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.05	3.75	0.9982	21.7	0.9921	0.0012
PNL-228	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.13	3.73	0.9982	21.7	0.9911	0.0012
PNL-214	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.36	3.73	0.9982	21.7	0.9968	0.0013
PNL-231	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.45	3.71	0.9982	21.7	0.9938	0.0012
PNL-127	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	0.64	0.9982	21.8	0.9934	0.0010
PNL-126	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	1.54	0.9982	21.8	0.9931	0.0010
PNL-123	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	3.80	0.9982	21.8	0.9943	0.0010
PNL-125	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	5.16	0.9982	21.8	0.9932	0.0010
PNL-124	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	INF	0.9982	21.8	0.9949	0.0010
PNL-123-S	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Steel	0	3.80	0.9982	22.1	0.9920	0.0010
PNL-124-S	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Steel	0	INF	0.9982	21.9	0.9962	0.0010
												Average	0.9941	0.0022

Table 6.5.1-1 KENO-Va and 27-Group Library Validation Statistics (continued)

Criticals	Configuration	wt % U^{235}	Pitch (cm)	Pellet OD (cm)	Clad OD (cm)	H/U	Sol. Boron (ppm)	Poison	B^{10}/cm^2 (gm)	Gap (cm)	Gap Density (gm/cm^3)	Ave. Group Fission	k_{eff}	σ
Set 5														
VCML	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0	22.0	0.9922	0.0013
VCML	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.0323	22.0	0.9889	0.0013
VCML	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.2879	22.1	0.9957	0.0013
VCML	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.5540	22.2	1.0053	0.0011
VCML	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	2.50	0.9982	22.3	0.9955	0.0012
VCML	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	5.00	0.9982	22.5	0.9948	0.0013
VCML	Square Lattice	4.74	1.26	0.79	0.94	1.8	0	na	na	na	na	22.2	0.9958	0.0012
VCML	Square Lattice	4.74	1.35	0.79	0.94	2.3	0	na	na	na	na	22.0	0.9952	0.0012
VCML	Square Lattice	4.74	1.60	0.79	0.94	3.8	0	na	na	na	na	23.3	0.9989	0.0013
VCML	Square Lattice	4.74	2.10	0.79	0.94	7.6	0	na	na	na	na	24.0	0.9974	0.0012
VCML	Square Lattice	4.74	2.52	0.79	0.94	11.5	0	na	na	na	na	24.2	0.9977	0.0011
												Average	0.9961	0.0041

Table 6.5.1-2 Correlation Coefficient for Linear Curve-Fit of Critical Benchmarks

Correlation Studied	Correlation Coefficient (R)
k_{eff} versus enrichment	0.361
k_{eff} versus rod pitch	0.328
k_{eff} versus H/U volume ratio	0.246
k_{eff} versus ^{10}B loading	0.069
k_{eff} versus average group causing fission	0.133
k_{eff} versus flux gap thickness	0.137

Table 6.5.1-3 Most Reactive Configuration System Parameters

Parameters	Value
Enrichment (wt % ^{235}U)	4.0
Rod pitch (cm)	1.26
H/U volume ratio	1.9
^{10}B loading (g/cm^2)	0.025
Average group causing fission	22.3
Flux gap thickness (cm)	2.2 to 3.75

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6.5.2 SCALE 4.4 Validation in Accordance with NUREG/CR-6361

This section implements the USLSTATS method of NUREG/CR-6361 to validate the SCALE 4.4 application of KENO Va with the 44-group ENDF-BV energy library for LWR transport and storage applications. All of the critical benchmarks utilized in the validation of the SCALE 4.3 application with 27-group ENDF-BV energy libraries are used in this validation of SCALE 4.4 application with 44-group ENDF-BV energy libraries. Benchmarks with Zircaloy clad fuel are also considered for a total of 70 LWR type criticality experiments, as listed in Table 6.5.2-1. The average calculated k_{eff} and standard deviation of all the critical experiments is 0.9987 ± 0.0041 with a range from 0.9881 to 1.0100 and with a slight negative bias of -0.13%.

The range of the parameters benchmarked, i.e., the area of applicability for this validation, is given in Table 6.5.2-2. Trending in k_{eff} was evaluated for the following independent variables: wt % ^{235}U, rod pitch, H/U volume ratio, average neutron group causing fission, ^{10}B loading for flux trap cases, and flux trap gap thickness. The trending data is plotted in Figures 6.5.2-1 through 6.5.2-6. No statistically significant trends were found, and a constant bias with associated uncertainty was determined. This is especially true for k_{eff} versus absorber sheet boron loading or flux trap gap spacing, both of which are important criticality safety features of LWR fuel storage and transport casks. However, a weak statistical correlation is exhibited for k_{eff} versus enrichment. This correlation was chosen for USLSTATS Upper Subcritical Limit (USL) determination. USLSTAT determines an upper safety limit incorporating administrative margin, methodology bias, uncertainty in the bias and experimental uncertainty. USLSTATS analysis of the trend of k_{eff} with enrichment yielded the following upper subcritical limits:

```
Upper subcritical limits: (  2.3500      <= X (wt% 235U) <=    5.7400     )
***** *********** *******

USL Method 1 (Confidence Band with
Administrative Margin)              USL1 = 0.9349 + ( 1.5346E-03)*X (X <     4.6988)
                                         = 0.9421                   (X >=    4.699 )

USL Method 2 (Single-Sided Uniform
Width Closed Interval Approach)     USL2 = 0.9761 + ( 1.5346E-03)*X (X <     4.6988)
                                         = 0.9834                   (X >=    4.699 )
```

A USL of 0.94134 bounds the calculated USL for the most reactive PWR fuel to be loaded into the UMS®, fuel enriched to 4.2 or 4.203 wt % ^{235}U. A USL of 0.94104 bounds the calculated USL for the most reactive BWR fuel to be loaded into the UMS®, fuel enriched to 4.0 wt % ^{235}U. The parameters of the design basis PWR fuel are also included in Table 6.5.2-2 to show that the most reactive configuration falls within the range of applicability of the validation.

Figure 6.5.2-1 KENO-Va Validation - 44-Group Library Results: k_{eff} versus Enrichment

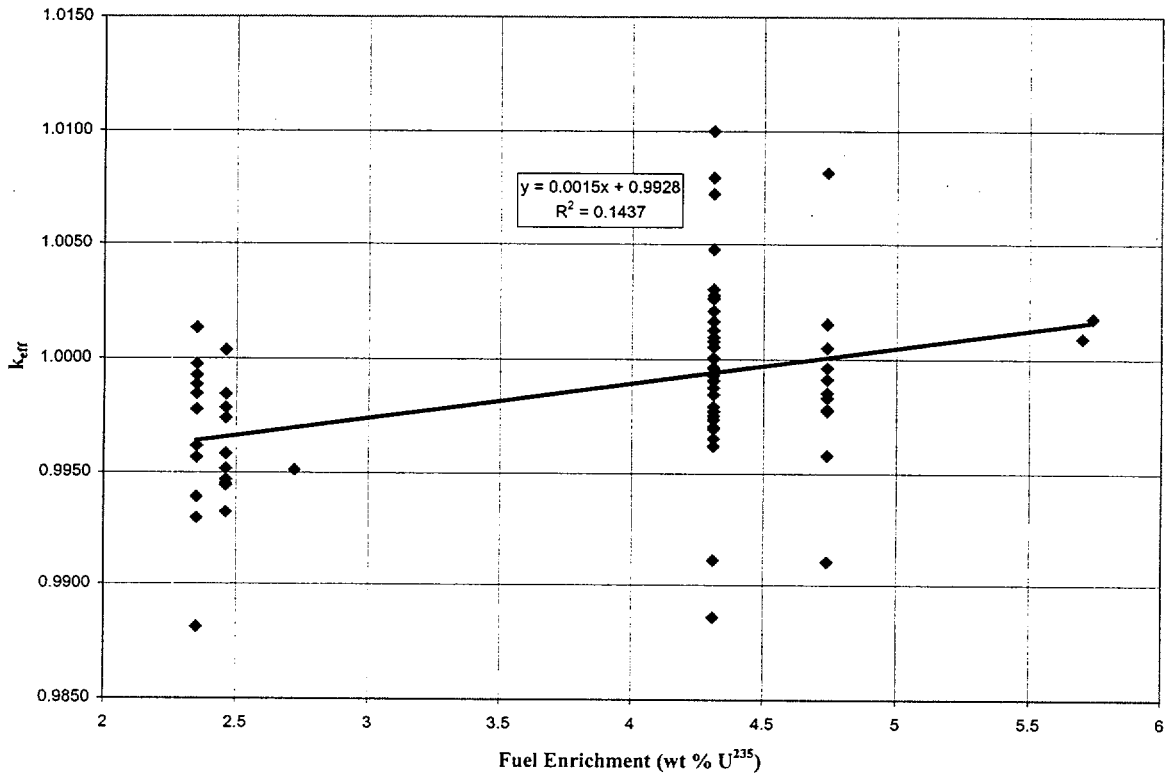

Figure 6.5.2-2 KENO-Va Validation - 44-Group Library Results: k_{eff} versus Rod Pitch

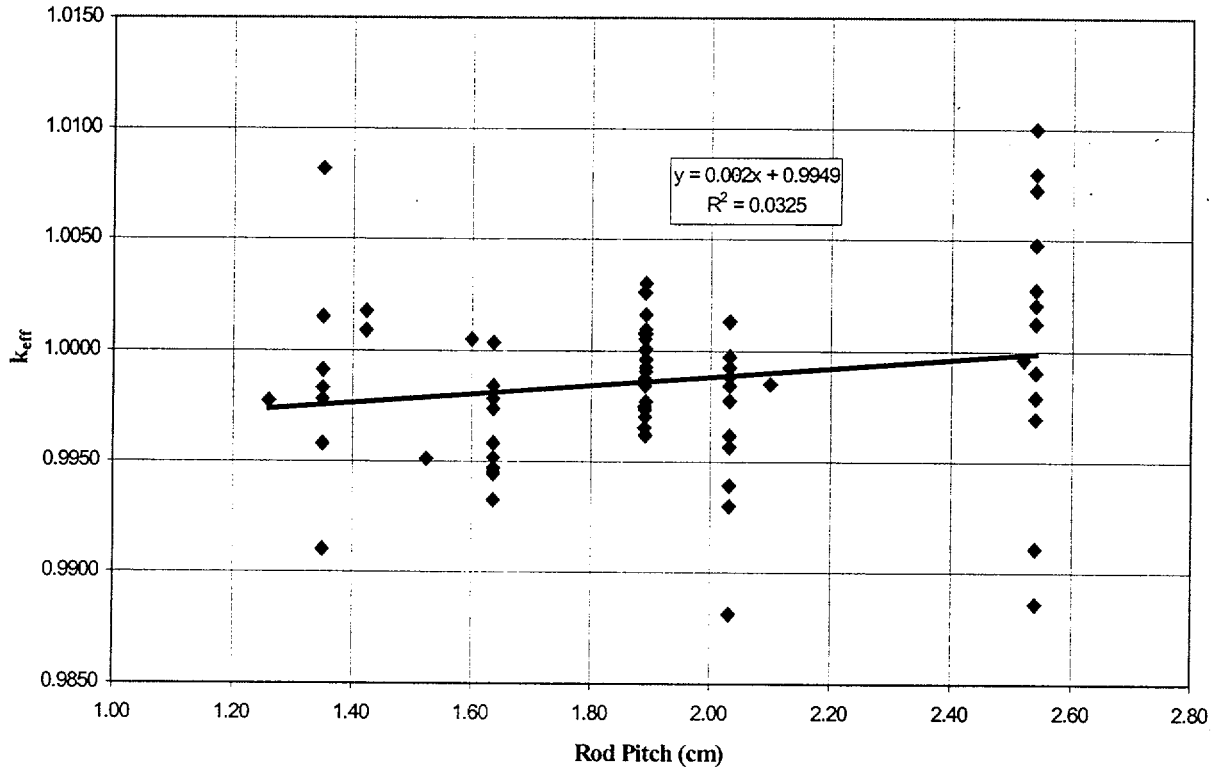

Figure 6.5.2-3 KENO-Va Validation - 44-Group Library Results: k_{eff} versus H/U Volume Ratio

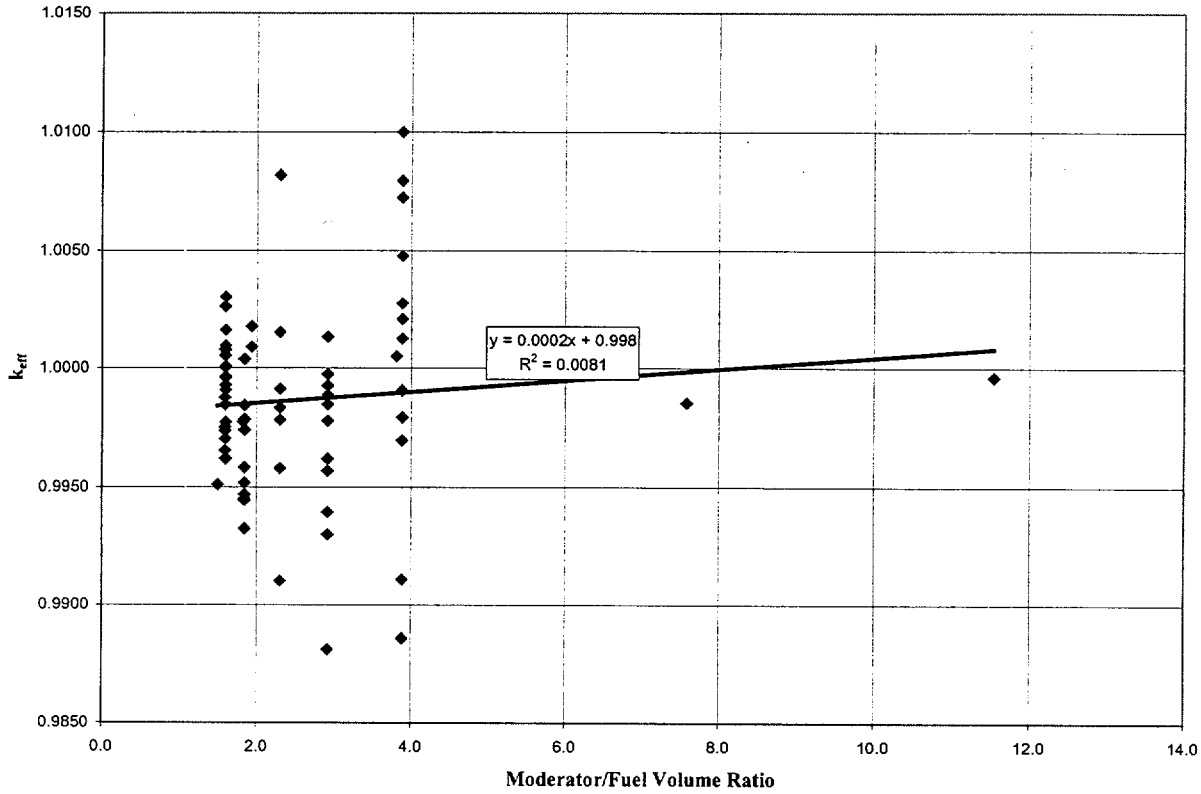

Figure 6.5.2-4 KENO-Va Validation - 44-Group Library Results: k_{eff} versus ^{10}B Loading for Flux Trap Criticals

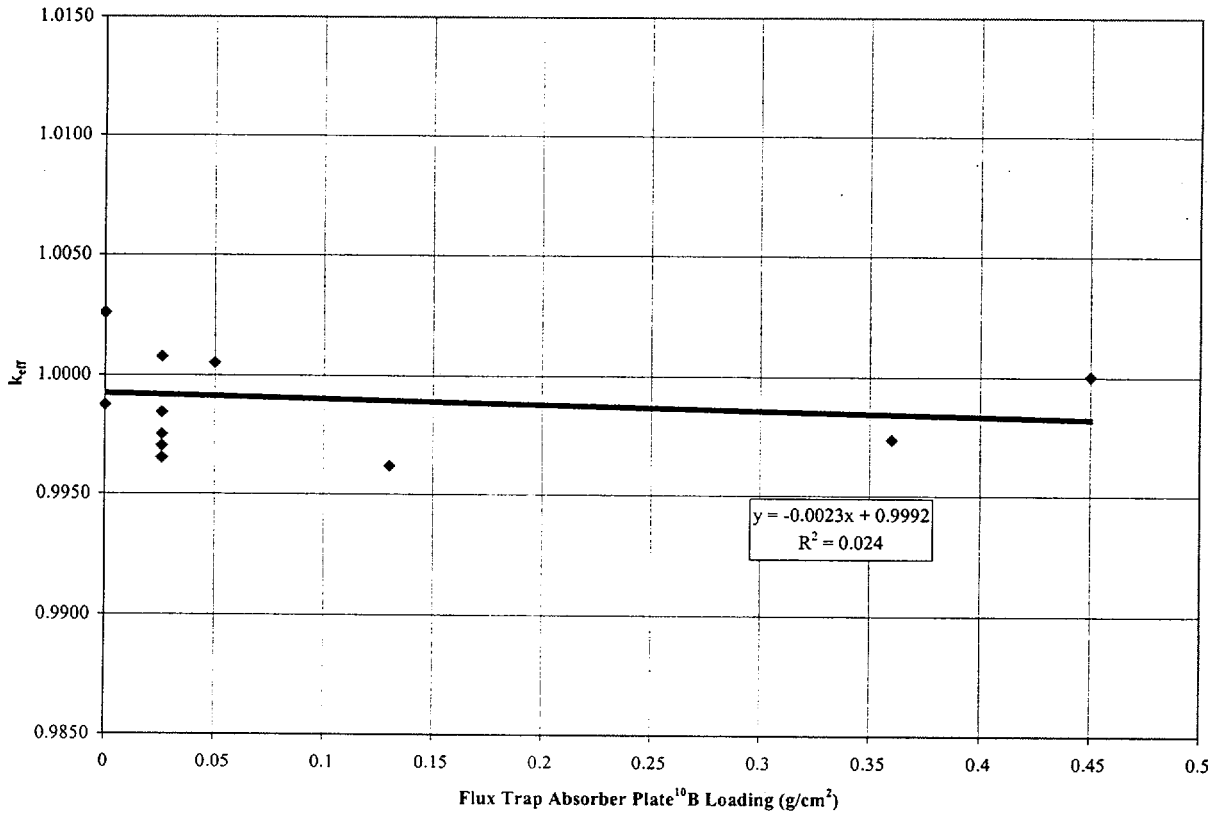

Figure 6.5.2-5 KENO-Va Validation - 44-Group Library Results: k_{eff} versus Average Group of Fission

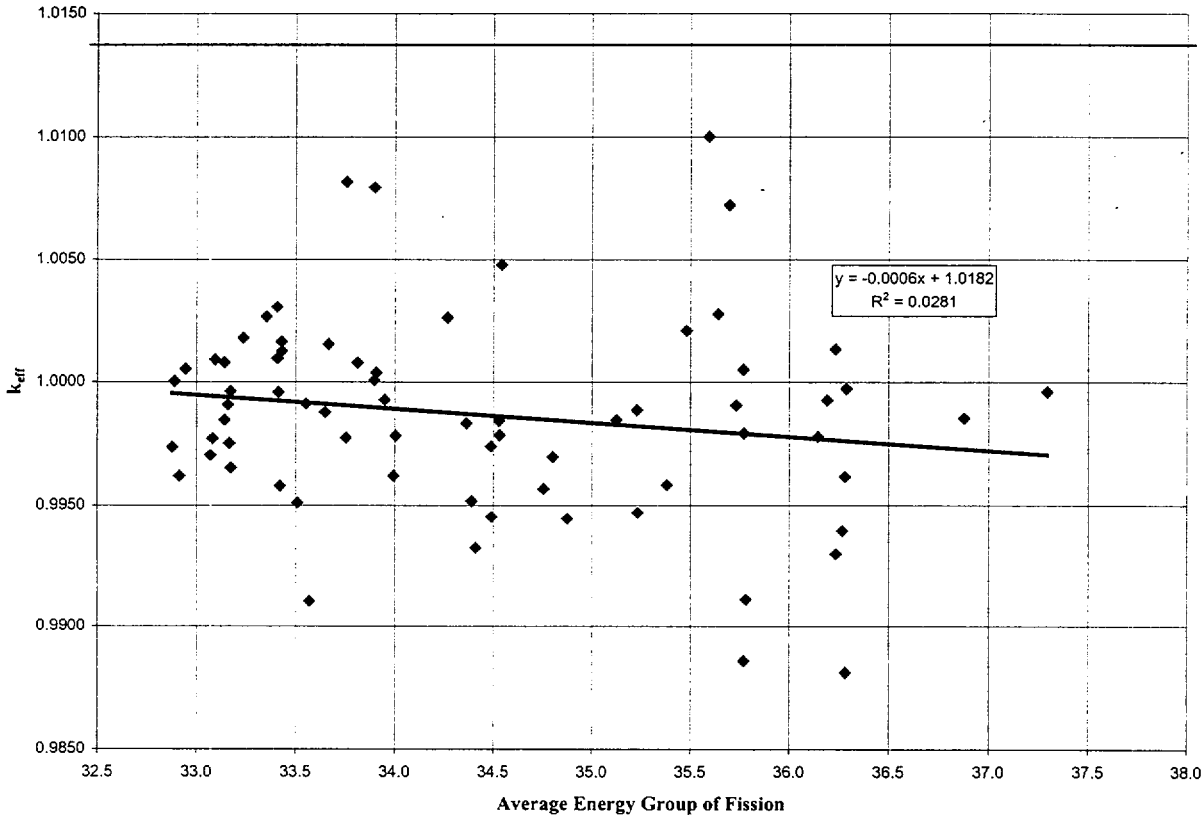

Figure 6.5.2-6 KENO-Va Validation - 44-Group Library Results: k_{eff} versus Cluster Gap Thickness

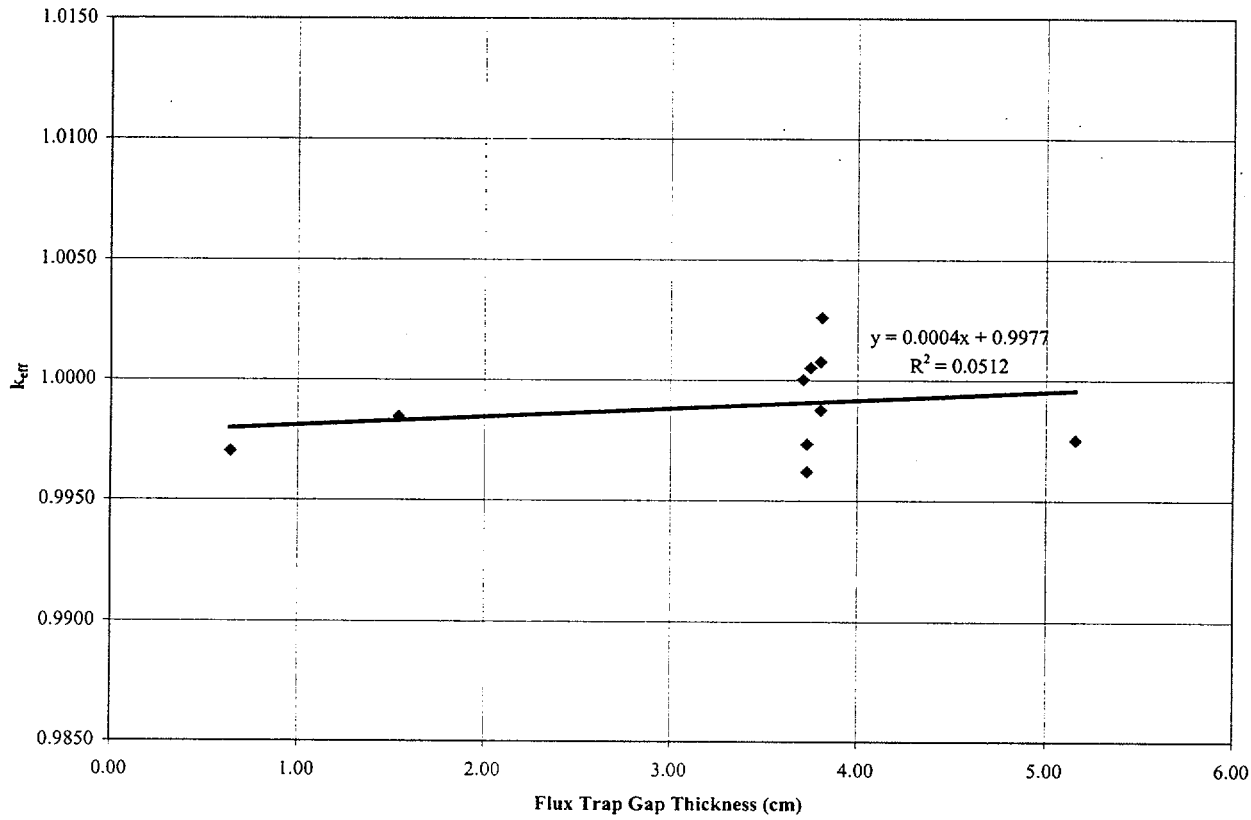

Table 6.5.2-1 KENO-Va and 44-Group Library Validation Statistics

Case No.	Configuration	wt % ^{235}U	Pitch (cm)	Pel. OD (cm)	Clad OD (cm)	M/F	Sol. B (ppm)	Absorber Plate	Plt. Loading (g B^{10}/cm^2)	Sep. Gap (cm)	Gap Den. (g/cm^3)	AEG	K_{eff}	σ
1	Cylindrical	2.46	1.636	1.03	1.206	1.8	0	na	na	0	na	34.9	0.99443	0.00125
2	3X3-14X14	2.46	1.636	1.03	1.206	1.8	1037	na	na	0	na	33.9	1.00036	0.00097
3	3X3-14X14	2.46	1.636	1.03	1.206	1.8	764	na	na	1.636	0.9982	34.5	0.99784	0.00105
4	3X3-14X14	2.46	1.636	1.03	1.206	1.8	0	na	na	6.543	0.9982	35.4	0.99582	0.00108
5	3X3-14X14	2.46	1.636	1.03	1.206	1.8	143	na	na	4.907	0.9982	35.2	0.99469	0.00126
6	3X3-14X14	2.46	1.636	1.03	1.206	1.8	514	Steel	0	1.636	0.9982	34.5	0.99739	0.00121
7	3X3-14X14	2.46	1.636	1.03	1.206	1.8	15	B-Al	0.0052	1.636	0.9982	34.5	0.99843	0.00139
8	3X3-14X14	2.46	1.636	1.03	1.206	1.8	92	B-Al	0.0040	1.636	0.9982	34.5	0.99451	0.00124
9	3X3-14X14	2.46	1.636	1.03	1.206	1.8	487	B-Al	0.0008	1.636	0.9982	34.4	0.99324	0.00120
10	3X3-14X14	2.46	1.636	1.03	1.206	1.8	634	B-Al	0.0003	1.636	0.9982	34.4	0.99517	0.00097
											Average & Std Dev.		0.99619	0.00223
11	17X13 Lattice	4.31	1.892	1.265	1.415	1.6	0	na	na	na	na	33.4	1.00302	0.00170
12	16X14 Lattice	4.31	1.892	1.265	1.415	1.6	0	na	na	na	na	33.4	1.00094	0.00157
13	14X16 Lattice	4.31	1.892	1.265	1.415	1.6	0	na	na	na	na	33.4	0.99959	0.00135
14	12x19 Lattice	4.31	1.892	1.265	1.415	1.6	0	na	na	na	na	33.4	1.00162	0.00125
15	4 11X14 Arrays	4.31	1.892	1.265	1.415	1.6	0	BORAL	0.066	2.83	0.9982	33.1	0.99772	0.00146
16	4 11X14 Arrays	4.31	1.892	1.265	1.415	1.6	0	BORAL	0.030	2.83	0.9982	33.2	0.99908	0.00130
17	4 11X14 Arrays	4.31	1.892	1.265	1.415	1.6	0	BORAL	0.026	2.83	0.9982	33.2	0.99963	0.00151
18	4 9X12 Arrays	4.31	1.892	1.265	1.415	1.6	0	Aluminum	0	2.83	0.9982	33.9	0.99928	0.00149
19	4 9X12 Arrays	4.31	1.892	1.265	1.415	1.6	0	Steel (.302)	0	2.83	0.9982	33.9	1.00007	0.00129
20	4 9X12 Arrays	4.31	1.892	1.265	1.415	1.6	0	Steel (.485)	0	2.83	0.9982	33.8	1.00076	0.00146
											Average & Std Dev.		1.00017	0.00148

Table 6.5.2-1 KENO-Va and 44-Group Library Validation Statistics (continued)

Case No.	Configuration	wt % ^{235}U	Pitch (cm)	Pel. OD (cm)	Clad OD (cm)	M/F	Sol. B (ppm)	Absorber Plate	Plt. Loading (g B^{10}/cm^2)	Cluster Gap (cm)	Wall/Clus Gap (cm)	AEG	K_{eff}	σ
21	3X1 H2O Refl	2.35	2.032	1.1176	1.27	2.9	0	na	na	8.31	inf.	36.3	0.98812	0.00114
22	3X1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	10.65	0.00	36.1	0.99777	0.00109
23	3X1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.20	1.32	36.2	0.99298	0.00163
24	3X1 St Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	10.36	2.62	36.3	0.99392	0.00135
25	3X1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.84	0.00	36.2	0.99926	0.00126
26	3X1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.72	0.66	36.2	1.00131	0.00125
27	3X1 Pb Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.25	2.62	36.3	0.99617	0.00115
28	3X1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	11.83	0.00	34.8	0.99567	0.00127
29	3X1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	14.11	1.96	35.1	0.99846	0.00131
30	3X1 DU Refl.	2.35	2.032	1.1176	1.27	2.9	0	na	na	13.70	2.62	35.2	0.99886	0.00112
31	3X1 H2O Refl	4.31	2.54	1.265	1.415	3.9	0	na	na	8.23	inf.	35.8	0.98859	0.00134
32	3X1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	12.89	0	35.5	1.00207	0.00138
33	3X1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	14.12	1.32	35.6	1.00274	0.00148
34	3X1 St Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	12.44	2.62	35.7	0.99906	0.00142
35	3X1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	20.62	0	35.6	1.01000	0.00135
36	3X1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	19.04	1.32	35.7	1.00723	0.00144
37	3X1 Pb Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	10.3	5.41	35.8	0.99109	0.00124
38	3X1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	15.38	0	33.4	1.00124	0.00135
39	3X1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	19.04	1.32	33.9	1.00794	0.00144
40	3X1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	18.05	3.91	34.5	1.00477	0.00146
41	3X1 DU Refl.	4.31	2.54	1.265	1.415	3.9	0	na	na	13.49	5.41	34.8	0.99696	0.00108
											Average & Std Dev.		0.99877	0.00558

Table 6.5.2-1 KENO-Va and 44-Group Library Validation Statistics (continued)

Case No.	Configuration	wt % ^{235}U	Pitch (cm)	Pel. OD (cm)	Clad OD (cm)	M/F	Sol. B (ppm)	Absorber Plate	Plt. Loading (g B^{10}/cm^2)	Sep. Gap (cm)	Gap Den. (g/cm^3)	AEG	K_{eff}	σ
42	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Aluminum	0	3.81	0.9982	34.3	1.00260	0.00118
43	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.05	3.75	0.9982	32.9	1.00052	0.00127
44	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.13	3.73	0.9982	32.9	0.99620	0.00137
45	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.36	3.73	0.9982	32.9	0.99736	0.00156
46	2x2 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.45	3.71	0.9982	32.9	1.00002	0.00135
47	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	0.64	0.9982	33.1	0.99703	0.00124
48	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	1.54	0.9982	33.1	0.99846	0.00150
49	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	3.80	0.9982	33.1	1.00077	0.00150
50	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	5.16	0.9982	33.2	0.99752	0.00124
51	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	BORAL	0.026	INF	0.9982	33.2	0.99653	0.00138
52	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Steel	0	3.80	0.9982	33.6	0.99876	0.00133
53	2x1 Flux Trap	4.31	1.89	1.265	1.415	1.6	0	Steel	0	INF	0.9982	33.4	1.00263	0.00143
											Average & Std Dev.		0.99903	0.00225
54	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0	33.6	0.99912	0.00138
55	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.0323	33.6	0.99102	0.00168
56	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.2879	33.7	1.00151	0.00137
57	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.5540	33.8	1.00817	0.00147
58	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	1.90	0.9982	34.0	0.99781	0.00147
59	2x2 Water Gap	4.74	1.35	0.79	0.94	2.3	0	na	na	5.00	0.9982	34.4	0.99832	0.00141
60	Square Lattice	4.74	1.26	0.79	0.94	1.8	0	na	na	na	na	33.8	0.99774	0.00152
61	Square Lattice	4.74	1.35	0.79	0.94	2.3	0	na	na	na	na	33.4	0.99579	0.00149
62	Square Lattice	4.74	1.60	0.79	0.94	3.8	0	na	na	na	na	35.8	1.00049	0.00161
63	Square Lattice	4.74	2.10	0.79	0.94	7.6	0	na	na	na	na	36.9	0.99853	0.00132
64	Square Lattice	4.74	2.52	0.79	0.94	11.5	0	na	na	na	na	37.3	0.99961	0.00118
											Average & Std Dev.		0.99892	0.00413

Table 6.5.2-1 KENO-Va and 44-Group Library Validation Statistics (continued)

Case No.	Configuration	wt % ^{235}U	Pitch (cm)	Pel. OD (cm)	Clad OD (cm)	M/F	Sol. B (ppm)	Absorber Plate	Plt. Loading (g B^{10}/cm^2)	Cluster Gap (cm)	Gap Den. (g/cm^3)	AEG	K_{eff}	σ
65	3-20x16	2.35	2.03	1.118	1.270	2.9	0	Zircaloy	na	8.79	0.9982	36.3	0.99974	0.00104
66	3-15x8	4.31	2.54	1.265	1.415	3.9	0	Zircaloy	na	10.92	0.9982	35.8	0.99792	0.00151
67	2-9x12	4.31	1.89	1.265	1.415	1.6	0	Zircaloy	na	11.04	0.9982	34.0	0.99620	0.00134
68	31x31 Zr Lattice	2.72	1.52	1.016	1.189	1.5	0	na	na	na	na	33.5	0.99511	0.00158
69	27x17 St Lattice	5.70	1.42	0.907	0.993	1.9	0	na	na	na	na	33.1	1.00089	0.00131
70	19x19 St Lattice	5.74	1.42	0.907	0.993	1.9	0	na	na	na	na	33.2	1.00176	0.00126
											Average & Std Dev.		0.99860	0.00264
												Overall Statistics		
												Average	0.9987	0.0014
												Std Dev.	0.0041	
												Bias	-0.0013	
											Min.	0.9881	Max.	1.0100

Table 6.5.2-2 SCALE 4.4 Range of Correlated Parameters

Parameter	**Benchmark Minimum Value**	**Benchmark Maximum Value**	**UMS® Design Basis WE 17 x 17 OFA**
Enrichment (wt % ^{235}U)	2.35	5.74	4.2
Rod pitch (cm)	1.26	2.54	1.26
Moderator/Fuel Volume Ratio	1.5	11.5	1.9
^{10}B loading (g/cm^2)	0.000	0.45	0.25
AEG causing fission	32.9	37.3	33.9
Cluster gap thickness (cm)	0.64	5.16	2.2-3.8

6.6 Criticality Evaluation for Site Specific Spent Fuel

This section presents the criticality evaluation for fuel assembly types or configurations, which are unique to specific reactor sites. Site specific spent fuel configurations result from conditions that occurred during reactor operations, participation in research and development programs, testing programs intended to improve reactor operations and from decommissioning activities. Site specific fuel includes fuel assemblies that are uniquely designed to accommodate reactor physics, such as axial fuel blanket and variable enrichment assemblies.

Site specific fuel assembly configurations are either shown to be bounded by the analysis of the standard design basis fuel assembly configuration of the same type (PWR or BWR), or are shown to be acceptable by specific evaluation of the configuration.

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6.6.1 Criticality Evaluation for Maine Yankee Site Specific Spent Fuel

In Section 6.4, loading the storage cask with the standard CE 14 x 14 fuel assembly is shown to be less reactive than loading the cask with the most reactive Westinghouse 17 x 17 OFA design basis spent fuel. This analysis addresses variations in fuel assembly dimensions, variable enrichment axial zoning patterns, annular axial fuel blankets, removed fuel rods or empty rod positions, fuel rods placed in guide tubes, fuel assemblies with a startup source or Boronometer source or other components in a guide tube, consolidated fuel assemblies, and damaged fuel and fuel debris. These configurations are not included in the standard fuel analysis, but are present in the site fuel inventory that must be stored.

6.6.1.1 Maine Yankee Fuel Criticality Model

The criticality evaluations of the Maine Yankee fuel inventory require the basket cell and basket in cask models described in Section 6.3 and 6.4. The basket cell model is principally employed in the most reactive dimension evaluation for the Maine Yankee intact fuel types. The basket cell model represents an infinite array of fuel tubes separated by one-inch flux traps and neglects the radial neutron leakage of the basket. This will result in k_{eff} values greater than 0.95. The basket cell model is, therefore, only used to determine relative reactivities of the various physical dimensions of the Maine Yankee fuel inventory, not to establish maximum k_s values for the basket loaded with Maine Yankee fuel assemblies. The basket in cask model is used for the evaluation of the remaining fuel configurations. The basket criticality model uses the nominal basket configuration with full moderation under accident conditions, where accident conditions implying the loss of fuel cladding integrity and flooding of the pellet to cladding gap in all fuel rods. The analyses presented are performed using the UMS® transport cask shield geometry. Based on the evaluation presented in Section 6.4 and the licensing analysis of the transport overpack, the most reactive transportable storage canister configuration is independent of the canister outer shell geometry (i.e., different casks – transport, transfer, or storage). Since the criticality evaluation is not sensitive to the shielding geometry outside of the canister, this result is applicable to the concrete storage cask and the transfer cask. The transport cask criticality model is identical to the transfer cask and storage cask models with the exception that the radial shielding outside of the canister is comprised of a total of 4.75 inches of steel, 2.75 inches of NS-4-FR neutron shielding and 2.75 inches of lead. The $k_{eff} + 2\sigma$ of this configuration is 0.9210, which is slightly lower than the wet gap $k_{eff} + 2\sigma$ values of 0.9238 and 0.9234 reported in Tables 6.4-6 and 6.4-7 for the transfer cask and storage cask, respectively.

6.6.1.2 Maine Yankee Intact Spent Fuel

The evaluation of the intact Maine Yankee spent fuel inventory demonstrates that, under all conditions, the maximum reactivity of the UMS® basket loaded with Maine Yankee fuel assemblies is bounded by the Westinghouse 17 x 17 OFA evaluation presented in Section 6.4. The intact fuel assembly evaluation includes the determination of maximum reactivity dimensions of the Maine Yankee fuel assemblies, and the reactivity effects of variably enriched assemblies, annular axial end blankets, removed rods, fuel in guide tubes, and consolidated fuel assemblies. Where necessary, loading restrictions are applied to limit the number and location of the basket payload evaluated.

6.6.1.2.1 Fuel Assembly Lattice Dimensional Variations

Maine Yankee 14 x 14 PWR fuel has been provided by Combustion Engineering, Exxon/ANF, and Westinghouse. The range of fuel assembly dimensions evaluated for Maine Yankee is shown in Table 6.6.1-1. Bounding fuel assembly dimensions are determined using the guidelines presented in Section 6.4.4 and are reported in Table 6.6.1-2. The dimensional perturbations that can increase the reactivity of an undermoderated array of fuel assemblies in a flooded system (including flooding the fuel-cladding gap) are:

- Decreasing the cladding outside diameter (OD)
- Increasing the cladding inside diameter (ID) (i.e., increasing the gap)
- Decreasing the pellet diameter
- Decreasing the guide tube thickness

To conservatively model the cladding thickness of the Maine Yankee standard fuel, the outside diameter of the cladding is decreased until the cladding thickness reaches the minimum. The pellet diameter is studied separately to determine which diameter maximizes the reactivity of the assembly. This study is performed using an infinite array of hybrid 14 x 14 fuel assemblies. These hybrid assemblies have the combination of the most reactive dimensions listed in Table 6.6.1-2 and are used in the evaluation of site specific fuel configurations as described in the following sections. The pellet diameter is modeled first at the maximum diameter; then it is iteratively decreased until a peak reactivity (H/U ratio) is reached. The results of this study are reported in Table 6.6.1-3. The maximum reactivity occurs at a pellet diameter of 0.3527 inches. This pellet diameter is conservatively used in the analyses of an assembly with 176 fuel rods.

The reactivity of an infinite array of basket unit cells containing infinitely tall, hybrid 14 x 14 fuel assemblies and a flooded fuel-cladding gap is $k_{eff} + 2\sigma = 0.96268$. This is less reactive than the same array of Westinghouse 17 x 17 OFA assemblies ($k_{eff} + 2\sigma = 0.9751$ from Table 6.4-1). Therefore, the design basis Westinghouse 17 x 17 OFA fuel criticality evaluation is bounding. The conservatism obtained by decreasing the pellet diameter below that of the reported Maine Yankee fuel pellet diameter is equivalent to a Δk_{eff} of 0.00247.

The most reactive lattice dimensions determined by the basket cell model are incorporated into the basket in cask model. Evaluating 24 hybrid 14 x 14 fuel assemblies with the most reactive pellet diameter for the accident condition produces a $k_{eff} + 2\sigma$ of 0.91014. This is less reactive than the accident condition for the transport cask loaded with the Westinghouse 17 x 17 OFA assemblies ($k_{eff} + 2\sigma$ of 0.9210). Therefore, the Westinghouse 17 x 17 OFA fuel criticality evaluation is bounding.

6.6.1.2.2 Variably Enriched Fuel Assemblies

Two batches of fuel used at Maine Yankee contain variably enriched fuel rods. Fuel rod enrichments of one batch are 4.21 wt % ^{235}U and 3.5 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.99 wt %. In the other batch, the fuel rod enrichments are 4.0 wt % and 3.4 wt % ^{235}U. The maximum planar average enrichment of this batch is 3.92 wt %. Loading 24 variably enriched fuel assemblies having both a maximum fuel rod enrichment of 4.21 wt % and a maximum planar average enrichment of 3.99 wt % results in a $k_{eff} + 2\sigma$ of 0.89940. Using a planar fuel rod enrichment of 4.2 wt % results in a $k_{eff} + 2\sigma$ of 0.91014. Therefore, all of the fuel rods are conservatively modeled as if enriched to 4.2 wt % ^{235}U for the remaining Maine Yankee analyses.

6.6.1.2.3 Assemblies with Annular Axial End Blankets

One batch of variably enriched fuel also incorporates 2.6 wt % ^{235}U axial end blankets with annular fuel pellets. The top and bottom 5% of the active fuel length of each fuel rod in this batch contains annular fuel pellets having an inner diameter of 0.183 inches.

This geometry is discretely modeled as approximately 5% annular fuel, 90% solid fuel and then 5% annular fuel, with all fuel materials enriched to 4.2 wt % ^{235}U. The diameter of all pellets is initially modeled as the most reactive pellet diameter. The accident case model, which includes flooding of the fuel cladding annulus, is used in this evaluation. Axial periodic boundary conditions are placed on the model, retaining the conservatism of the infinite fuel length. Use of

a smaller pellet diameter is not considered to be conservative when evaluating the annular fuel pellets. The smaller pellet diameter is the most reactive diameter under the assumption that it is solid and not an annulus. Flooding the axial end blanket annulus provides additional moderator to the fuel lattice. Therefore, the diameter of the annular pellets is also modeled as the maximum pellet diameter of 0.380 inch. The 0.380-inch diameter is applied to the annular pellets, while the smaller diameter is applied to the solid pellets. The results of both evaluations are reported in Table 6.6.1-4.

The most reactive annular fuel model for the annular axial end blankets results in a slightly more reactive system than the hybrid fuel accident evaluation. However, this annular condition is less reactive than the evaluation including Westinghouse 17 x 17 OFA assemblies. Therefore, the Westinghouse 17 x 17 OFA fuel criticality evaluation is bounding.

6.6.1.2.4 Assemblies with Removed Fuel Rods

Some of the Maine Yankee fuel assemblies have had fuel rods removed from the 14 x 14 lattice or have had poison rods replaced by hollow Zircaloy rods. The exact number and location of removed rods and hollow rods differs from one assembly to another. To determine a bounding reactivity for these assemblies, an analysis changing the location and the number of removed rods is performed. The removed rod analysis bounds that of the hollow rod analysis since the Zircaloy tubes displace moderator in the under moderated assembly lattice. For each case, all 24 assemblies are centered in the fuel tubes and have the same number and location of removed fuel rods. Various patterns of removed fuel rod locations are analyzed when the number of removed fuel rods is small enough to allow a different and possibly more reactive geometry. As the number of removed fuel rods increases, the number of possible highly reactive locations for these removed rods decreases. The fuel pellet diameter is modeled first at the most reactive diameter (0.3527 inches as determined in Section 6.6.1.2.1), and then at the maximum diameter of 0.380 inches.

The results of these analyses, which determine the most reactive number and geometry of removed rods for any Maine Yankee assembly, are presented in Tables 6.6.1-5 and 6.6.1-6. Table 6.6.1-5 contains the results based on a 0.3527-inch fuel pellet. All of the removed fuel rod cases using the smaller pellet diameter show cask reactivity levels lower than those of Westinghouse 17 x 17 OFA fuel. Table 6.6.1-6 contains the results of the evaluation using the maximum pellet diameter of 0.380 inch. Using the maximum pellet diameter provides for a more reactive system, since moderator is added (at the removed rod locations), to an assembly that contains more fuel. The most reactive removed fuel rod case occurs when 24 fuel rods are removed in the diamond shaped geometry shown in Figure 6.6.1-1, from the model containing the largest allowed pellet diameter.

This case represents the bounding number and geometry of removed fuel rods for the Maine Yankee fuel assemblies. It results in a more reactive system than either the Maine Yankee hybrid 14 x 14 fuel accident case or the Westinghouse 17 x 17 OFA accident case assuming unrestricted loading. However, as shown in Table 6.6.1-6, when the loading of any assembly with less than 176 fuel rods or filler rods is restricted to the four corner fuel tubes, the reactivity of the worse case drops well below that of the Westinghouse 17 x 17 OFA fuel assemblies. Therefore, loading of Maine Yankee fuel assemblies with removed fuel rods, or with hollow Zircaloy rods, is restricted to the four corner fuel tube positions of the basket. With this loading restriction, the Westinghouse 17 x 17 OFA criticality evaluation remains bounding.

6.6.1.2.5 Assemblies with Fuel Rods in the Guide Tubes

A few of the Maine Yankee intact assemblies may contain up to two intact fuel rods in some of the guide tubes (i.e., allowing for the potential storage of individual intact fuel rods in an intact fuel assembly). To evaluate loading of these assemblies into the canister, an analysis adding 1 and then 2 intact fuel rods into 1, 2, 3 and then 5 guide tubes is made. This approach considers a fuel assembly with up to 186 fuel rods. The results of the evaluation of these configurations are shown in Table 6.6.1-7. While higher in reactivity than the Maine Yankee hybrid base case, any fuel configuration with up to 2 fuel rods per guide tube is less reactive than the accident case for the Westinghouse 17 x 17 OFA fuel assemblies. Therefore, the Westinghouse 17 x 17 OFA fuel criticality evaluation is bounding.

Fuel rods may also be inserted in the guide tubes of fuel assemblies from which the fuel rods were removed (i.e., fuel rods removed from a fuel assembly and re-installed in the guide tubes of the same fuel assembly). These fuel rods may be intact or damaged. The maximum number of fuel rods in these assemblies, including fuel rods in the guide tubes remains 176. These configurations are restricted to loading in a Maine Yankee fuel can in a corner fuel position in the basket. As shown in Section 6.6.1.2.4 for the removed fuel rods, and Section 6.6.1.3 for the damaged fuel, the maximum reactivity of Maine Yankee assemblies containing 176 fuel rods in various configurations is bounded by the Westinghouse 17 x 17 OFA evaluation. These non-standard Maine Yankee assemblies are restricted to the corner fuel positions.

In addition to the fuel rods, some Maine Yankee assemblies may contain poison shim rods in guide tubes. These solid fill rods will serve as parasitic absorber and displace moderator and are, therefore, not included in the criticality model but are bounded by the evaluation performed.

6.6.1.2.6 Consolidated Fuel

The consolidated fuel is a 17 x 17 array of intact fuel rods with a pitch of 0.492 inches. Some of the locations in the array contain solid fill rods and some are empty. To determine the reactivity of the consolidated fuel lattice with empty fuel rod positions, an analysis changing the location and the number of empty positions is performed. This analysis considers 24 consolidated fuel lattices in the basket. All 24 consolidated fuel lattices are centered in the fuel tubes and have the same number and location of empty fuel rod positions.

As shown in Section 6.6.1.2.4, the removed fuel rod configuration with a 0.380-inch pellet diameter provides a more reactive system than a system using the optimum pellet diameter from Section 6.6.1.2.1. The larger pellet cases are more reactive, since moderator is added at the empty fuel rod positions to an assembly that contains more fuel. Therefore, the consolidated assembly empty rod position evaluation is performed with the 0.380-inch pellet diameter.

The results of this evaluation are shown in Table 6.6.1-8. Configurations having more than 73 empty positions result in a more reactive system than the Westinghouse 17 x 17 OFA model. The most reactive consolidated assembly case occurs with 113 empty rod positions in the geometry shown in Figure 6.6.1-2. However, when the loading of the consolidated fuel is restricted to the four corner fuel tubes, the reactivity of the system is lower than the accident condition of the basket loaded with Westinghouse 17 x 17 OFA assemblies. Therefore, loading of the consolidated fuel is restricted to the four corner fuel tube positions of the basket. With this loading restriction, the Westinghouse 17 x 17 OFA fuel criticality evaluation is bounding.

6.6.1.2.7 Conclusions

The criticality analyses for the Maine Yankee site specific fuel demonstrates that the UMS® basket loaded with these fuel assemblies results in a system that is less reactive than loading the basket with the Westinghouse 17 x 17 OFA fuel assemblies, provided that loading is restricted to the four corner fuel tube positions in the basket for:

- All 14 x 14 fuel assemblies with less than 176 fuel rods or solid filler rods
- All 14 x 14 fuel assemblies with hollow rods
- All 17 x 17 consolidated fuel lattices
- All 14 x 14 fuel assemblies with fuel rods in the guide tubes and a maximum of 176 fuel rods or solid rods and fuel rods.

The following Maine Yankee fuels are not restricted as to loading position within the basket:

- All 14x14 fuel assemblies with 176 fuel rods or solid filler rods at a maximum enrichment of 4.2 wt % ^{235}U.
- Variably enriched fuel with a maximum fuel rod enrichment of 4.21 wt % ^{235}U with a maximum planar average enrichment of 3.99 wt % ^{235}U.
- Fuel with solid stainless steel filler rods, solid Zircaloy filler rods or solid poison shim rods in any location.
- Fuel with annular axial end blankets of up to 4.2 wt % ^{235}U.
- Fuel with a maximum of 2 intact fuel rods in each guide tube for a total of 186 fuel rods.

Assemblies defined as unrestricted may be loaded into the basket in any basket location and may be mixed in the same basket. While not analyzed in detail, CEAs and ICI thimble assemblies may be loaded into any intact assemblies. These components displace a significant amount of water in the fuel lattice while adding parasitic absorber, thereby reducing system reactivity.

Since the storage cask and the transfer cask loaded with the Westinghouse 17 x 17 OFA fuel assemblies is criticality safe, it is inherent that the same cask loaded with the less reactive fuel assemblies employed at Maine Yankee, using the fuel assembly loading restrictions presented above, is also criticality safe.

6.6.1.3 Maine Yankee Damaged Spent Fuel and Fuel Debris

Damaged fuel assemblies are placed in a Maine Yankee fuel can prior to loading in the basket (See Drawings 412-501 and 412-502). The Maine Yankee fuel can has screened openings in the baseplate and the lid to permit drainage, vacuum drying, and inerting of the can. This evaluation conservatively considers 100% of the fuel rods in the fuel can as damaged.

Fuel debris can be loaded in a rod or tube structure that is subsequently loaded into a Maine Yankee fuel can. The mass of fuel debris placed in the rod or tube is restricted to the mass equivalent of a fuel rod of an intact fuel assembly.

The Maine Yankee spent fuel inventory includes fuel assemblies with fuel rods inserted in the guide tubes of the assembly. If the integrity of the cladding of the fuel rods in the guide tubes cannot be ascertained, then those fuel rods are assumed to be damaged.

6.6.1.3.1 Damaged Fuel Rods

All of the spent fuel classified as damaged, and all of the spent fuel not in its original lattice, are stored in a Maine Yankee fuel can. This fuel is analyzed using a 100% fuel rod failure assumption. The screened fuel can is designed to preclude the release of pellets and gross particulate to the canister cavity. Evaluation of the canister with four (4) Maine Yankee fuel cans containing CE 14 x 14 fuel assemblies that have up to 176 damaged fuel rods, or consolidated fuel consisting of up to 289 fuel rods, considers 100% dispersal of the fuel from these rods within the fuel can. The Maine Yankee fuel can is restricted to loading in the four corner positions of the basket.

All loose fuel in each analysis is modeled as a homogeneous mixture of fuel and water of which the volume fractions of the fuel versus the water are varied from 0-100. By varying the fuel fraction up to 100%, this evaluation addresses fuel masses significantly larger than those available in a standard or consolidated fuel assembly. First, loose fuel from damaged fuel rods within a fuel assembly is evaluated between the remaining rods of the most reactive missing rod array. The results of this analysis, provided in Table 6.6.1-9, show a slight decrease in the reactivity of the system. This results from adding fuel to the already optimized H/U ratio of the bounding missing rod array. This effectively returns the system to an undermoderated state. Second, loose fuel is considered above and below the active fuel region of this most reactive missing rod array. This analysis is performed within a finite cask model. The results of this study, provided in Table 6.6.1-10, show that any possible mixture combination of fuel and water above and below the active fuel region; and hence, above and below the BORAL sheet coverage, will not significantly increase the reactivity of the system beyond that of the missing rod array. Loose fuel is also considered to replace all contents of the Maine Yankee fuel can in each four corner fuel tube location. The results of this study, provided in Table 6.6.1-11, show that any mixture of fuel and water within this cavity will not significantly increase the reactivity of the system beyond that of the missing rod array.

Damaged fuel within the fuel can may also result from a loss of integrity of a consolidated fuel assembly. As described in Section 6.6.1.2.6, the consolidated assembly missing rod study shows that a potentially higher reactivity heterogeneous configuration does not increase the overall reactivity of the system beyond that of loading 24 Westinghouse 17 x 17 OFA assemblies when this configuration is restricted to the four corner locations. The homogeneous mixture study of loose fuel and water replacing the contents of the Maine Yankee fuel can (in each of the four corner fuel tube locations) considers more fuel than is present in the 289 fuel rod consolidated

assembly. This study shows that a homogeneous mixture at an optimal H/U ratio within the fuel can also does not affect the reactivity of the system.

The transfer and the storage casks loaded with the Westinghouse 17 x 17 OFA fuel assemblies remain subcritical. Therefore, it is inherent that a statistically equivalent, or less reactive, canister loading of 4 Maine Yankee fuel cans containing assemblies with up to 176 damaged rods, or consolidated assemblies with up to 289 rods and 20 of the most reactive Maine Yankee fuel assemblies, will remain subcritical. Consequently, assemblies with up to 176 damaged rods and consolidated assemblies with up to 289 rods are allowed contents as long as they are loaded into Maine Yankee fuel cans.

6.6.1.3.2 Fuel Debris

Prior to loading fuel debris into the screened Maine Yankee fuel can, fuel debris must be placed into a rod type structure. Placing the debris into rods confines the spent nuclear material to a known volume and allows the fuel debris to be treated identically to the damaged fuel for criticality analysis.

Based on the arguments presented in Section 6.6.1.3.1, the maximum k_s of the UMS® canister with fuel debris will be less than 0.95, including associated uncertainty and bias.

6.6.1.3.3 Fuel Debris from 24 Fuel Rods not in Maine Yankee Fuel Cans

This section presents the evaluation of the release of fuel debris from up to 24 fuel rods, not in Maine Yankee fuel cans (uncanned), into the canister cavity. The release of material from these rods may be the result of damage to the fuel rod cladding from any existing failure (i.e., greater than pinhole leaks or hairline cracks) or postulated future cladding failure (high burnup fuel cladding oxide layer). It has been conservatively assumed that this fuel debris failure occurs during storage, and there is moderator intrusion during long-term dry storage. The number of rods classified as damaged in uncanned assemblies is restricted to a maximum of 24 per canister. Any assembly containing such rods must be loaded into a peripheral basket location. For the discussion in this section, these fuel rods are referred to as damaged. The 24 damaged fuel rods are allowed in addition to the canned damaged fuel in the four corner locations of the basket. Dispersal of this fuel in the form of pellets and homogeneous mixtures of fuel/moderator is considered in the basket flux traps, the fuel assembly guide tubes, and in basket fuel tube regions above and below the active fuel/BORAL zone.

Evaluation of this condition is performed using the SCALE 4.4 CSAS25 calculation sequence and the 44-group ENDF/B-V cross-section library. As shown in Section 6.5.2, validation of this version of SCALE with this cross-section library results in a bounding upper safety limit of 0.94134 for fuel enriched to 4.2 or 4.203 wt % ^{235}U. For comparison, the nominal basket in a cask model with the design basis Westinghouse 17 x 17 OFA loading, the "Nominal" case from Table 6.4.5, is also analyzed with the SCALE 4.4 CSAS25 calculation sequence and the 44-group ENDF/B-V cross-section library. This results in a system reactivity, k_{eff} +2σ, of 0.9254 with the fuel clad gap of every rod flooded.

Since the Maine Yankee fuel inventory includes some fuel enriched to 4.203 wt % ^{235}U, the enrichment of the dispersed fuel is taken to be 4.21 wt % ^{235}U. To maximize the amount of fuel dispersed as pellets, the maximum pellet outer diameter of 0.4826 cm and an optimum pitch of 1.63 cm are used. When considering dispersal of the fuel in the form of a homogenous mixture, a maximum reactivity mixture is used. The mixture is 20% UO_2 fuel (enriched to 4.21 wt % ^{235}U) and 80% water. The diameter of all intact fuel rods is modeled as the maximum pellet outer diameter of 0.4826 cm.

The base case for this evaluation is a canister containing 4.2 wt % ^{235}U intact fuel and Maine Yankee fuel cans (with damaged fuel) in the four corner locations. For extra conservatism in evaluating dispersed fuel, the fuel can body is neglected and its contents are extended to the fuel tube. In this scenario, the optimum configuration of the contents of a fuel can is a mixture of 20% UO_2 fuel (enriched to 4.2 wt % ^{235}U) and 80% water. In order to maximize the reactivity of the assemblies, six damaged rods are removed from four peripheral assemblies as this is the most reactive removed rod geometry for the 24 uncanned damaged fuel rods. The location of the six removed rods within these four assemblies is shown in Figure 6.6.1-3. The fuel clad gap of all intact rods is modeled as void and all components are centered. This base case infinite height model yields a k_{eff} +2σ of 0.9080. This base case finite height model yields a k_{eff} +2σ of 0.9050.

Once the most reactive locations of dispersed fuel are identified, the fuel-clad gap of the intact rods in these models is flooded, and these cases are then rerun to allow an accident condition comparison to loading the cask with the Westinghouse 17 x 17 OFA fuel.

Evaluation of Dispersed Pellets

The pellets from up to 24 fuel rods are considered to disperse into any open basket location other than the active fuel region lattice, i.e., the flux traps, guide tubes, and assembly plenum/hardware regions. The pellet stacks are assumed to be at an optimum pitch of 1.63 cm, except in the guide tubes where one stack is modeled.

In the case where pellets are dispersed into the flux traps, which is a highly conservative assumption in the storage configuration, 24 rods with an active fuel length of 347.218 cm are dispersed into the 4.9784 cm gaps between the steel and aluminum disks. There are approximately 29 steel disks and 28 aluminum disks (56 gaps) over the active fuel region of the Maine Yankee assemblies. The total gap length is therefore 56 x 4.9784 cm = 278.79 cm., which is less than the active fuel length. Thus, in order to preserve the total length (mass) of fuel pellets, 347.218 cm/278.79 cm x 24 rods = 29.89 or 30 pellet stacks need to be modeled in the flux traps between disks of the axially infinite model. The basket has flux trap gaps of 1.5 in. (3.81 cm), 1.0 in. (2.54 cm) and 0.875 in (2.2225 cm). These flux trap gaps can accommodate pellet stacks with a pitch of 1.63 cm in arrangements of 30 x 1, 15 x 2 and two 15 x 1. The results of evaluating various pellets in flux trap configurations are shown in Table 6.6.1-12. The two highest reactivity configurations of the group occur when the pellets are in the flux traps between the fuel assemblies in the center of the basket, configurations B and H as shown in Figures 6.6.1-4 and 6.6.1-5, respectively. Flooding the fuel clad gap of the intact rods in either of these configurations results in a system reactivity that is below the Westinghouse 17 x 17 design basis k_{eff} of 0.9254 and the Upper Safety Limit (USL) of 0.94134.

The analysis of intact fuel rods within the guide tubes, reported in Section 6.6.1.2.5, shows that one rod (or stack of pellets) per guide tube is the most reactive heterogeneous fuel in guide tube geometry. Undamaged and damaged fuel assemblies with one additional fuel rod (unclad) dispersed into the five guide tubes is evaluated. For model simplicity, five such fuel assemblies within the basket are evaluated. This conservatively amounts to 25 failed fuel rods that are dispersed, one more rod than the 24-rod design basis. Clusters of fuel assemblies with pellets in the guide tubes are evaluated both on the interior and the exterior of the basket. The results of this analysis are reported in Table 6.6.1-12. The most reactive pellet in guide tube configuration occurs when the pellets are in the guide tubes of the fuel assemblies, as shown in Figure 6.6.1-6. However, this configuration results in a system reactivity that is less reactive than dispersal of pellets into the flux traps.

In the situation where pellets are dispersed into the upper and lower fuel assembly hardware regions, the finite UMS® cask KENO-Va geometry from Section 6.6.1.3 is used. Pellet stacks with a 1.63 cm optimum pitch are placed above and below the active fuel/BORAL zone. Given that the active length of stacked pellets is 24 rods x 347.218 cm active fuel = 8333.2 cm and given the fuel assembly plenum/upper end fitting length of approximately 22.5 cm, then 8333.2 cm/22.5 cm = 370 pellet stacks could be dispersed into the upper hardware regions. For model simplicity and conservatism, this is modeled as three 13 x 13 arrays (507 pellet stacks) at

optimum pitch in the hardware zones above the active fuel/BORAL zone. Given that the lower end fitting zone is approximately 7.90 cm, then 8333.2 cm/7.90 cm = 1055 pellet stacks could be dispersed into the lower hardware regions. This is modeled conservatively as seven 13 x 13 arrays (1183 pellet stacks) at optimum pitch in the hardware zones below the active fuel/BORAL zone. The results of evaluating various configurations of pellets in the hardware regions are shown in Table 6.6.1-14. Configurations A (shown in Figure 6.6.1-7), B and C have pellet stacks dispersed in the upper hardware zones, while configurations D, E and F (shown in Figure 6.6.1-8) have pellet stacks dispersed in the lower hardware zones. Very little difference in reactivity from the base case is observed in the situation where pellets are dispersed into the hardware zones.

Evaluation of Dispersed Homogeneous Mixture of Fuel and Moderator

The mixture of fuel from up to 24 fuel rods and moderator is also considered to disperse into any open basket location other than the active fuel region lattice, i.e., the flux traps, guide tubes, and assembly plenum/hardware regions. The homogeneous mixture of fuel and moderator is assumed to be an optimum mixture of 20% UO_2 (enriched to 4.21wt % ^{235}U) and 80% H_2O.

The maximum amount of UO_2 in 24 Maine Yankee fuel rods would normally occupy 6,097.3 cm^3, but with a volume fraction of 0.20, it occupies a volume of 30,486.5 cm^3. In the cases where the fuel material from 24 rods is dispersed as a homogeneous mixture of 20% UO_2 and 80% H_2O into the flux trap regions of the active fuel, the maximum area this mixture would occupy is (30,486.5 cm^3/278.79 cm) 109.35 cm^2. Given the basket has flux trap thicknesses of 3.81, 2.54 and 2.2225 cm, rectangular areas of the following widths are calculated:

109.35 cm^2/3.81 cm = 28.70 cm,
109.35 cm^2/2.54 cm = 43.05 cm, and
109.35 cm^2/2.2225 cm = 49.20 cm.

These rectangular regions of homogeneous UO_2 and H_2O are inserted in the model as cuboids in the center of the flux trap regions of the basket. The resulting reactivity for the various configurations of homogeneous UO_2 and H_2O within the flux trap regions is provided in Table 6.6.1-15. The two highest reactivity configurations of the group occur when the mixture is in the flux traps between the fuel assemblies in the center of the basket, configurations A and B as shown in Figure 6.6.1-9 and Figure 6.6.1-10. Flooding the fuel clad gap of the intact rods in either of these configurations results in a system reactivity that is below the Westinghouse 17 x 17 design basis k_{eff} of 0.9254 and the USL of 0.94134.

The guide tube volume in the active fuel region is 1903 cm^3. Thus, a homogeneous mixture of UO_2 and H_2O with a fuel volume fraction of 0.20 could occupy up to 16 guide tubes. The system reactivity of various homogeneous fuel/guide tube configurations is reported in Table 6.6.1-16. The most reactive configuration of the group occurs when three out of five guide tubes contain the fuel/moderator mixture, configuration C. This configuration, shown in Figure 6.6.1-11, is less reactive than dispersal of fuel/moderator in the flux traps.

In the cases where a homogeneous mixture of UO_2 and H_2O is dispersed into the upper hardware regions above the active fuel, the maximum height it would occupy within a basket tube is 30,486.5 $cm^3/(22.352\ cm)^2 \cong 61$ cm. Since there is 22.5 cm of upper hardware length, the homogeneous mixture is conservatively distributed in three tubes above the active fuel/BORAL zone. In the case of the lower hardware regions with 7.9 cm of length, the homogeneous mixture is distributed in 8 tubes below the active fuel/BORAL zone. The results of evaluating various configurations of homogeneous UO_2 and H_2O in the hardware regions are shown in Table 6.6.1-17. Configurations A, B (shown in Figure 6.6.1-12) and C model the fuel mixture dispersed in the upper (top) hardware zones and configurations D, E and F (shown in Figure 6.6.1-13) model the fuel mixture dispersed in the lower (bottom) hardware zones. The most reactive configuration, configuration B, is less reactive than dispersal of fuel/moderator in the flux traps.

Conclusions

The greatest increases in reactivity are observed when pellets or homogeneous mixtures are dispersed in the interior flux trap regions of the basket, which is a highly conservative assumption in the storage configuration. The worst configuration of fuel dispersal in the flux traps is less reactive than loading the basket with the design basis Westinghouse 17 x 17 OFA fuel. Since the transport cask loaded with the Westinghouse 17 x 17 OFA fuel assemblies is subcritical, it is inherent that this less reactive canister loading is also subcritical. Therefore, assemblies with a maximum of 24 damaged fuel rods may be directly loaded into any of the peripheral basket locations as long as the total number of uncanned damaged fuel rods in the canister does not exceed 24.

6.6.1.4 Fuel Assemblies with a Source or Other Component in Guide Tubes

The effect on reactivity from loading Maine Yankee fuel assemblies with components inserted in the center or corner guide tube positions is also evaluated. These components include startup and Boronometer sources, Control Element Assembly (CEA) fingertips, and a 24-inch segment of an ICI thimble. Startup sources must be inserted in the center guide tube. The Boronometer source, CEA fingertips and ICI segment must be inserted in a corner guide tube that is closed at the bottom end of the assembly and closed at the top using a CEA flow plug.

6.6.1.4.1 Assemblies with Startup Sources

Maine Yankee has three Pu-Be sources and two Sb-Be sources that will be installed in the center guide tubes of 14 x 14 assemblies that subsequently must be loaded in one of the four corner fuel positions of the basket. Each source is designed to fit in the center guide tube of an assembly. All five of these startup sources contain Sb-Be pellets, which are 50% beryllium (Be) by volume. The moderation potential of the Be is evaluated to ensure that this material will not increase the reactivity of the system beyond that reported for the accident condition. The antimony (Sb) content is ignored. The startup source is assumed to remain within the center guide tube for all conditions. The base case infinite height model used for comparison is the bounding Maine Yankee fuel assembly with 24 empty rod positions as reported in Table 6.6.1-11. The center guide tube of this model is filled with 50% water and 50% Be. The analysis assumes that assemblies with startup sources are loaded in all four of the basket corner fuel positions. This configuration, resulting in a system reactivity of $k_{eff} \pm \sigma$, of 0.91085 ± 0.00087, shows that loading Sb-Be sources or the used Pu-Be sources into the center guide tubes of the assemblies in the four corner locations of the basket does not significantly impact the reactivity of the system.

One of the three Pu-Be sources was never irradiated. Analysis of this source is equivalent to assuming that the spent Pu-Be sources are fresh. The unused source has 1.4 grams of plutonium in two capsules. All of this material is conservatively assumed to be in one capsule and is modeled as ^{239}Pu. The diameter of the capsule cavity is 0.270 inch and its length is 9.75 inches. This corresponds to a capsule volume of approximately 9.148 cubic centimeters. Thus, the 1.4 grams of ^{239}Pu occupies ~0.77% of the volume at a density of 19.84 g/cc. This material composition is then conservatively assumed to fill the entire center guide tube, which models considerably more ^{239}Pu than is actually present within the Pu-Be source. The remaining volume of the guide tube is analyzed at various fractions of Be, water and/or void to ensure that any combination of these materials is considered. The results of these analyses, provided in Table 6.6.1-18, show that loading a fresh Pu-Be startup source into the center guide tube of each of the four corner assemblies does not significantly impact the reactivity of the system. Both heterogeneous and homogeneous analyses are performed.

6.6.1.4.2 One Assembly with a Boronometer Source

Maine Yankee has one Boronometer source that will be inserted in one of the four corner guide tubes of a 14 x 14 assembly to be loaded in one of the four corner positions of the basket. A CEA flow plug inserted into the top nozzle will retain the Boronometer source within the guide tube, which is closed on the bottom by the end plate. The Boronometer source contains 16

grams of plutonium and 8 grams of beryllium. All of the plutonium is assumed to be ^{239}Pu. This material is contained within a tube that has an inner diameter of 0.562 inch and a height of 0.670 inch. The end caps that are placed inside each end of the tube are 0.1 inch tall. For criticality analysis purposes, the Boronometer source is conservatively modeled as being within the center guide tube of the assembly (but must be loaded in a corner guide tube).

The first analysis of the Boronometer source models the Pu-Be material heterogeneously. The ^{239}Pu is modeled as a sphere with a radius of 0.5774 cm. This sphere is centered inside a cylinder of beryllium with a height of 0.670 inch and a diameter of 0.562 inch. The cylinder is conservatively modeled in the center of the center guide tube near the middle of the active fuel region. All structural material is modeled as a void that is flooded with water, except for the end caps, which are encompassed by the beryllium cylinder. The volume fraction of beryllium is iteratively replaced with water to ensure that any combination of beryllium and/or water is considered along with the ^{239}Pu. The results of this heterogeneous analysis, presented in Table 6.6.1-19, show that loading a fuel assembly that contains a Boronometer source into any of the four corner locations of the basket will not impact the reactivity of the system. Figures 6.6.1-14 and 6.6.1-15 show the heterogeneous model geometry. The second analysis assumes the Pu-Be material is a homogeneous mixture. As previously modeled, the Pu-Be material is contained within a tube that has an inner diameter of 0.562 inch and a height of 0.670 inch. The end caps that are placed inside each end of the tube are 0.1 inch tall. The composition of the Pu-Be material is calculated assuming that the end caps are fully inserted into the tube. This implies that the tube cavity height is 0.470 inch, which results in a cavity volume of 1.098 cm^3. Thus, the 16 grams of ^{239}Pu occupies approximately 42% of the volume at a density of 19.84 g/cc. For conservatism, the (approximately) 42% ^{239}Pu and (approximately) 58% Be mixture is assumed to fill a cylinder with a height of 0.670 inch and a diameter of 0.562 inch. This implies that 22.8 grams of ^{239}Pu is modeled instead of the 16 grams of plutonium that is actually present. The cylinder is also conservatively modeled in the center of the center guide tube near the middle of the active fuel region. All structural material is modeled as a void that is flooded with water. The volume fraction of beryllium is iteratively replaced with water to ensure that any combination of beryllium and/or water is considered along with the plutonium. The results of this homogeneous analysis, presented in Table 6.6.1-20, also show that loading a fuel assembly that contains a Boronometer source into any of the four corner locations of the basket will not increase the reactivity of the system.

6.6.1.4.3 Fuel Assemblies with Inserted CEA Fingertips or ICI String Segment

Maine Yankee fuel assemblies may have CEA finger ends (fingertips) or an ICI thimble segment inserted in one of the four corner guide tubes of the same 14 x 14 assembly that holds the Boronometer source. The ICI segment is approximately 24 inches long. These components do not contain fissile or moderating material. Therefore, it is conservative to ignore these components, as they displace moderator when the basket is flooded, thereby reducing reactivity.

6.6.1.4.4 Maine Yankee Miscellaneous Component Loading Restrictions

Based on the evaluation of Maine Yankee fuel assemblies with startup sources, a Boronometer source, CEA fingertips, or an ICI thimble segment inserted in guide tubes, the following loading restrictions apply:

1) Any Maine Yankee fuel assembly having a component evaluated in this section inserted in a corner or center guide tube must be loaded in one of the four corner fuel loading positions of the UMS® basket. Basket corner positions are also peripheral positions and are marked "P/C" in Figure 2.1.3.1-1.
2) Startup sources shall be restricted to loading in the center guide tubes of fuel assemblies classified as intact and must be loaded in a Class 1 canister.
3) Only one startup source may be loaded into any intact fuel assembly.
4) The Boronometer source must be loaded in a guide tube location that is closed at the bottom end (corner guide tube) and shall not be loaded into a fuel assembly that also holds a startup source.
5) The CEA finger tips, ICI segment and Boronometer source must be loaded in a guide tube location that is closed at the bottom end (corner guide tubes) of an intact fuel assembly. The guide tube must be closed at the top end using a CEA flow plug.
6) Fuel assemblies having a CEA flow plug installed must be loaded in a Class 2 canister.
7) Up to 24 fuel rods classified as damaged, in one or more fuel assemblies that are not in Maine Yankee fuel cans, may be loaded in the periphery positions of the basket.
8) Up to four intact fuel assemblies with inserted startup sources may be loaded in any canister (using the four corner positions of the basket).

When loaded in accordance with these restrictions, the evaluated components do not significantly impact the reactivity of the system.

6.6.1.5 Maine Yankee Fuel Comparison to Criticality Benchmarks

The most reactive system configuration parameters for Maine Yankee fuel have been compared to the range of applicability of the critical benchmarks evaluated using the KENO-Va code of the SCALE 4.3 CSAS sequence. As shown below, all of the Maine Yankee fuel parameters fall within the benchmark range.

Parameter	Benchmark Minimum Value	Benchmark Maximum Value	Maine Yankee Fuel Most Reactive Configuration
Enrichment (wt. % ^{235}U)	2.35	4.74	4.2
Rod pitch (cm)	1.26	2.54	1.50
H/U volume ratio	1.6	11.5	2.6
^{10}B areal density (g/cm^2)	0.00	0.45	0.025
Average energy group causing fission	21.7	24.2	22.5
Flux gap thickness (cm)	0.64	5.16	2.22 to 3.81
Fuel diameter (cm)	0.790	1.265	0.896
Clad diameter (cm)	0.940	1.415	1.111

The H/U volume ratio for the assembly is shown. The lattice H/U volume ratio is 2.2 for the clad gap flooded scenario.

The results of the NAC-UMS® Storage System benchmark calculations are provided in Section 6.5.1.

Figure 6.6.1-1 24 Removed Fuel Rods - Diamond Shaped Geometry, Maine Yankee Site Specific Fuel

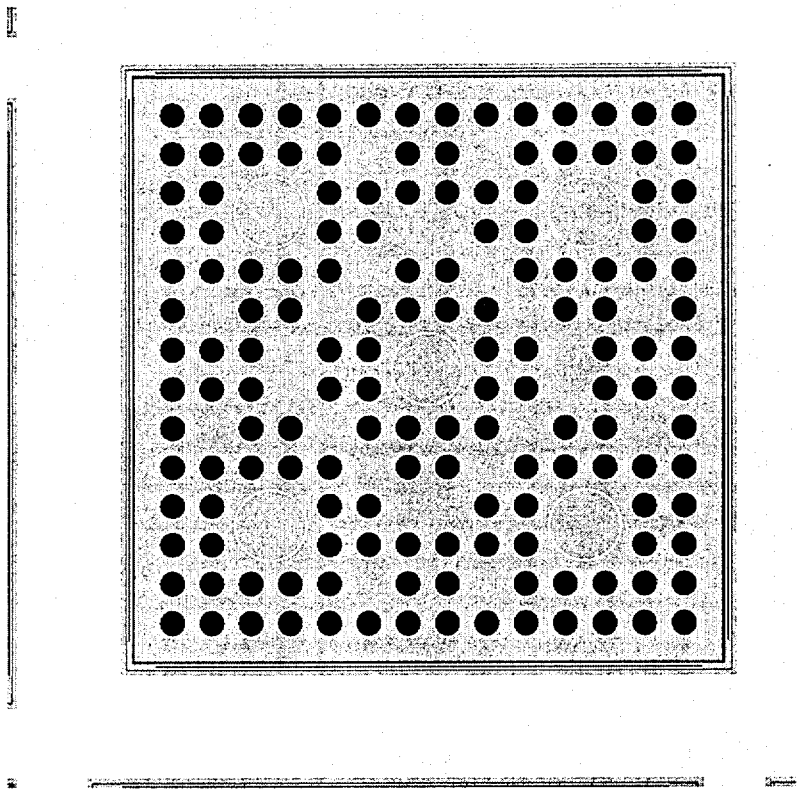

Figure 6.6.1-2 Consolidated Fuel Geometry, 113 Empty Fuel Rod Positions, Maine Yankee Site Specific Fuel

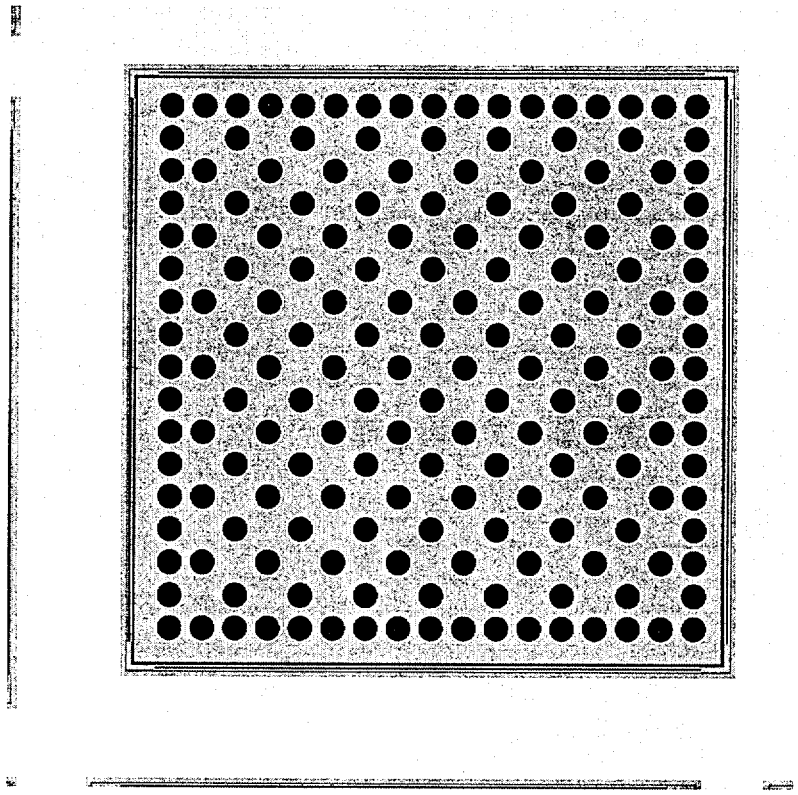

Figure 6.6.1-3 Geometry of 6 Removed Fuel Rods from 4 Periphery Assemblies

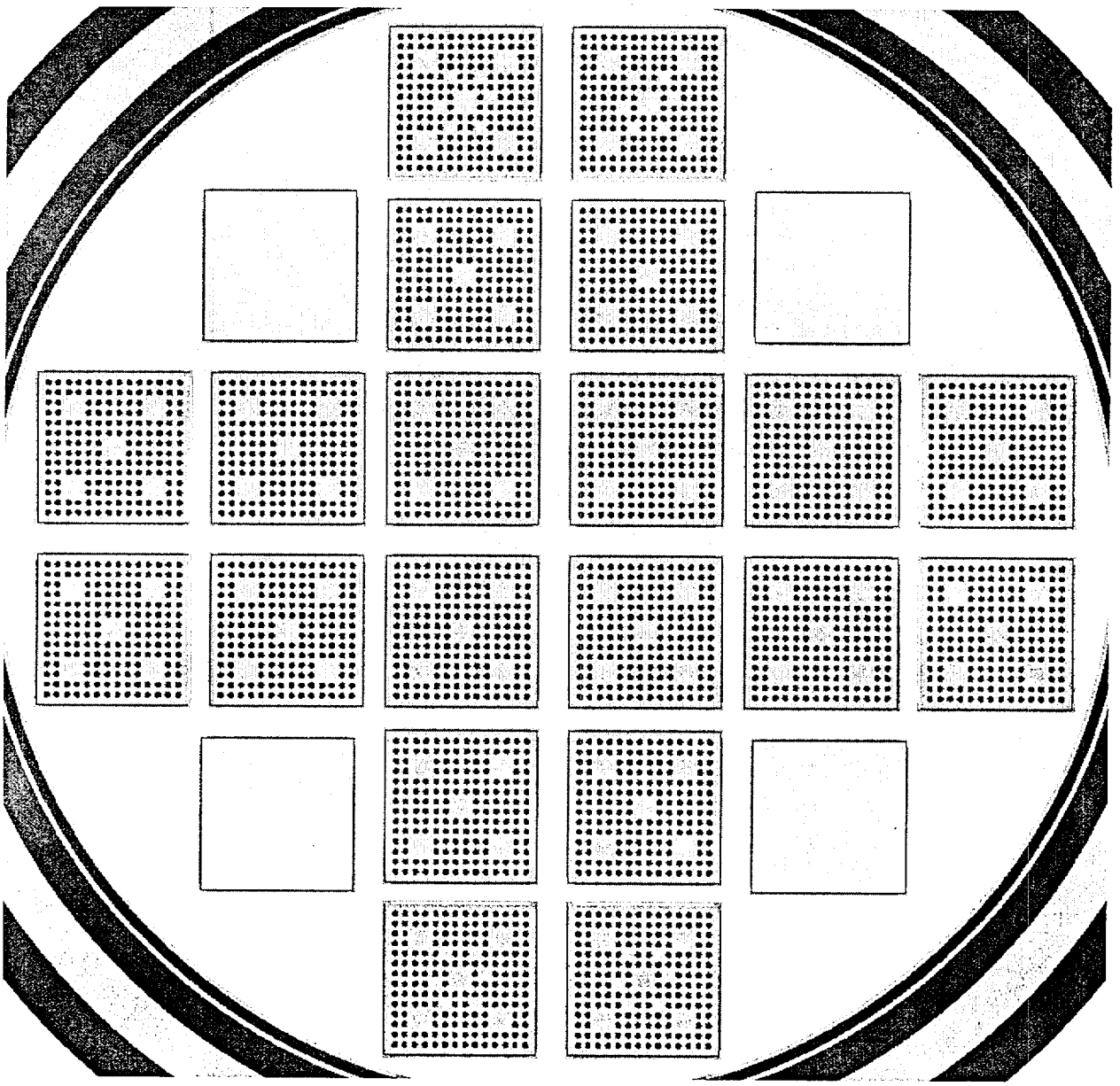

Figure 6.6.1-4 Model with Pellets in Flux Traps - Configuration B

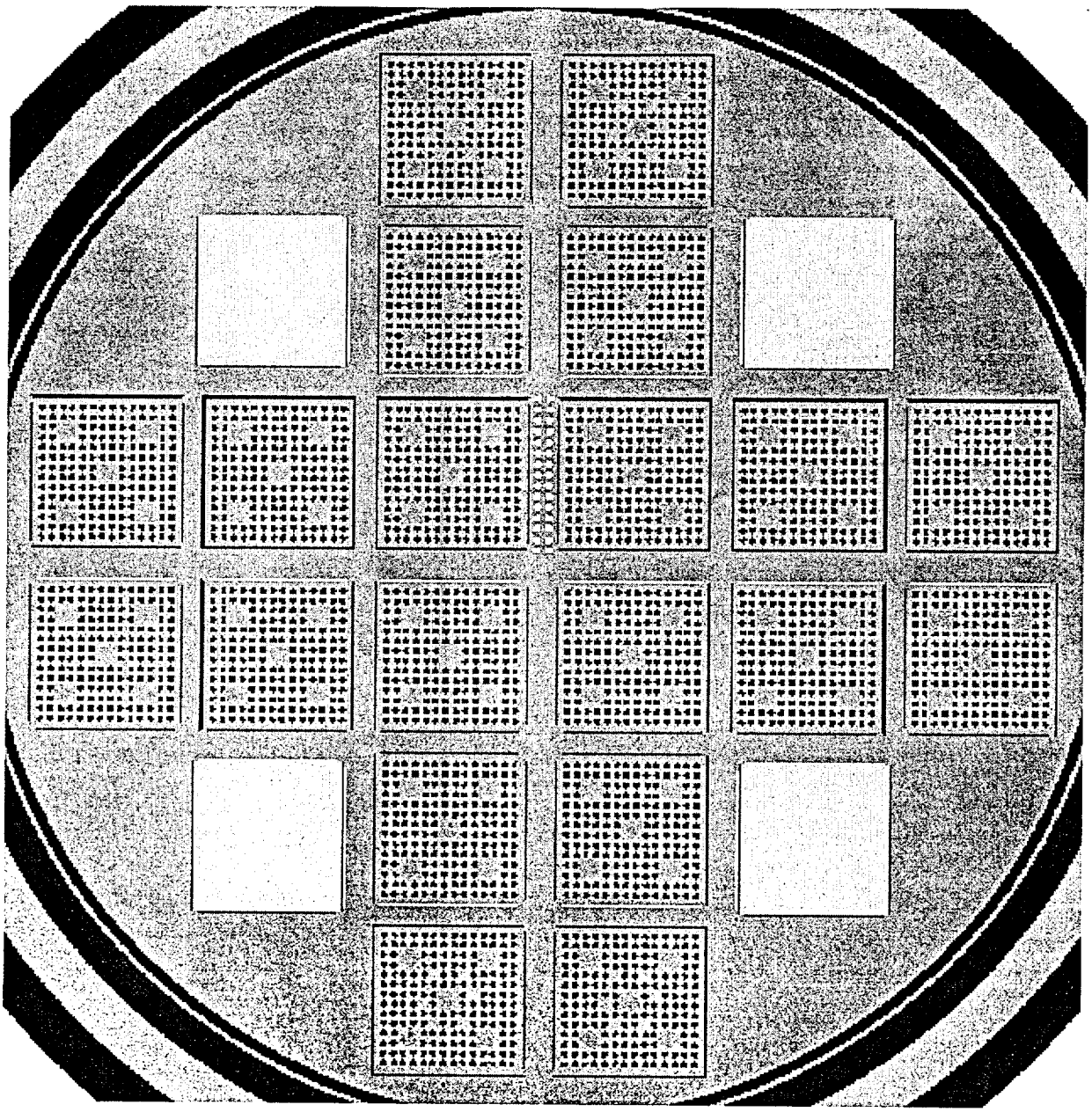

Figure 6.6.1-5 Model with Pellets in Flux Traps - Configuration H

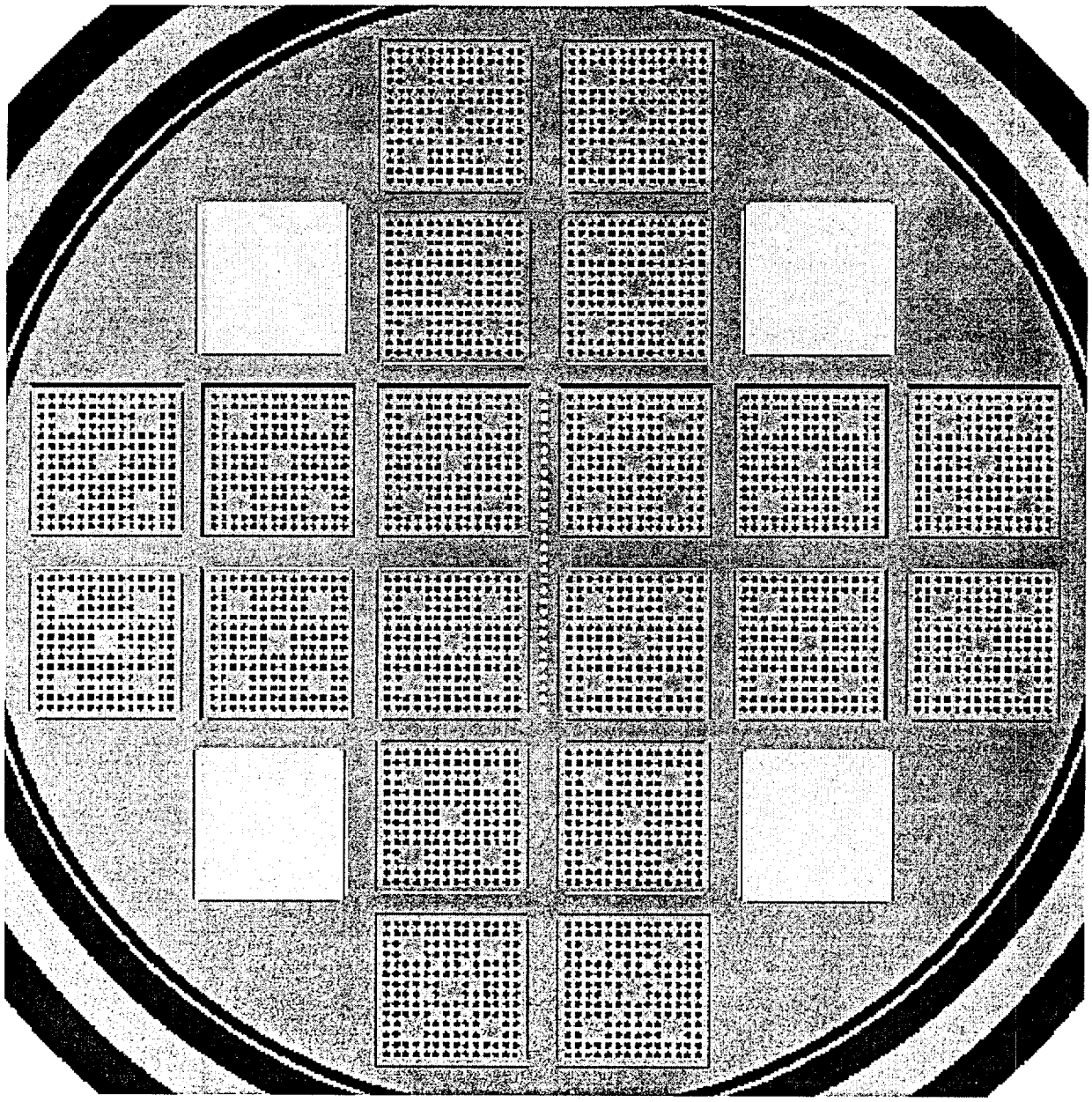

Figure 6.6.1-6 Model with Pellets in Guide Tubes – Configuration D

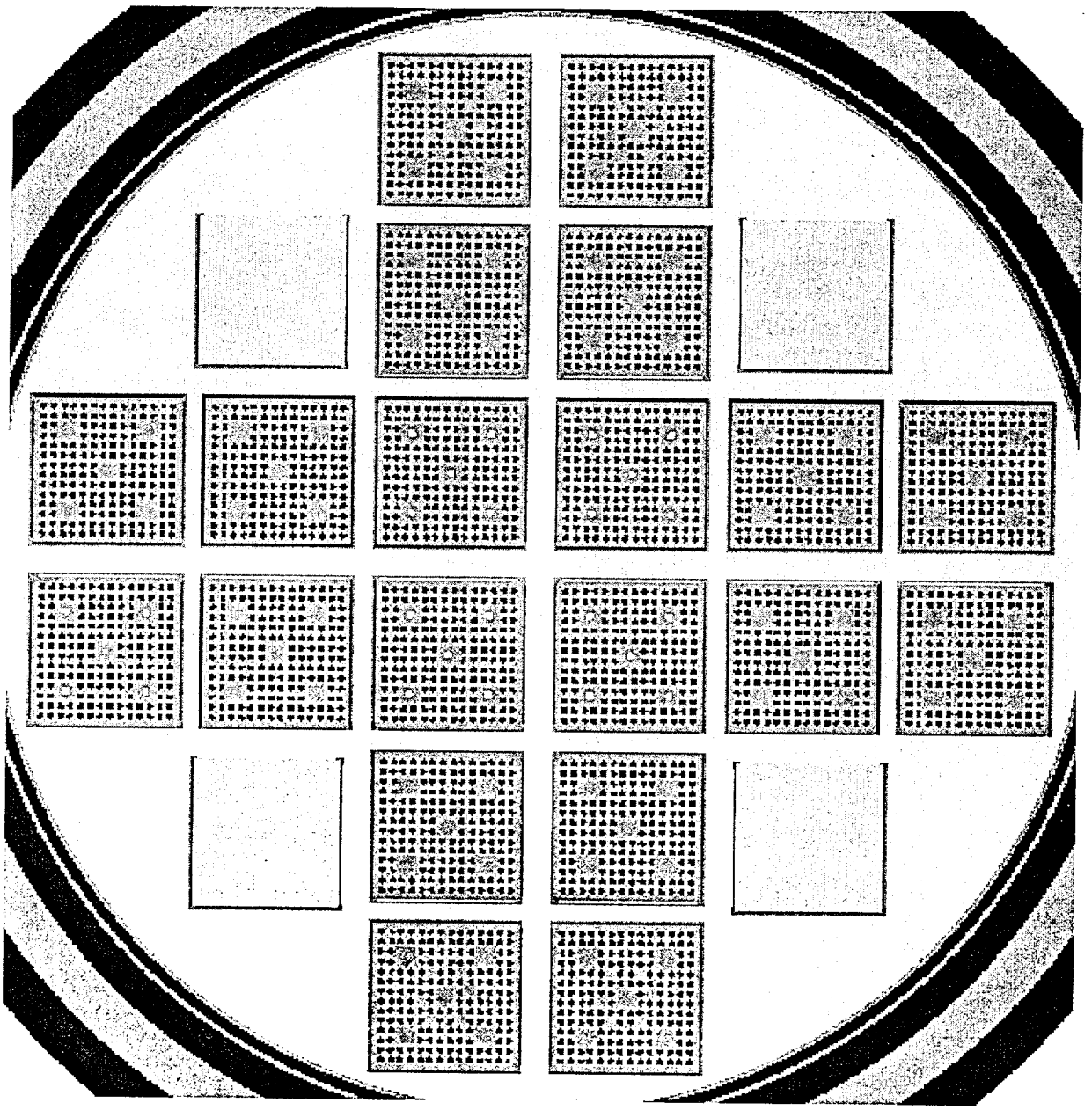

Figure 6.6.1-7 Model with Pellets in Upper Hardware Zones- Configuration A

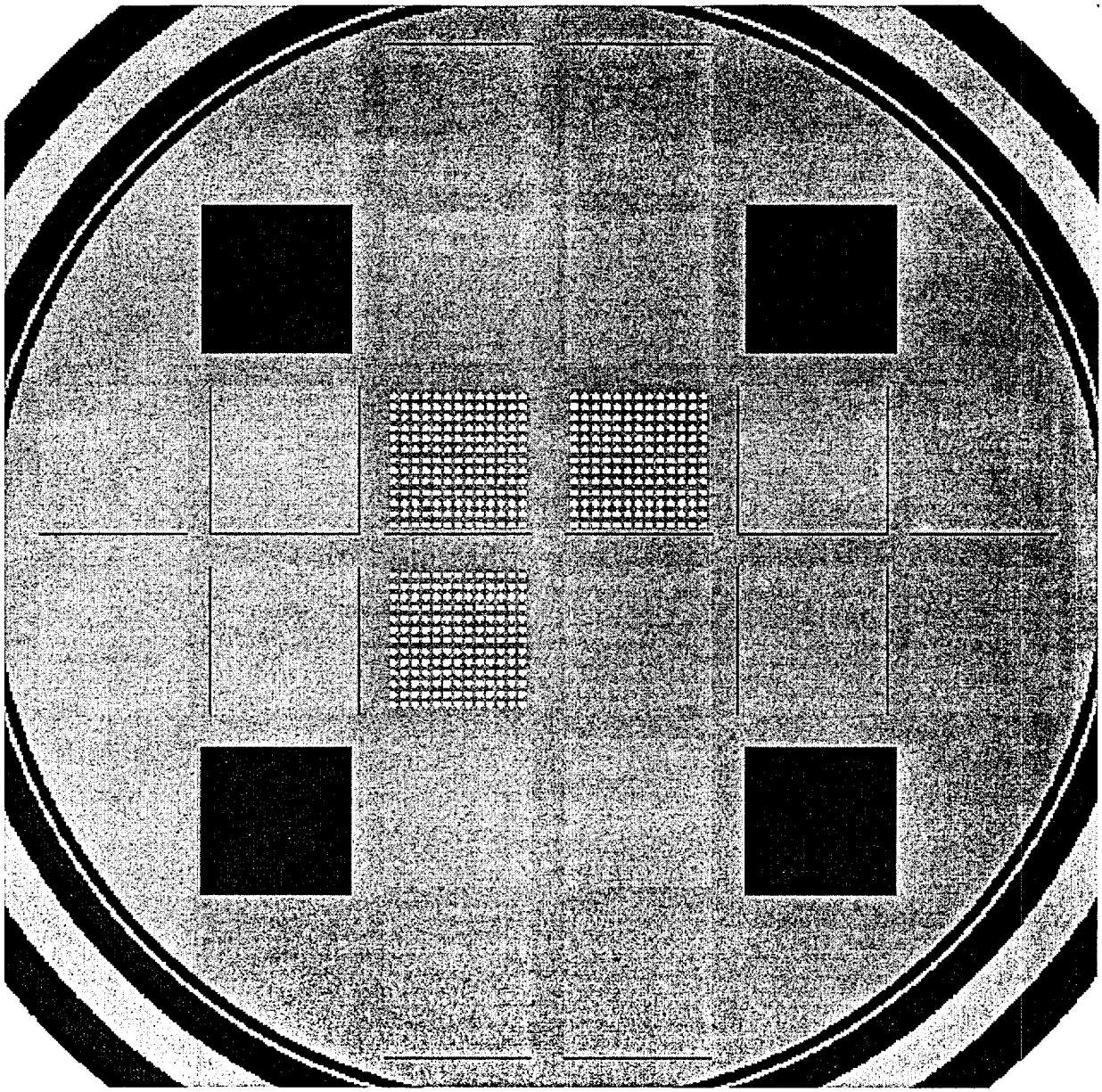

Figure 6.6.1-8 Model with Pellets in Lower Hardware Zones- Configuration F

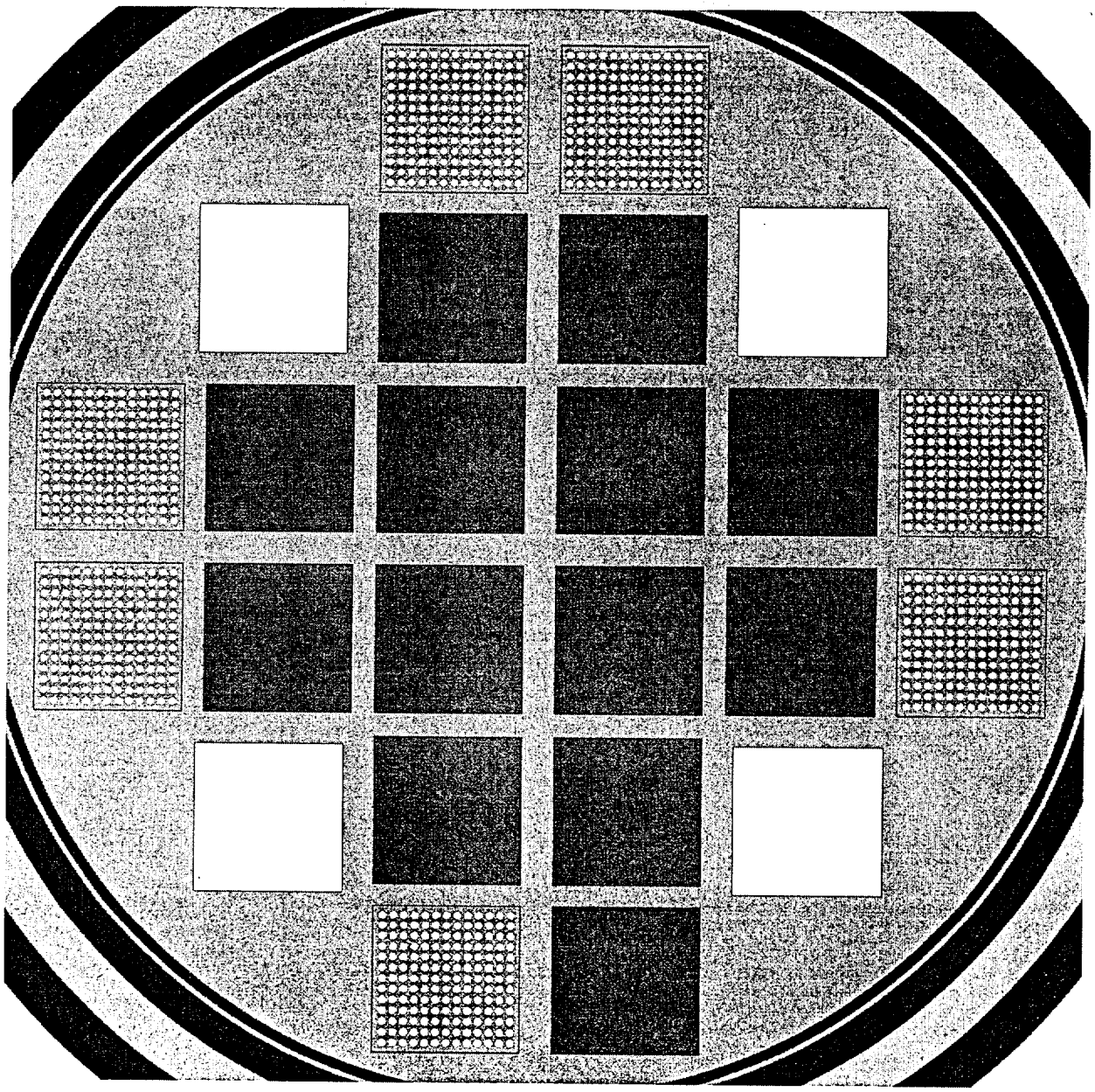

Figure 6.6.1-9 Model with Dispersed Fuel Mixture in Flux Trap - Configuration A

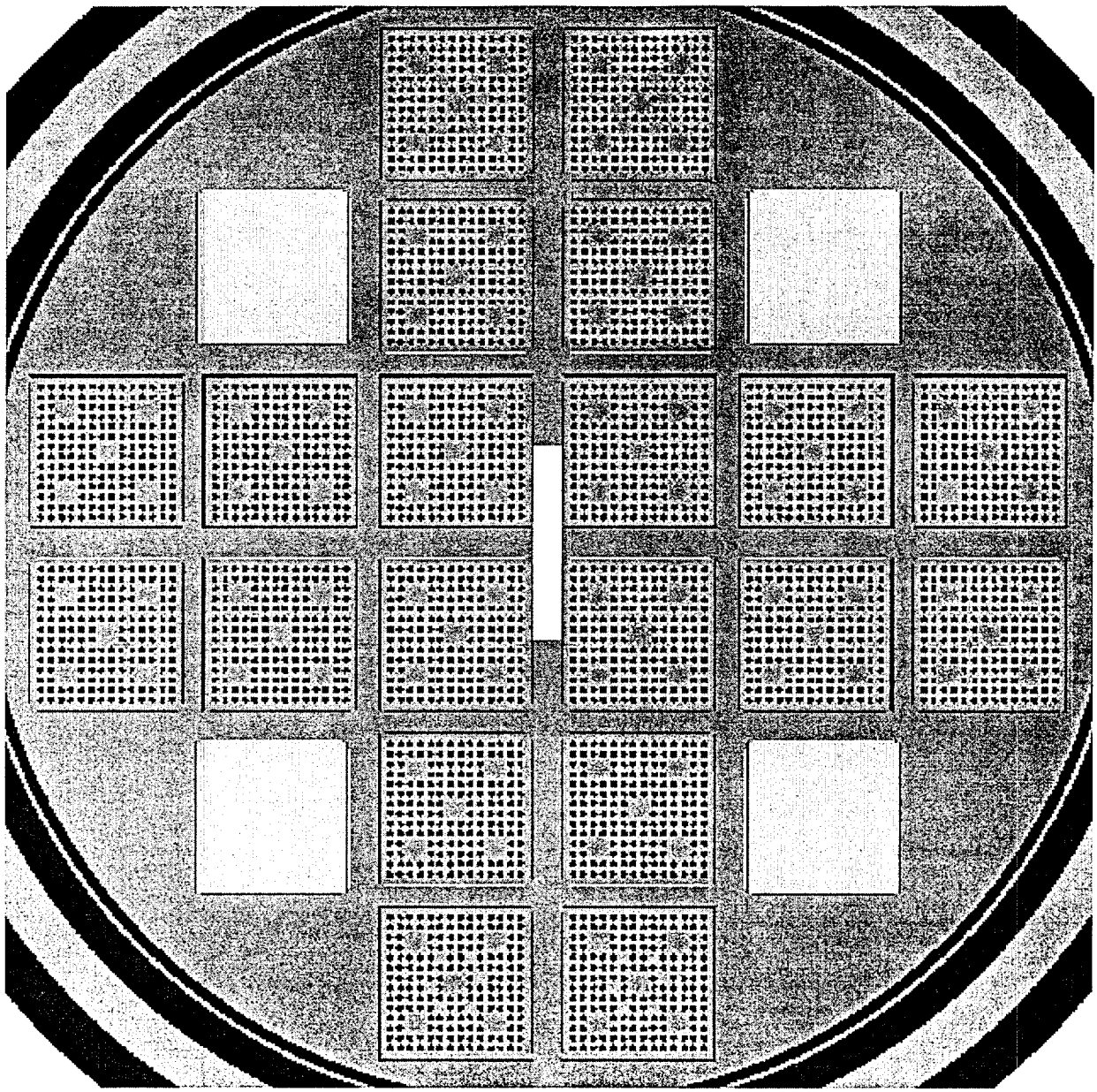

Figure 6.6.1-10 Model with Dispersed Fuel Mixture in Flux Trap - Configuration B

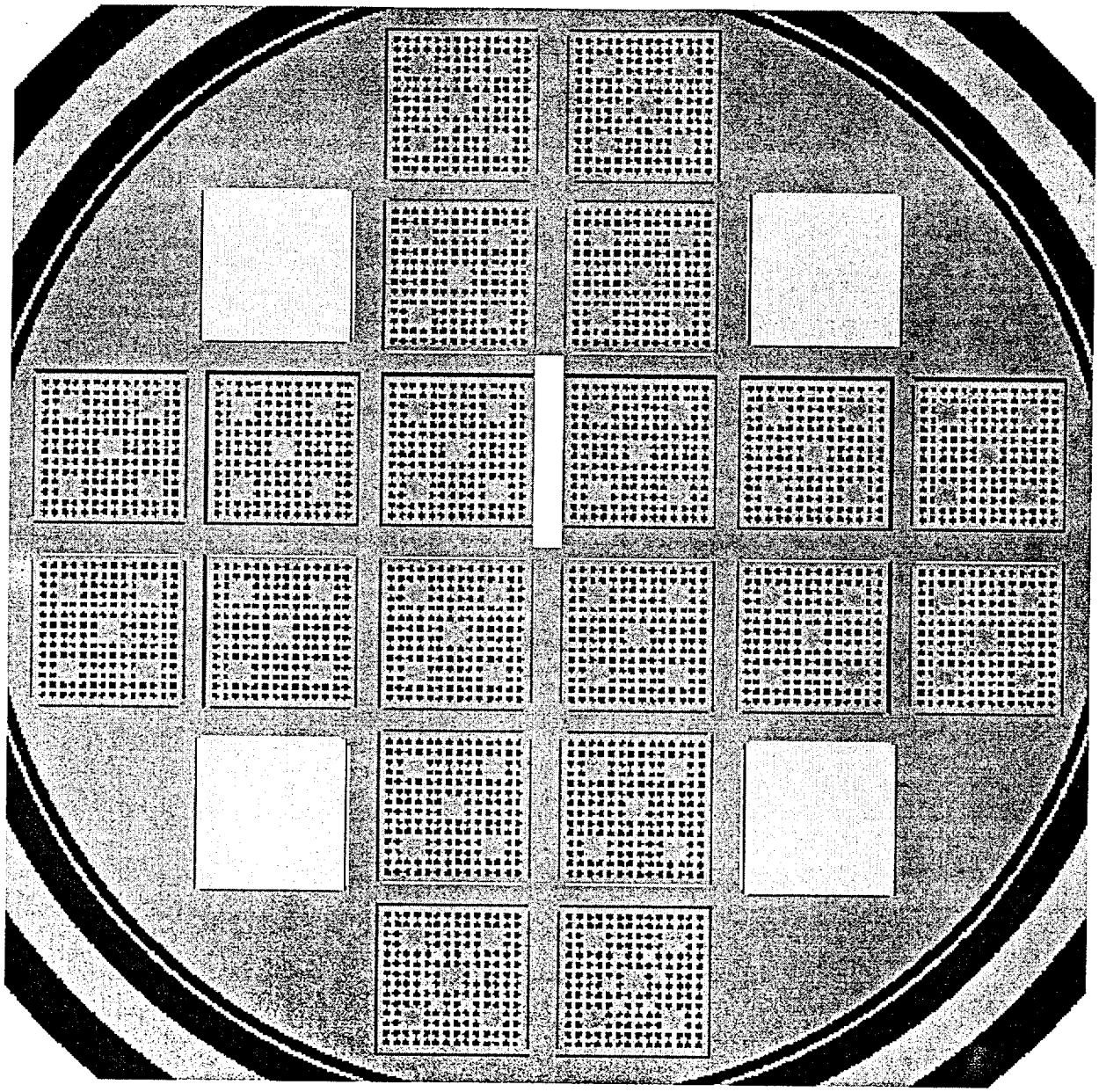

Figure 6.6.1-11 Model with Dispersed Fuel Mixture in Guide Tubes - Configuration C

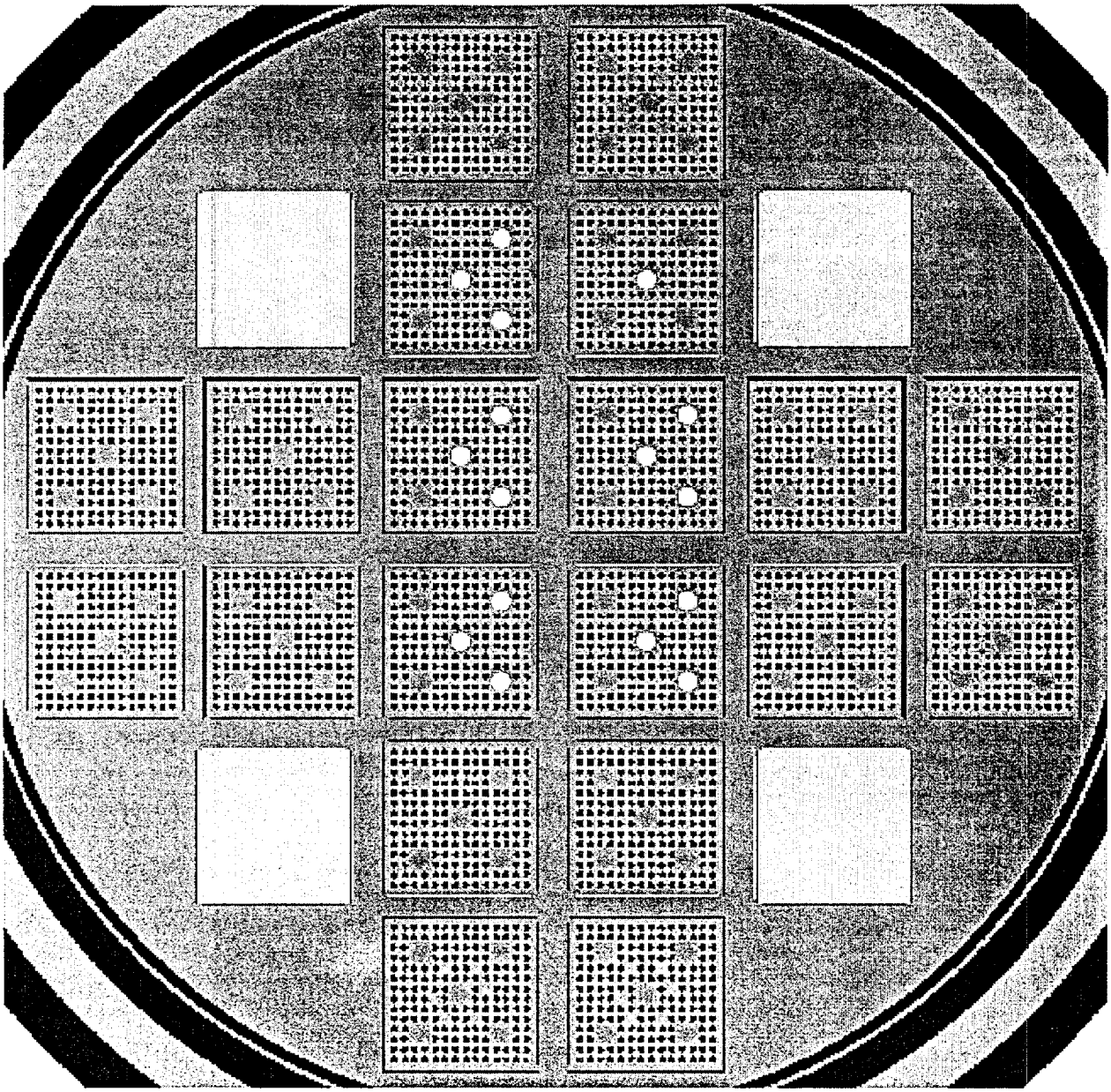

Figure 6.6.1-12 Model with Dispersed Fuel Mixture in Top Hardware - Configuration B

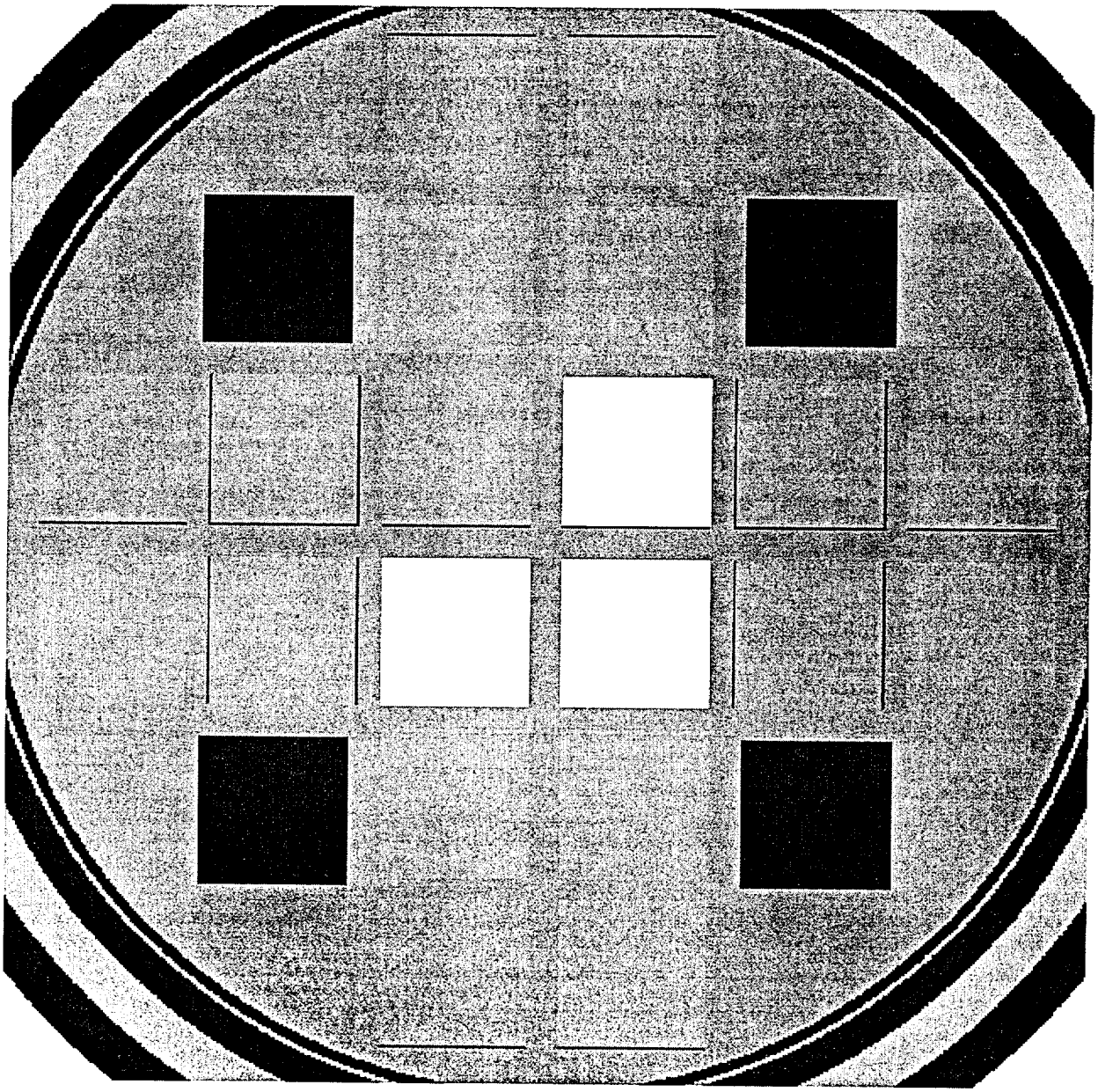

Figure 6.6.1-13 Model with Dispersed Fuel Mixture in Bottom Hardware - Configuration F

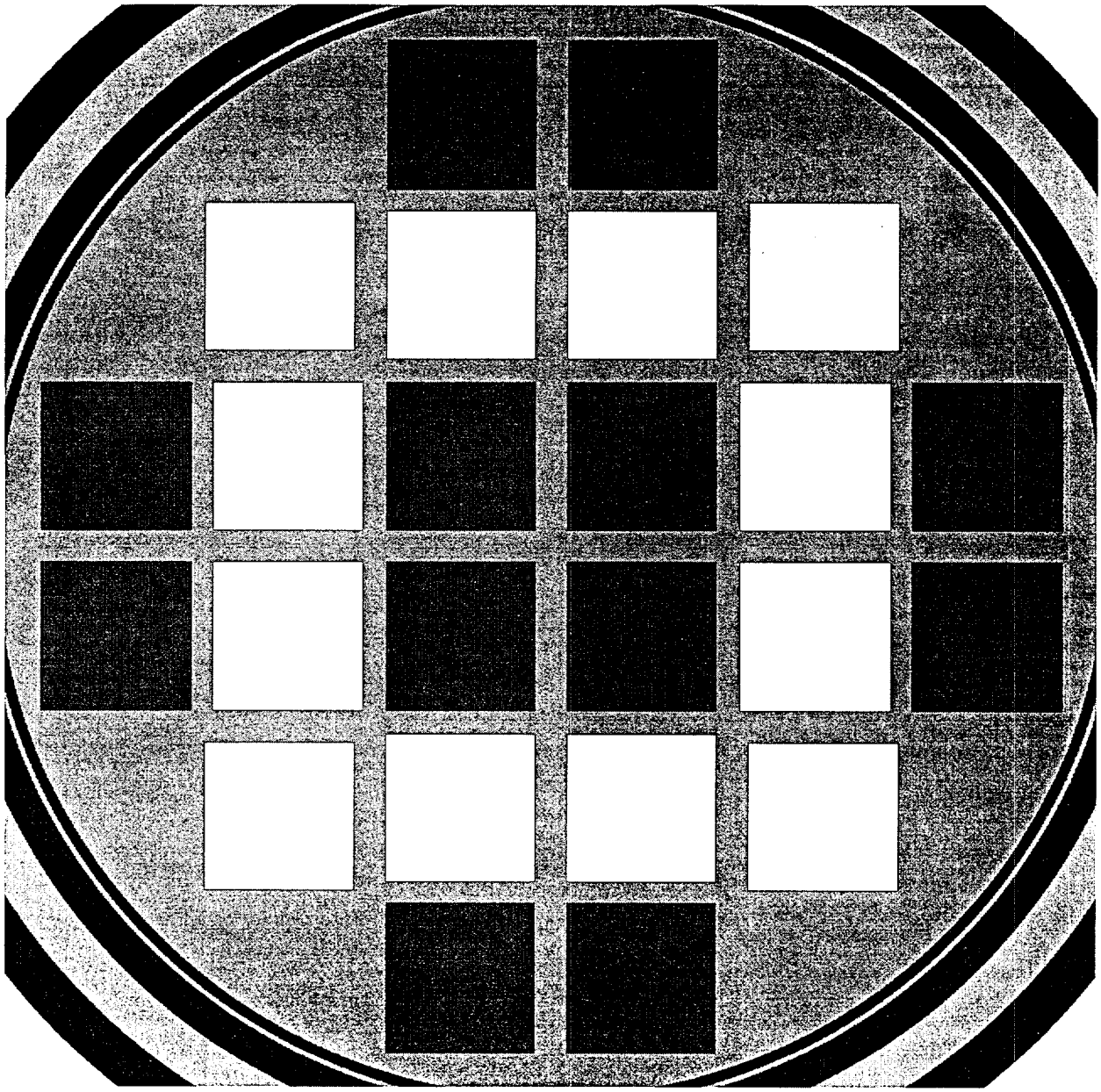

Figure 6.6.1-14 Top View of Boronometer Source Heterogeneous Model Geometry

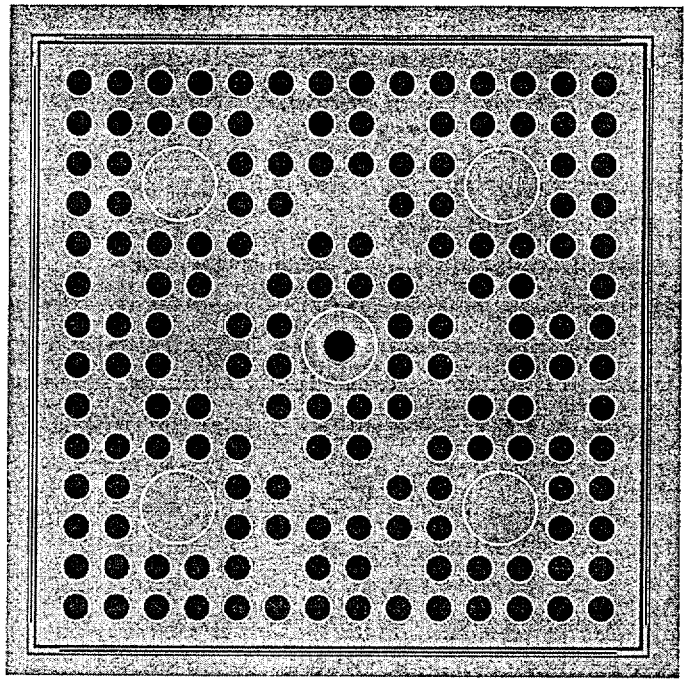

Figure 6.6.1-15 Side View of Boronometer Source Heterogeneous Model Geometry

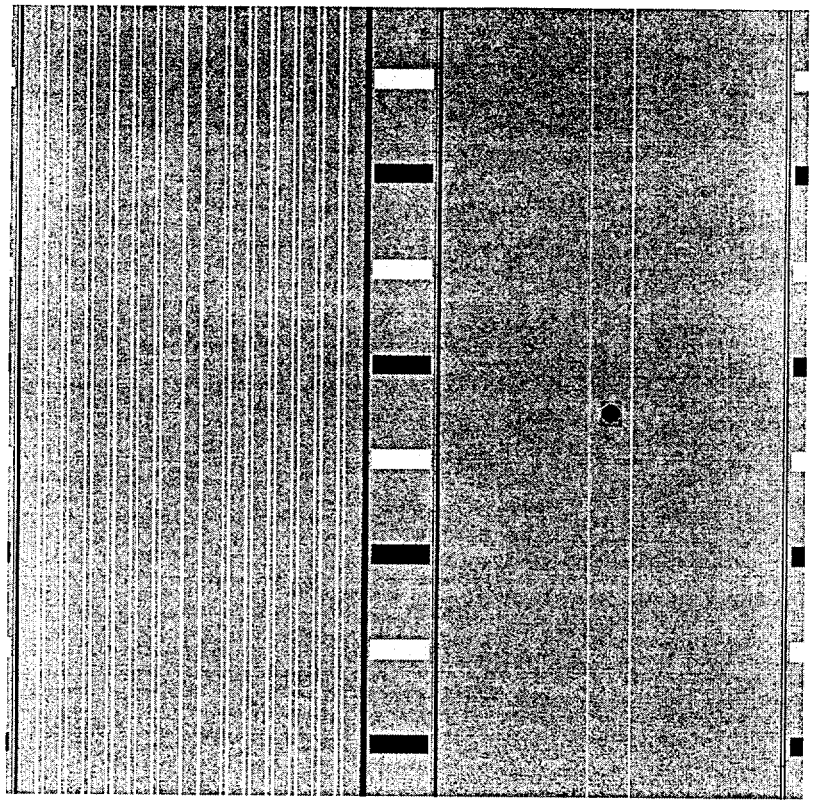

Table 6.6.1-1 Maine Yankee Standard Fuel Characteristics

Fuel Class[1]	Vendor	Array	Version	Number of Fuel Rods	Pitch (in.)	Rod Diameter (in.)	Clad ID (in.)	Clad Thickness (in.)	Pellet Diameter (in.)	GT[2] Thickness (in.)
1	CE	14x14	Std.	160[3]-176	0.570-0.590	0.438-0.442	0.3825-0.3895	0.024-0.028	0.376-0.380	0.036-0.040
1	Ex/ANF	14x14	CE	164[4]-176	0.580	0.438-0.442	0.3715-0.3795	0.0294-0.031	0.3695-0.3705	0.036-0.040
1	WE	14x14	CE	176	0.575-0.585	0.438-0.442	0.3825-0.3855	0.0262-0.028	0.376-0.377	0.034-0.038

1. All fuel rods are Zircaloy clad.
2. Guide Tube thickness.
3. Up to 16 fuel rod positions may have solid filler rods or burnable poison rods.
4. Up to 12 fuel rod positions may have solid filler rods or burnable poison rods.

Table 6.6.1-2 Maine Yankee Most Reactive Fuel Dimensions

Parameter	Bounding Dimensional Value
Maximum Rod Enrichment[1]	4.2 wt % ^{235}U
Maximum Number of Fuel Rods[2]	176
Maximum Pitch (in.)	0.590
Maximum Active Length (in.)	N/A – Infinite Model
Minimum Clad OD (in.)	0.4375
Maximum Clad ID (in.)	0.3895
Minimum Clad Thickness (in.)	0.024
Maximum Pellet Diameter (in.)	0.3800 - Study
Minimum Guide Tube OD (in.)	1.108
Maximum Guide Tube ID (in.)	1.040
Minimum Guide Tube Thickness (in.)	0.034

1. Variably enriched fuel assemblies may have a maximum fuel rod enrichment of 4.21 wt % ^{235}U with a maximum planar average enrichment of 3.99 wt % ^{235}U.
2. Assemblies with less than 176 fuel rods or solid dummy rods are addressed after the determination of the most reactive dimensions.

Table 6.6.1-3 Maine Yankee Pellet Diameter Study

Diameter (inches)	k_{eff}	σ	$k_{eff} + 2\sigma$
0.3800	0.95585	0.00085	0.95755
0.3779	0.95784	0.00080	0.95944
0.3758	0.95714	0.00085	0.95884
0.3737	0.95863	0.00082	0.96027
0.3716	0.95862	0.00084	0.96030
0.3695	0.95855	0.00083	0.96021
0.3674	0.95863	0.00085	0.96033
0.3653	0.95982	0.00084	0.96150
0.3632	0.95854	0.00088	0.96030
0.3611	0.95966	0.00083	0.96132
0.3590	0.95990	0.00084	0.96158
0.3569	0.96082	0.00082	0.96246
0.3548	0.96053	0.00083	0.96219
0.3527	0.96104	0.00082	0.96268
0.3506	0.95964	0.00087	0.96138
0.3485	0.95993	0.00086	0.96165
0.3464	0.95916	0.00084	0.96084
0.3443	0.95847	0.00083	0.96013
0.3422	0.95876	0.00083	0.96042
0.3401	0.95865	0.00081	0.96027
0.3380	0.95734	0.00084	0.95902

Table 6.6.1-4 Maine Yankee Annular Fuel Results

Case Description	k_{eff}	σ	$k_{eff} + 2\sigma$
All pellets with a diameter of 0.3527 inches	0.90896	0.00083	0.91061
Annular pellet diameter changed to 0.3800 inches	0.91013	0.00087	0.91187

Table 6.6.1-5 Maine Yankee Removed Rod Results with Small Pellet Diameter

Number of Removed Rods	Number of Fuel Rods	k_{eff}	σ	$k_{eff} + 2\sigma$
4	172	0.91171	0.00088	0.91347
4	172	0.91292	0.00086	0.91464
4	172	0.91479	0.00081	0.91640
4	172	0.91125	0.00087	0.91299
6	170	0.91418	0.00087	0.91592
6	170	0.91264	0.00085	0.91435
6	170	0.91314	0.00086	0.91487
6	170	0.90322	0.00086	0.90493
8	168	0.91555	0.00087	0.91729
8	168	0.91490	0.00093	0.91676
8	168	0.91457	0.00088	0.91633
8	168	0.91590	0.00087	0.91764
8	168	0.89729	0.00088	0.89905
12	164	0.91654	0.00086	0.91827
12	164	0.91469	0.00085	0.91639
12	164	0.91149	0.00083	0.91315
16	160	0.91725	0.00084	0.91893
16	160	0.91567	0.00084	0.91735
16	160	0.90986	0.00088	0.91162
16	160	0.90849	0.00083	0.91015
16	160	0.90704	0.00086	0.90876
24	152	0.91572	0.00083	0.91739
32	144	0.91037	0.00088	0.91213
48	128	0.89385	0.00085	0.89554
48	128	0.84727	0.00079	0.84886
64	112	0.79602	0.00083	0.79768
96	80	0.69249	0.00077	0.69402
Westinghouse 17 x 17 OFA		0.9192	0.0009	0.9210

Table 6.6.1-6 Maine Yankee Removed Fuel Rod Results with Maximum Pellet Diameter

Number of Removed Rods	Number of Fuel Rods	k_{eff}	σ	$k_{eff} + 2\sigma$
4	172	0.91078	0.00086	0.91250
4	172	0.90916	0.00085	0.91085
4	172	0.91164	0.00087	0.91338
4	172	0.90809	0.00085	0.90979
6	170	0.91223	0.00085	0.91393
6	170	0.91223	0.00080	0.91384
6	170	0.91270	0.00086	0.91442
6	170	0.90245	0.00086	0.90416
6	170	0.89801	0.00086	0.89972
8	168	0.91567	0.00085	0.91736
8	168	0.91448	0.00085	0.91618
8	168	0.91355	0.00086	0.91526
8	168	0.91293	0.00085	0.91463
12	164	0.91639	0.00090	0.91818
12	164	0.91803	0.00086	0.91974
12	164	0.91235	0.00083	0.91401
16	160	0.91665	0.00091	0.91847
16	160	0.92136	0.00087	0.92310
16	160	0.91231	0.00084	0.91400
16	160	0.90883	0.00087	0.91057
24	152	0.92227	0.00087	0.92400
32	144	0.92164	0.00088	0.92340
48	128	0.91212	0.00081	0.91373
48	128	0.86308	0.00082	0.86472
64	112	0.81978	0.00080	0.82138
88	88	0.72087	0.00083	0.72247
24 (Four Corners)	152	0.91153	0.00085	0.91323
Westinghouse 17 x 17 OFA		0.9192	0.0009	0.9210

Table 6.6.1-7 Maine Yankee Fuel Rods in Guide Tube Results

Number of Guide Tubes with Rods	Number of Rods in Each	k_{eff}	σ	$k_{eff} + 2\sigma$
1	1	0.91102	0.00089	0.91280
2	1	0.91059	0.00088	0.91234
3	1	0.91172	0.00087	0.91346
5	1	0.91411	0.00086	0.91583
1	2	0.91169	0.00090	0.91349
2	2	0.91201	0.00087	0.91375
3	2	0.91173	0.00086	0.91344
5	2	0.91357	0.00086	0.91529
Design Basis Westinghouse 17 x 17 OFA		0.9192	0.0009	0.9210

Table 6.6.1-8 Maine Yankee Consolidated Fuel Empty Fuel Rod Position Results

Number of Empty Positions	Number of Fuel Rods	k_{eff}	σ	$k_{eff} + 2\sigma$
4	285	0.79684	0.00082	0.79848
9	280	0.80455	0.00081	0.80616
9	280	0.80812	0.00079	0.80970
13	276	0.81573	0.00083	0.81739
24	265	0.84187	0.00080	0.84347
25	264	0.84017	0.00083	0.84182
25	264	0.84634	0.00081	0.84795
25	264	0.84583	0.00083	0.84750
25	264	0.85524	0.00083	0.85690
25	264	0.83396	0.00081	0.83558
25	264	0.84625	0.00083	0.84790
27	262	0.85438	0.00083	0.85604
29	260	0.85179	0.00081	0.85340
31	258	0.85930	0.00084	0.86098
33	256	0.86407	0.00082	0.86571
35	254	0.86740	0.00082	0.86904
37	252	0.87372	0.00084	0.87541
45	244	0.88630	0.00081	0.88793
45	244	0.87687	0.00079	0.87844
52	237	0.90062	0.00083	0.90228
57	232	0.87975	0.00087	0.88149
61	258	0.89055	0.00083	0.89221
73	216	0.90967	0.00082	0.91131
84	205	0.93261	0.00091	0.93443
85	204	0.94326	0.00086	0.94499
113	176	0.95626	0.00084	0.95794
117	172	0.95373	0.00088	0.95549
119	170	0.95315	0.00085	0.95485
125	164	0.95020	0.00086	0.95192
141	148	0.94348	0.00086	0.94521
145	144	0.93868	0.00089	0.94047
113 (Four Corners)	176	0.91292	0.00087	0.91466
Design Basis Westinghouse 17 x 17 OFA		0.9192	0.0009	0.9210

Table 6.6.1-9 Fuel Can Infinite Height Model Results of Fuel - Water Mixture Between Rods

Volume Fraction of UO_2 in Water	k_{eff}	Δk_{eff} to 24 (Four Corners)[1]
0.000	0.91090	-0.00063
0.001	0.91138	-0.00015
0.002	0.91120	-0.00033
0.003	0.91177	0.00024
0.004	0.91285	0.00132
0.005	0.90908	-0.00245
0.006	0.91001	-0.00152
0.007	0.90895	-0.00258
0.008	0.91005	-0.00148
0.009	0.90986	-0.00167
0.010	0.90864	-0.00289
0.020	0.91003	-0.00150
0.030	0.90963	-0.00190
0.040	0.91063	-0.00090
0.050	0.90931	-0.00222
0.060	0.90765	-0.00388
0.070	0.90753	-0.00400
0.080	0.91088	-0.00065
0.090	0.91122	-0.00031
0.100	0.90879	-0.00274
0.150	0.90968	-0.00185
0.200	0.90952	-0.00201
0.250	0.90815	-0.00338
0.300	0.90748	-0.00405
0.350	0.90581	-0.00572
0.400	0.90963	-0.00190
0.450	0.90547	-0.00606
0.500	0.90603	-0.00550
0.550	0.90753	-0.00400
0.600	0.90674	-0.00479
0.650	0.90589	-0.00564
0.700	0.90594	-0.00559
0.750	0.90568	-0.00585
0.800	0.90532	-0.00621
0.850	0.90693	-0.00460
0.900	0.90639	-0.00514
0.950	0.90684	-0.00469
1.000	0.90677	-0.00476

Table 6.6.1-10 Fuel Can Finite Model Results of Fuel-Water Mixture Outside BORAL Coverage

Volume Fraction of UO_2 in Water	k_{eff}	Δk_{eff} to 0.00 UO_2 in Water	Δk_{eff} to 24 (Four Corners)[1]
0.00	0.91045[2]	NA	-0.00108
0.05	0.90781	-0.00264	-0.00372
0.10	0.90978	-0.00067	-0.00175
0.15	0.91048	0.00003	-0.00105
0.20	0.90916	-0.00129	-0.00237
0.25	0.90834	-0.00211	-0.00319
0.30	0.90935	-0.00110	-0.00218
0.35	0.90786	-0.00259	-0.00367
0.40	0.90892	-0.00153	-0.00261
0.45	0.91015	-0.00030	-0.00138
0.50	0.91011	-0.00034	-0.00142
0.55	0.91003	-0.00042	-0.00150
0.60	0.90874	-0.00171	-0.00279
0.65	0.91165	0.00120	0.00012
0.70	0.90977	-0.00068	-0.00176
0.75	0.90813	-0.00232	-0.00340
0.80	0.90909	-0.00136	-0.00244
0.85	0.91028	-0.00017	-0.00125
0.90	0.91061	0.00016	-0.00092
0.95	0.91129	0.00084	-0.00024
1.00	0.91076	0.00031	-0.00077

1. See Table 6.6.1-6.
2. $\sigma = 0.00084$.

Table 6.6.1-11 Fuel Can Finite Model Results of Replacing All Rods with Fuel-Water Mixture

Volume Fraction of UO_2 in Water	k_{eff}	Δk_{eff} to 24 (Four Corners) Finite Height Model	Δk_{eff} to 24 (Four Corners) Infinite Height Model
0	0.90071	-0.00974	-0.01082
5	0.90194	-0.00851	-0.00959
10	0.90584	-0.00461	-0.00569
15	0.90837	-0.00208	-0.00316
20	0.91008	-0.00037	-0.00145
25	0.91086	0.00041	-0.00067
30	0.90964	-0.00081	-0.00189
35	0.90828	-0.00217	-0.00325
40	0.90805	-0.00240	-0.00348
45	0.90730	-0.00315	-0.00423
50	0.90637	-0.00408	-0.00516
55	0.90672	-0.00373	-0.00481
60	0.90649	-0.00396	-0.00504
65	0.90632	-0.00413	-0.00521
70	0.90435	-0.00610	-0.00718
75	0.90792	-0.00253	-0.00361
80	0.90376	-0.00669	-0.00777
85	0.90528	-0.00517	-0.00625
90	0.90454	-0.00591	-0.00699
95	0.90360	-0.00685	-0.00793
100	0.90416	-0.00629	-0.00737

Table 6.6.1-12 Various Configurations of Pellets Dispersed in Flux Traps

Configuration	k_{eff}	σ	$k_{eff}+2\sigma$
A	0.9118	0.0006	0.9130
B	0.9132	0.0006	0.9144
C	0.9110	0.0007	0.9124
D	0.9062	0.0006	0.9074
E	0.9087	0.0007	0.9101
F	0.9102	0.0007	0.9116
G	0.9081	0.0007	0.9095
H	0.9133	0.0006	0.9145
I	0.9115	0.0007	0.9129
J	0.9084	0.0007	0.9098
K	0.9094	0.0007	0.9108
B_accident	0.9201	0.0007	0.9215
H_accident	0.9201	0.0006	0.9213

Table 6.6.1-13 Various Configurations of Pellets Dispersed in Guide Tubes

Configuration	k_{eff}	σ	$k_{eff}+2\sigma$
A	0.9057	0.0007	0.9071
B	0.9057	0.0006	0.9069
C	0.9056	0.0006	0.9068
D	0.9067	0.0007	0.9081
E	0.9056	0.0007	0.9070
F	0.9047	0.0007	0.9061
G	0.9067	0.0006	0.9079
H	0.9064	0.0006	0.9076
I	0.9044	0.0006	0.9056
J	0.9038	0.0006	0.9050

Table 6.6.1-14 Various Configurations of Pellets Dispersed in Hardware

Configuration	k_{eff}	σ	$k_{eff} + 2\sigma$
A	0.9041	0.0005	0.9051
B	0.9035	0.0004	0.9043
C	0.9035	0.0005	0.9045
D	0.9032	0.0004	0.9040
E	0.9035	0.0005	0.9045
F	0.9038	0.0004	0.9046

Table 6.6.1-15 Various Configurations of Dispersed Fuel Mixture in Flux Traps

Configuration	k_{eff}	σ	$k_{eff} + 2\sigma$
A	0.9136	0.0007	0.9150
B	0.9136	0.0007	0.9150
C	0.9132	0.0007	0.9146
D	0.9066	0.0006	0.9078
E	0.9124	0.0006	0.9136
F	0.9084	0.0006	0.9096
A_accident	0.9215	0.0007	0.9229
B_accident	0.9211	0.0007	0.9225

Table 6.6.1-16 Various Configurations of Dispersed Fuel Mixture in Guide Tubes

Configuration	k_{eff}	σ	$k_{eff} + 2\sigma$
A	0.9044	0.0006	0.9056
B	0.9037	0.0006	0.9049
C	0.9057	0.0007	0.9071

Table 6.6.1-17 Various Configurations of Dispersed Fuel Mixture in Hardware

Configuration	k_{eff}	σ	$k_{eff} + 2\sigma$
A	0.9038	0.0004	0.9046
B	0.9096	0.0006	0.9108
C	0.9040	0.0005	0.9050
D	0.9036	0.0004	0.9044
E	0.9033	0.0005	0.9043
F	0.9042	0.0004	0.9050

Table 6.6.1-18 Infinite Height Analysis of Maine Yankee Startup Sources

Pu Vf	Be Vf	H_2O Vf	Void Vf	k_{eff}	sd	k_{eff}+2sd	Delta K*
0	0.5	0.5	0	0.91085	0.00087	0.91259	-0.00068
0.008	0.992	0	0	0.91034	0.00089	0.91212	-0.00119
0.008	0.9	0.092	0	0.91151	0.00087	0.91325	-0.00002
0.008	0.8	0.192	0	0.91138	0.00087	0.91312	-0.00015
0.008	0.7	0.292	0	0.91042	0.00085	0.91212	-0.00111
0.008	0.6	0.392	0	0.91231	0.00086	0.91403	0.00078
0.008	0.5	0.492	0	0.90922	0.00083	0.91088	-0.00231
0.008	0.4	0.592	0	0.91197	0.00087	0.91371	0.00044
0.008	0.3	0.692	0	0.91203	0.00086	0.91375	0.00050
0.008	0.2	0.792	0	0.90922	0.00084	0.91090	-0.00231
0.008	0.1	0.892	0	0.91140	0.00085	0.91310	-0.00013
0.008	0	0.992	0	0.91149	0.00086	0.91321	-0.00004
0.008	0.9	0	0.092	0.91075	0.00087	0.91249	-0.00078
0.008	0.8	0	0.192	0.91143	0.00091	0.91325	-0.00010
0.008	0.7	0	0.292	0.91182	0.00086	0.91354	0.00029
0.008	0.6	0	0.392	0.91072	0.00082	0.91236	-0.00081
0.008	0.5	0	0.492	0.90984	0.00085	0.91154	-0.00169
0.008	0.4	0	0.592	0.90982	0.00091	0.91164	-0.00171
0.008	0.3	0	0.692	0.91055	0.00087	0.91229	-0.00098
0.008	0.2	0	0.792	0.91054	0.00085	0.91224	-0.00099
0.008	0.1	0	0.892	0.91006	0.00088	0.91182	-0.00147
0.008	0	0	0.992	0.90957	0.00086	0.91129	-0.00196

*Change in reactivity from case "24 (Four Corners)" in Table 6.6.1-6.

Table 6.6.1-19 Heterogeneous Finite Height Analysis of Boronometer Source

Pu Vf of Sphere	Be Vf	H_2O Vf	k_{eff}	sd	k_{eff}+2sd	Delta-k*
1	0	1	0.90932	0.00086	0.91104	-0.00113
1	0.1	0.9	0.90890	0.00085	0.91060	-0.00155
1	0.2	0.8	0.91007	0.00089	0.91185	-0.00038
1	0.3	0.7	0.91024	0.00085	0.91194	-0.00021
1	0.4	0.6	0.90925	0.00084	0.91093	-0.00120
1	0.5	0.5	0.90908	0.00086	0.91080	-0.00137
1	0.6	0.4	0.90934	0.00086	0.91106	-0.00111
1	0.7	0.3	0.90944	0.00087	0.91118	-0.00101
1	0.8	0.2	0.90877	0.00085	0.91047	-0.00168
1	0.9	0.1	0.90995	0.00089	0.91173	-0.00050
1	1	0	0.90998	0.00085	0.91168	-0.00047

*Change in reactivity from case w/ 0.00 UO_2 outside of BORAL in Table 6.6.1-10.

Table 6.6.1-20 Homogeneous Finite Height Analysis of Boronometer Source

Pu Vf	Be Vf	H_2O Vf	k_{eff}	sd	k_{eff}+2sd	Delta-k*
0.422	0	0.578	0.90889	0.00088	0.91065	-0.00156
0.422	0.05	0.528	0.90967	0.00089	0.91145	-0.00078
0.422	0.10	0.478	0.91004	0.00084	0.91172	-0.00041
0.422	0.15	0.428	0.90807	0.00087	0.90981	-0.00238
0.422	0.20	0.378	0.90988	0.00081	0.9115	-0.00057
0.422	0.25	0.328	0.91029	0.00085	0.91199	-0.00016
0.422	0.30	0.278	0.90899	0.00085	0.91069	-0.00146
0.422	0.35	0.228	0.90863	0.00084	0.91031	-0.00182
0.422	0.40	0.178	0.90825	0.00086	0.90997	-0.00220
0.422	0.45	0.128	0.90878	0.00085	0.91048	-0.00167
0.422	0.50	0.078	0.91017	0.00085	0.91187	-0.00028
0.422	0.55	0.028	0.90841	0.00081	0.91003	-0.00204
0.422	0.578	0.000	0.90926	0.00081	0.91088	-0.00119

*Change in reactivity from case w/ 0.00 UO_2 outside of BORAL in Table 6.6.1-10.

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9.2 Maintenance Program

This section presents the maintenance requirements for the UMS® Universal Storage System and for the transfer cask.

9.2.1 UMS® Storage System Maintenance

The UMS® Universal Storage System is a passive system. No active components or systems are incorporated in the design. Consequently, only a minimal amount of maintenance is required over its lifetime.

The UMS®-Universal Storage System has no valves, gaskets, rupture discs, seals, or accessible penetrations. Consequently, there is no maintenance associated with these types of features.

The routine surveillance requirements are described in Technical Specification LCO 3.1.6 in Appendix 12A of Chapter 12. It is not necessary to inspect the concrete cask or canister during the storage period as long as the thermal performance is normal, based on daily temperature verification.

The ambient air temperature and air outlet temperature of each Vertical Concrete Cask must be recorded upon placement in service. Thereafter, the temperatures shall be recorded on a daily basis to verify the continuing thermal performance of the system.

In the event of a decline in thermal performance, the heat removal system must be restored to acceptable operation. The user should perform a visual inspection of air inlets and outlets for evidence of blockage and verify that the inlet and outlet screens are whole, secure and in place.

The user must also visually inspect the Vertical Concrete Cask within 4 hours of any off-normal, accident or natural phenomena event, such as an earthquake.

An annual inspection of the Vertical Concrete Cask exterior is required, to include:

- Visual inspection of concrete surfaces for chipping, spalling or other surface defects. Any defects larger than one inch in diameter (or width) and deeper than one inch shall be regrouted, according to the grout manufacturer's recommendations.
- Reapplication of corrosion-inhibiting (external) coatings on accessible surfaces, including concrete cask lifting lugs, if present.

9.2.2 Transfer Cask Maintenance

The transfer cask trunnions and shield door assemblies shall be visually inspected for gross damage and proper function prior to each use. Annually, the lifting trunnions, shield doors and shield door rails shall be either dye penetrant or magnetic particle examined, using the examination method appropriate to the material. The examination method shall be in accordance with Section V of the ASME Code. The acceptance criteria shall be in accordance with Section III, Subsection NF, Article NF-5350 or NF-5340 as appropriate to the examination method, as required by ANSI N14.6.

The annual examination may be omitted in periods of nonuse of the transfer cask, provided that the transfer cask examination is performed prior to the next use of the transfer cask.

Annually, the coating applied to the carbon steel surfaces of the transfer cask shall be inspected, and any chips, cracks or other defects in the coating shall be repaired.

10.3 Estimated On-Site Collective Dose Assessment

Occupational radiation exposures (person-mrem) resulting from the use of the Universal Storage System are calculated using the estimated exposure rates presented in Sections 5.1.3, 5.4.3 and 10.2.1. Exposure is evaluated by identifying the tasks and estimating the duration and number of personnel performing those tasks based on industry experience. The tasks identified are based on the design basis operating procedures, as presented in Chapter 8.0.

Dose rates are calculated using the shielding analysis design basis fuel assemblies. The shielding design basis PWR assembly is the Westinghouse 17X17 Standard fuel assembly, with an initial enrichment of 3.7 wt % ^{235}U. The design basis BWR assembly is the GE 9X9, with 79 fuel rods and an initial enrichment of 3.25 wt % ^{235}U. Both design basis fuel assemblies have an assumed burnup of 40,000 MWD/MTU, and a cool time of 5 years. The selection of these assemblies for the shielding design basis is described in Section 5.1. The principal parameters of these assemblies are presented in Table 2.1-1.

10.3.1 Estimated Collective Dose for Loading a Single Universal Storage System

This section estimates the collective dose due to the loading, sealing, transfer and placement on the ISFSI pad, of the Universal Storage System. The analysis assumes that the exposure incurred by the operators is independent of background radiation, as background radiation varies from site to site. The number of persons allocated to task completion is generally the minimum number required for the task. Working area exposure rates are assigned based on the orientation of the worker with respect to the source and take into account the use of temporary shielding.

Table 10.3-1 summarizes the estimated total exposure by task, attributable to the loading, transfer, sealing and placement of a design basis Universal Storage System. Exposures associated with shield lid operations are based on the presence of a 5-inch thick steel temporary shield.

This estimated dose is considered to be conservative as it assumes the loading of a cask with design basis fuel, and does not account for efficiencies in the loading process that occur with experience.

10.3.2 Estimated Annual Dose Due to Routine Operations

Once in place, the ISFSI requires limited ongoing inspection and surveillance throughout its service life. The annual dose evaluations presented in Tables 10.3-4 through 10.3-7 estimate the exposure due to a combination of inspection and surveillance activities and other tasks that are anticipated to be representative of an operational facility. The visual inspection exposure, based on a daily inspection of the storage cask or storage cask array, is provided for information only since a daily inspection is not required as long as the temperature monitoring system is operational. Other than an inspection of the Vertical Concrete Cask surface, no annual maintenance of the storage system is required. Collective dose due to design basis off-normal conditions and accident events, such as clearing the blockage of air vents, is accounted for in Chapter 11.0, and is not included in this evaluation.

Routine operations are expected to include:

- Daily electronic measurement of air outlet temperatures. Outlet temperature indicators are located away from the cask array. Remote temperature measurement is not assumed to contribute to operator dose.

- A daily security inspection of the fence and equipment surrounding the storage area. The security inspection is assumed to make no additional contribution to operator dose.

- Grounds maintenance performed every other week by 1 maintenance technician. Grounds maintenance is assumed to require 0.5 hour.

- Quarterly radiological surveillance. The surveillance consists of a radiological survey comprised of a surface radiation measurement on each cask, the determination and/or verification of general area exposure rates and radiological postings. This surveillance is assumed to require 1 hour and 1 person.

- Annual inspection of the general condition of the casks. This inspection is estimated to require 15 minutes per cask and require 2 technicians.

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11.2.15.1.5 Buckling Evaluation for Maine Yankee High Burnup Fuel Rods

This section presents the buckling evaluation for Maine Yankee high burnup fuel (burnup between 45,000 and 50,000 MWD/MTU) having cladding oxide layers that are 80 and 120 microns thick. A similar evaluation is presented in Section 11.2.15.1.6 for Maine Yankee high burnup fuel with an oxide layer thickness of 80 microns that is also mechanically damaged. These analyses show that the high burnup fuel and the damaged high burnup fuel do not buckle in the design basis accident events. An end drop orientation is considered with an acceleration of 60 g, which subjects the fuel rod to axial loading. A reduced clad thickness is assumed, due to the cladding oxide layer.

In the end drop orientation, the fuel rods are laterally restrained by the grids and may come into contact with the fuel assembly base. The only vertical constraint for the fuel rod is the base of the assembly. The weight of the fuel pellets is included as the pellets are considered to be vertically supported by the cladding. A two-dimensional model comprised of ANSYS BEAM3 elements, shown in Figure 11.2.15.1.5-1, is used for the evaluation. This evaluation is considered to be the bounding condition (as opposed to an evaluation that considers the only the fuel cladding).

80 Micron Oxide Layer Thickness Evaluation

During the end drop, the fuel rod impacts the fuel assembly base. The fuel rod itself will respond as an elastic bar under a sudden compression load at its bottom end. The duration of this impact is bounded by the first extentional mode shape of the fuel rod. Contribution of higher frequency extentional modes of the rod would tend to shorten the duration of impact of the fuel rod with the fuel assembly base. The fuel rod, upon initiation of impact, corresponds to an undeformed state. In the process of the impact, the compression of the fuel rod will increase to a maximum and then return to a near uncompressed state, at which point the time of impact has been completed. This actually represents half of a cycle of the lowest frequency mode shape of the fuel rod. The shape of the time dependence of the deformation is sinusoidal. The single extentional mode shape can also be considered to be a single degree of freedom with a corresponding mass and stiffness. In viewing such an event as a spring mass system, the time variation of the deformation during the impact is expected to be sinusoidal.

The buckling mode for the fuel rod is governed by the boundary conditions. For this configuration, the grids provide a lateral support, but no vertical support. The only vertical restraint is considered to be at the point of contact of the fuel rod and the base of the assembly. The weight of the fuel rod pellets and cladding is assumed to be uniformly distributed along the length of the fuel rod. In the

end drop, this results in the maximum compressive load occurring at the base of the fuel rod. The first buckling mode shape corresponding to these conditions is computed as shown in Figure 11.2.15.1.5-2.

Typically eigenvalue buckling is applied for static environments. For dynamic loading, it is assumed that the duration of the loading is sufficiently long to allow the system to experience the complete load, even as the deformation associated with the buckling is commenced. For dynamic loading, the lateral motion, which would correspond to the buckled shape, will correspond to the lowest mode shape. This lowest frequency mode shape is shown in Figure 11.2.15.1.5-2 and corresponds to a frequency of 25.9 Hz. The similarity of the two shapes shown in Figure 11.2.15.1.5-2 is expected, since both have the same displacement boundary conditions, the same stiffness matrix, and the same governing finite element equations, i.e.,

$$[K]\{\phi_i\} = \lambda_i\,[A]\{\phi_i\}$$

where:

$[K]$ = structure stiffness matrix
$\{\phi_i\}$ = eigenvector
λ_i = eigenvalue
$[A]$ = mass matrix for the mode shape calculation or stress stiffening matrix for the buckling evaluation

Based on the time duration of the impact and the inherent inability of the fuel rod to rapidly displace in the lateral direction, the effect of the actual lateral motion of buckling can be computed with a dynamic load factor (DLF) [47]. The expression for the DLF for a half-sine loading for a single degree of freedom is given by

$$\mathrm{DLF} = \frac{2\beta\cos(\pi/2\beta)}{1-\beta^2}$$

where:

β = ratio of the first extentional mode frequency to the first lateral mode frequency

These values, computed in this section, are $\beta = 8.32$ and DLF = 0.244.

This DLF is applied to the end drop acceleration of 60g, which is the bounding load to potentially result in the buckling of the fuel rod. The product of 60g × DLF (= 14.4g) is well below the vertical acceleration corresponding to the first buckling mode shape, 37.9g as computed in this section. This indicates that the time duration of the impact of the fuel onto the fuel assembly base is of sufficiently short nature that buckling of the fuel rod cannot occur.

An effective cross-sectional property is used in the model to consider the properties of the fuel pellet and the fuel cladding. The modulus of elasticity (EX) for the fuel pellet has a nominal value of 26.0 x 10^6 psi [48]. To be conservative, only 50 percent of this value is used in the evaluation. The EX for the fuel pellet was, therefore, taken to be 13.0 x 10^6 psi. The value of EX (10.47 x 10^6 psi) was used for the irradiated Zircaloy cladding (ISG-12). Reference information shows that there is no additional reduction of the ductility of the cladding due to extended burnup into the 45,000 – 50,000 MWD/MTU range [49].

The bounding dimensions and physical data (minimum clad thickness, maximum rod length and minimum number of support grids) for the Maine Yankee fuel rod used in the model are:

Outer diameter of cladding (inches)	0.434
Cladding thickness (inches)	0.023
Cladding density (lb/in^3)	0.237
Fuel pellet density (lb/in^3)	0.396

The cladding is reduced from its nominal value of 0.026 inches by the assumed 80 micron oxidation layer (0.003 inches) to 0.023 inches. Similarly, the fuel rod outer diameter is reduced from the nominal value of 0.44 inches to 0.434 inches.

The elevation of the grids, measured from the bottom of the fuel assembly are: 2.3, 33.0, 51.85, 70.7, 89.6, 108.4, 127.3 and 144.9 (inches).

The effective cross-sectional properties (EI_{eff}) for the beam are computed by adding the value of EI for the cladding and the pellet, where:

E = modulus of elasticity (lb/in^2)
I = cross-sectional moment of inertia (in^4)

The lowest frequency for the extentional mode shape was computed to be 218.9 Hz. The first mode shape corresponds to a frequency of 25.9 Hz. Using the expression for the DLF previously discussed, the DLF is computed to be 0.244 ($\beta = 8.32$).

120 Micron Oxide Layer Thickness Evaluation

The buckling calculation used the same model employed for the mode shape calculation. The load that would potentially buckle the fuel rod in the end drop is due to the deceleration of the rod. This loading was implemented by applying a 1g acceleration in the direction that would result in compressive loading of the fuel rod. The acceleration required to buckle the fuel rod is computed to be 37.3g. This acceleration is much higher than the effective g-load of 14.3g corresponding to the end drop. Therefore, the fuel rods do not buckle during a 60g end drop.

Using the same fuel rod model, the acceleration required to buckle the fuel rods is found to be 37.3 g, which is much higher than the calculated effective g-load (14.3 g) due to the 60 g end drop. Therefore, the fuel rods with a 120 micron cladding oxide layer do not buckle in the 60 g end drop event.

Figure 11.2.15.1.5-1 Two-Dimensional Beam Finite Element Model for Maine Yankee Fuel Rod

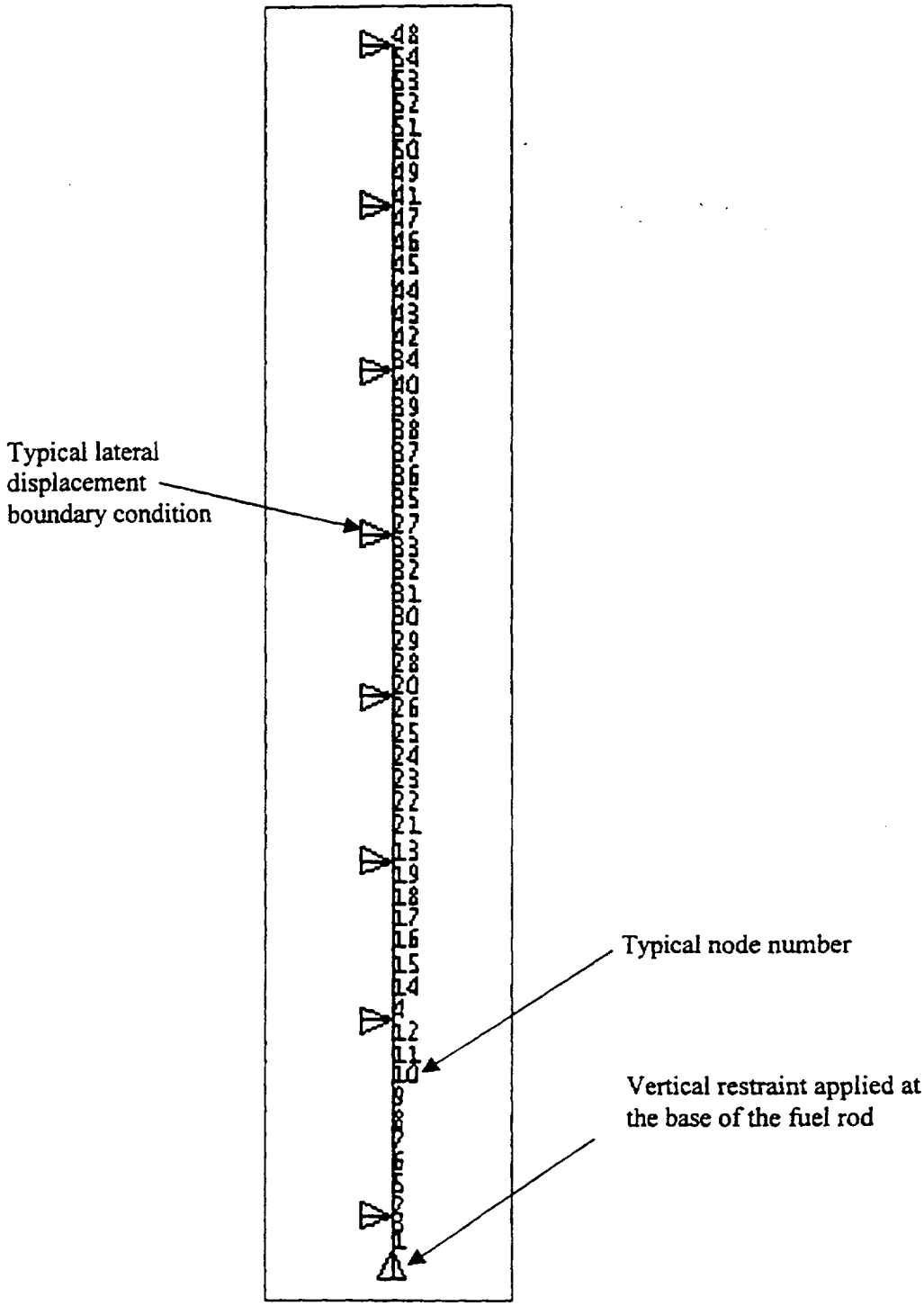
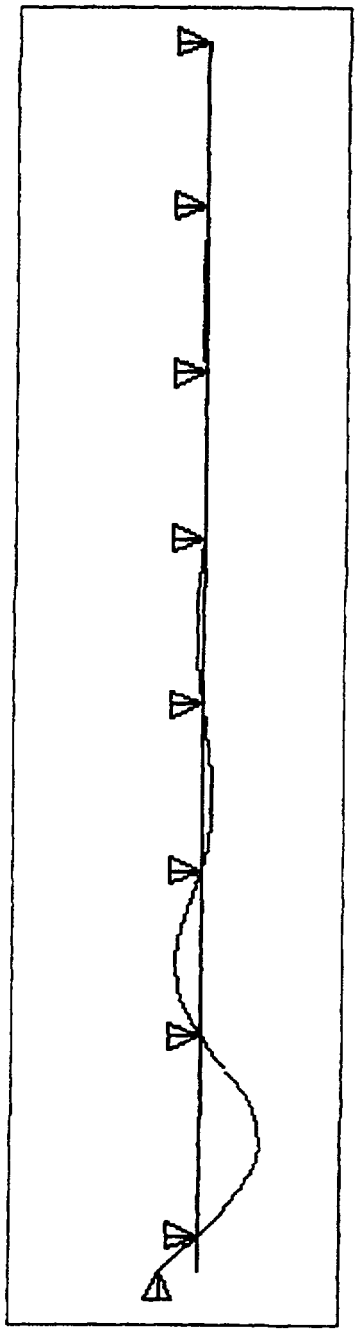
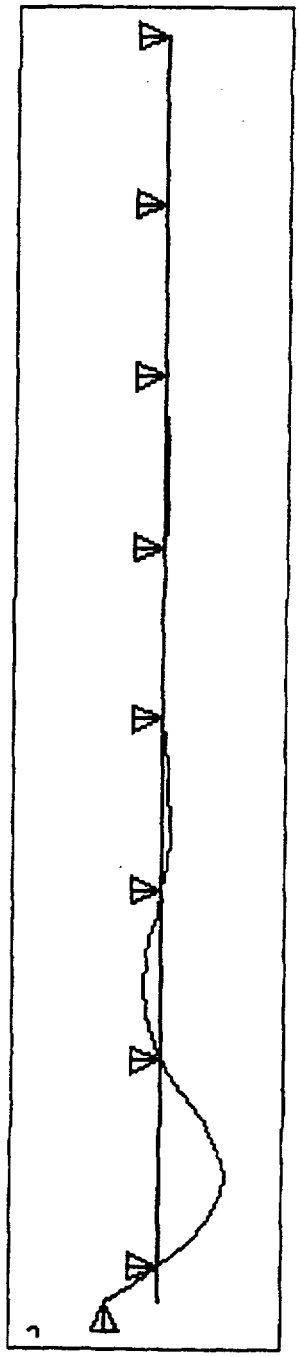


Figure 11.2.15.1.5-2 Mode Shape and First Buckling Shape for the Maine Yankee Fuel Rod

First Lateral Dynamic
Mode Shape at 25.9 Hz



First Buckling
Shape at 37.9g



11.2.15.1.6 Buckling Evaluation for High Burnup Fuel with Mechanical Damage

This section presents the buckling evaluation for high burnup fuel having an 80 micron cladding oxide layer thickness and with mechanical damage consisting of one or more missing support grids up to an unsupported fuel rod length of 60 inches.

End Drop Evaluation

The buckling load is maximized at the bottom of the fuel assembly. The bounding evaluation is the removal of the grid strap that maximizes the spacing at the lowest vertical elevation. The elevations of the grids in the model, measured from the bottom of the fuel assembly are: 2.3, 51.85, 70.7, 89.6, 108.4, 127.3 and 144.9 inches (Figure 11.2.15.1.6-1). The grid at the 33.0-inch elevation is removed, resulting in a grid spacing of approximately 50.0 inches. The grid located at 51.85 inches is conservatively assumed to be located at 62.3 inches, resulting in an unsupported rod length of 60.0 inches.

The case of the missing grid is evaluated using the methodology presented in Section 11.2.15.1.5 for the fuel assembly with all the grids being present. The dimensions and physical data for the Maine Yankee fuel rod used in the model are:

Outer diameter of cladding (inches)	0.434
Cladding thickness (inches)	0.023
Cladding density (lb/in^3)	0.237
Fuel pellet density (lb/in^3)	0.396
Fuel pellet Modulus of Elasticity (psi)	13.0×10^6
Zircaloy cladding Modulus of Elasticity (psi)	10.47×10^6

The cladding is reduced from its nominal value of 0.026 inches by the assumed 80 micron oxidation layer thickness (0.003 inches) to 0.023 inches. Similarly, the fuel rod outer diameter is reduced from the nominal value of 0.44 inches to 0.434 inches. The fuel pellet modulus of elasticity is conservatively reduced 50%. The modulus of elasticity of the Zircaloy cladding is taken from ISG-12 [50].

With the grid missing, the frequency of the fundamental lateral mode shape is 7.8 Hz. The natural frequency of the fundamental extensional mode was determined to be 218.9 Hz. The DLF is computed to be 0.072, resulting in an effective acceleration of 0.072 x 60 = 4.3g. Using the same method to compute the acceleration at which buckling occurs, the lowest buckling acceleration is 14.4g, which is significantly greater than 4.3g. Therefore, the fuel rod does not buckle during an

end drop. Figures 11.2.15.1.6-1 and 11.2.15.1.6-2 show the finite element model and buckling results and mode shape.

Side Drop Evaluation

The Maine Yankee fuel rod is evaluated for a 60 g side drop with a missing support grid in the fuel assembly. Using the same assumptions as for the end drop evaluation, the span between support grids is assumed to be 60.0 inches.

For this analysis, the dimensions and physical data used are:

Fuel rod OD	0.434 in. (80 micron oxidation layer)
Clad ID	0.388 in.
E_{clad}	10.47E6 psi
E_{fuel}	13.0E6 psi
Clad density	0.237 lb/in^3
Fuel density	0.396 lb/in^3
A_{clad}	0.030 in^2 (cross-sectional area)
A_{fuel}	0.118 in^2 (cross-sectional area)

The mass of the fuel rod per unit length is:

$$m = \frac{0.396(0.122) + 0.237(0.030)}{386.4} = 0.000143 \text{ lb-s}^2/\text{in}^2$$

For the fuel rod, the product of the Modulus of Elasticity (E) and Moment of Inertia (I), is:

$$EI_{clad} = 10.47E6 \frac{\pi(0.22^4 - 0.197^4)}{4} = 6{,}878 \text{ lb-in}^2$$

$$EI_{fuel} = 13.0E6 \frac{\pi(0.197^4)}{4} = 15{,}378 \text{ lb-in}^2$$

$$EI = 6878 + 15378 = 22{,}256 \text{ lb-in}^2$$

During a side drop, the maximum deflection of a fuel rod is based on the fuel rod spacing of the fuel assembly. The pitch (center-to-center spacing) of fuel rods is 0.58 inches [51]. The maximum pitch is across the diagonal of the fuel assembly. The maximum pitch is:

$$dp = \frac{0.58}{\sin 45} = 0.82 \text{ in.}$$

The maximum deflection of a fuel rod is at the top of the fuel assembly and the minimum deflection is at the bottom of the fuel assembly.

Assuming a 17 x 17 array (which envelopes the Maine Yankee 14 x 14 array), the maximum fuel rod deflection is:

(17-1) x (0.82-0.43) = 6.18 in.

The deflection of a simply supported beam with a distributed load is given by the equation:

$$\Delta = \frac{5\omega l^4}{384EI} = \frac{5(g\omega)l^4}{384(EI_{total})} \qquad [52]$$

$$g = \frac{384\Delta(EI_{total})}{5\omega l^4}$$

The cladding bending stress is given by the equation:

$$S = \frac{Mc}{I} = \frac{\left(\frac{(g\omega l^2)}{8}\right)c}{I_{clad}}\left(\frac{EI_{clad}}{EI_{total}}\right)$$

Inserting the equation for 'g':

$$S = \frac{384\Delta c E_{clad}}{40 \times L^2}$$

where:

c = 0.217 inch distance from center of fuel rod to extreme outer fiber
L = 60 inches (the unsupported fuel rod length)
Δ = 6.18 inches (the maximum deflection)

The bending stress in the fuel rod is:

$$S = \frac{384 \times 6.18 \times 0.217 \times 10.47E6}{40(60)^2} = 37.4\,\text{ksi}$$

The maximum hoop stress due to the fuel rod internal pressure is determined to be 19.1 ksi (131.4 MPa per Tables 4.4.7-3 and 4.5.1.2-1). Therefore, the maximum axial stress is 9.6 ksi (one half of the hoop stress [53]).

The bearing stress between two fuel rods under a 60 g load is:

$$S_{brg} = 0.591\sqrt{\frac{\omega E}{K_D}} = 0.591\sqrt{\frac{(0.000143 \times 386.4) \times 60 \times 10.47E6}{0.22}} = 7.4\,\text{ksi} \quad [53]$$

where:

$$K_D = \frac{D_1 D_2}{D_1 + D_2} = \frac{0.434 \times 0.434}{0.434 + 0.434} = 0.22$$

The total stress is:

$$S = 37.4 + 9.6 + 7.4 = 54.4\,\text{ksi}$$

The ultimate strength allowable for irradiated Zircaloy –4 is 83.4 ksi (Figure 3-2 [54]). Therefore, the margin of safety for ultimate strength is:

$$MS = \frac{83.4}{54.4} - 1 = 0.53$$

The yield strength allowable for irradiated Zircaloy –4 is 78.3 ksi (Figure 3-2 [54]). Therefore, the margin of safety for yield strength is:

$$MS = \frac{78.3}{54.4} - 1 = 0.44$$

The maximum bearing stress occurs between the bottom fuel rod and the fuel tube. The bearing stress is:

$$S_{brg} = 0.591\sqrt{\frac{17 \times 0.000143 \times 386.4 \times 60 \times 10.47E6}{0.44}} = 21.6\,\text{ksi}$$

The bending stress is negligible because the maximum deflection is equal to the spacing of the fuel rods established by the grid. Therefore, the top fuel rod is bounding.

Consequently, the fuel rods are demonstrated to be structurally adequate for the 60g side drop loading condition.

Figure 11.2.15.1.6-1 Two-Dimensional Beam Finite Element Model for a Fuel Rod with a Missing Grid

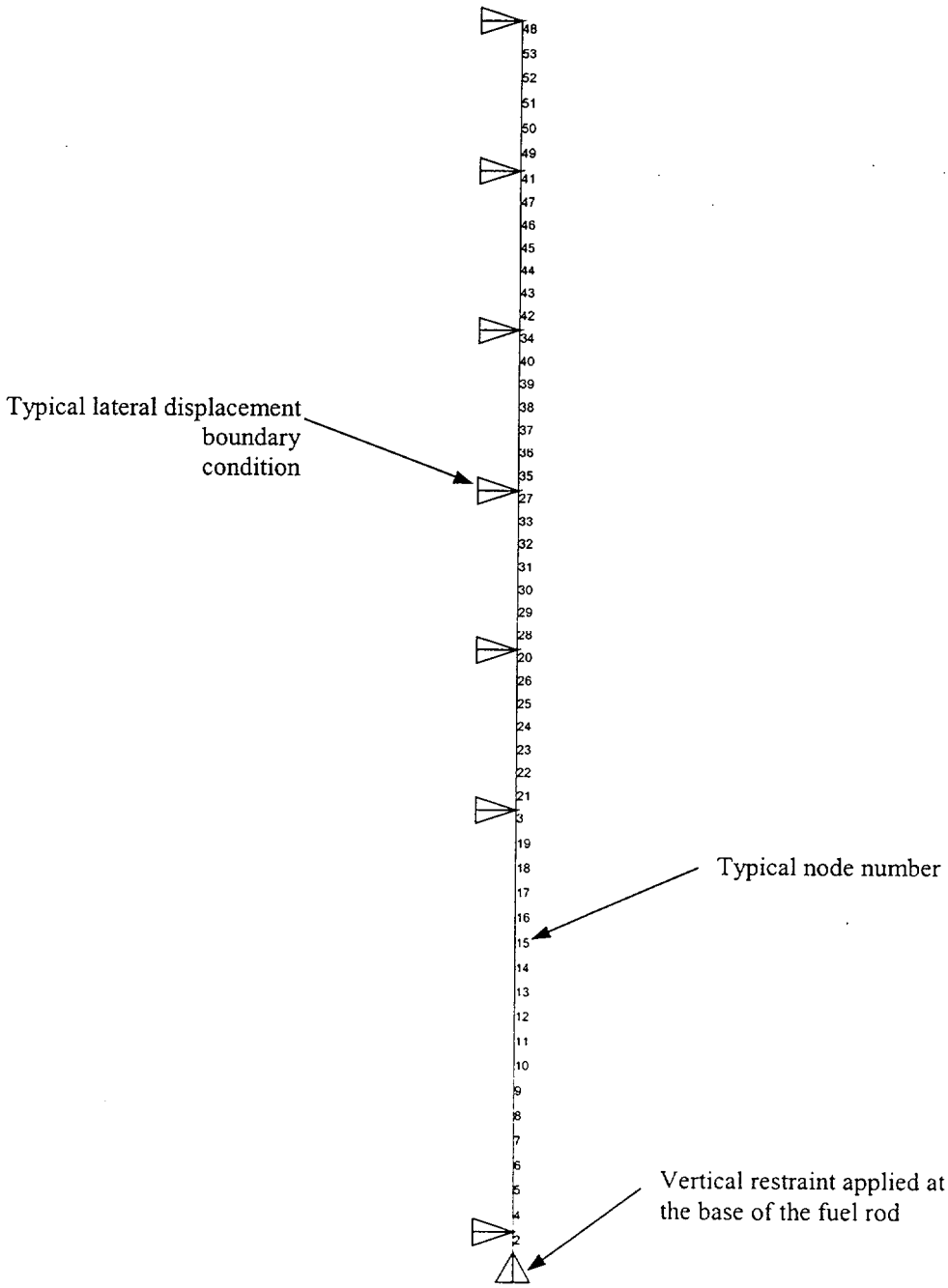
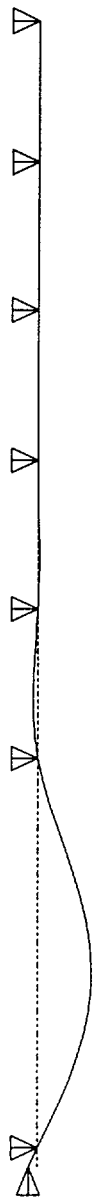
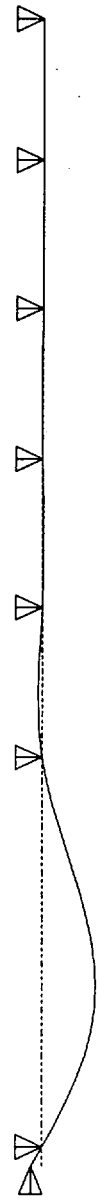


Figure 11.2.15.1.6-2 Modal Shape and First Buckling Mode Shape for a Fuel Rod with a Missing Grid

First Lateral Dynamic Mode Shape at 7.8 Hz



First Buckling Mode Shape at 14.4g



11.2.15.1.7 Fuel Debris from 24 Damaged Fuel Rods

Each canister may contain up to a total of 24 fuel rods integral to one or more fuel assemblies that are classified as damaged, and which are loaded into the canister as intact fuel. It is assumed that the fuel contents of the 24 fuel rods are allowed to escape the fuel cladding as "debris" and the debris is dispersed inside of the canister. Since the canister is in the vertical position for normal conditions of storage, the debris is considered to be located at the bottom of the canister. The total weight of the fuel debris from 24 fuel rods is less than 170 pounds. Therefore, the fuel debris is considered to have a negligible effect on the structural performance of the basket or canister. In a concrete cask tip-over event, the basket component near the canister bottom experiences minimal g-load. Therefore, the fuel debris is considered to have negligible effect on the structural performance of the basket or canister in the tip-over event.

APPENDIX 12A

TECHNICAL SPECIFICATIONS FOR THE NAC-UMS® SYSTEM

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STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI, while an NAC-UMS® SYSTEM containing spent fuel is located on the storage pad within the ISFSI perimeter.
TRANSFER CASK	TRANSFER CASK is a shielded lifting device that holds the CANISTER during LOADING and UNLOADING OPERATIONS and during closure welding, vacuum drying, leak testing, and non-destructive examination of the CANISTER closure welds. The TRANSFER CASK is also used to transfer the CANISTER into and from the CONCRETE CASK and into the transport cask.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities involved in moving a loaded NAC-UMS® CONCRETE CASK and CANISTER to and from the ISFSI. TRANSPORT OPERATIONS begin when the NAC-UMS® SYSTEM is first secured on the transporter and end when the NAC-UMS® SYSTEM is at its destination and no longer secured on the transporter.
TRANSPORTABLE STORAGE CANISTER (CANISTER)	TRANSPORTABLE STORAGE CANISTER is the sealed container that consists of a tube and disk fuel basket in a cylindrical canister shell that is welded to a baseplate, shield lid with welded port covers, and structural lid. The CANISTER provides the confinement boundary for the confined spent fuel.
TRANSFER OPERATIONS	TRANSFER OPERATIONS include all licensed activities involved in transferring a loaded CANISTER from a CONCRETE CASK to another CONCRETE CASK or to a TRANSPORT CASK.

(continued)

UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on a NAC-UMS® SYSTEM to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the NAC-UMS® SYSTEM is no longer secured on the transporter and end when the last fuel assembly is removed from the NAC-UMS® SYSTEM.
VERTICAL CONCRETE CASK (CONCRETE CASK)	VERTICAL CONCRETE CASK is the cask that receives and holds the sealed CANISTER. It provides the gamma and neutron shielding and convective cooling of the spent fuel confined in the CANISTER.
STANDARD FUEL	Irradiated fuel assemblies having the same configuration as when originally fabricated consisting generally of the end fittings, fuel rods, guide tubes, and integral hardware. For BWR fuel, the channel is considered to be integral hardware. The design basis fuel characteristics and analysis are based on the STANDARD FUEL configuration.
DAMAGED FUEL	A fuel assembly or fuel rod with known or suspected cladding defects greater than pinhole leaks or hairline cracks.

(continued)

HIGH BURNUP FUEL	A fuel assembly having a burnup between 45,000 and 50,000 MWD/MTU, which must be preferentially loaded in periphery positions of the basket. An intact HIGH BURNUP FUEL assembly in which no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and in which no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, as determined by measurement and statistical analysis, may be stored as INTACT FUEL. HIGH BURNUP FUEL assemblies not meeting the cladding oxide thickness criteria for INTACT FUEL or that have an oxide layer that has become detached or spalled from the cladding are classified as DAMAGED FUEL.
FUEL DEBRIS	An intact or a partial fuel rod or an individual intact or partial fuel pellet not contained in a fuel rod. Fuel debris is inserted into a 9 x 9 array of tubes in a lattice that has approximately the same dimensions as a standard fuel assembly.
CONSOLIDATED FUEL	A nonstandard fuel configuration in which the individual fuel rods from one or more fuel assemblies are placed in a single container or a lattice structure that is similar to a fuel assembly. CONSOLIDATED FUEL is stored in a MAINE YANKEE FUEL CAN.

(continued)

SITE SPECIFIC FUEL	Spent fuel configurations that are unique to a site or reactor due to the addition of other components or reconfiguration of the fuel assembly at the site. It includes fuel assemblies, which hold nonfuel-bearing components, such as control components or instrument and plug thimbles, or which are modified as required by expediency in reactor operations, research and development or testing. Modification may consist of individual fuel rod removal, fuel rod replacement of similar or dissimilar material or enrichment, the installation, removal or replacement of burnable poison rods, or containerizing damaged fuel. Site specific fuel includes irradiated fuel assemblies designed with variable enrichments and/or axial blankets, fuel that is consolidated and fuel that exceeds design basis fuel parameters.
MAINE YANKEE FUEL CAN	A specially designed stainless steel screened can sized to hold INTACT FUEL, CONSOLIDATED FUEL or DAMAGED FUEL. The screens preclude the release of gross particulate from the can into the canister cavity. The MAINE YANKEE FUEL CAN may be loaded only in a Class 1 canister.
UNCANNED DAMAGED FUEL	Fuel classified as DAMAGED FUEL that is not in a MAINE YANKEE FUEL CAN. Up to 24 fuel rods classified as damaged (integral to one or more fuel assemblies) may be loaded in each CANISTER using one or more periphery loading positions.

A 3.1 NAC-UMS® SYSTEM

A 3.1.6 CONCRETE CASK Heat Removal System

LCO 3.1.6 The CONCRETE CASK Heat Removal System shall be OPERABLE.

APPLICABILITY: During STORAGE OPERATIONS

ACTIONS

----------NOTE----------

Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CONCRETE CASK Heat Removal System inoperable	A.1 Restore CONCRETE CASK Heat Removal System to OPERABLE status	8 hours
B. Required Action A.1 and associated Completion Time not met	B.1 Perform SR 3.1.6.1 AND B.2.1 Restore CONCRETE CASK Heat Removal System to OPERABLE status	Immediately and every 6 hours thereafter 12 hours

(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.6.1	Verify the difference between the average CONCRETE CASK air outlet temperature and ISFSI ambient temperature is ≤ 102°F (for the PWR CANISTER) and ≤ 92°F (for the BWR CANISTER)	24 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.1.1	Verify that the removable contamination on the accessible exterior surfaces of the CANISTER is within limits	Once, prior to TRANSPORT OPERATIONS
SR 3.2.1.2	Verify that the removable contamination on the accessible interior surfaces of the TRANSFER CASK do not exceed limits	Once, prior to TRANSPORT OPERATIONS

A 3.2 NAC-UMS® SYSTEM Radiation Protection

A 3.2.2 CONCRETE CASK Average Surface Dose Rates

LCO 3.2.2 The average surface dose rates of each CONCRETE CASK shall not exceed the following limits unless required ACTIONS A.1 and A.2 are met.

a. 50 mrem/hour (neutron + gamma) on the side (on the concrete surfaces);

b. 50 mrem/hour (neutron + gamma) on the top;

c. 100 mrem/hour (neutron + gamma) at air inlets and outlets.

APPLICABILITY: During LOADING OPERATIONS

ACTIONS

----------NOTE----------
Separate Condition entry is allowed for each NAC-UMS® SYSTEM.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. CONCRETE CASK average surface dose rate limits not met	A.1 Administratively verify correct fuel loading AND	24 hours

(continued)

A 5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

A 5.1 Training Program

A training program for the NAC-UMS® Universal Storage System shall be developed under the general licensee's systematic approach to training (SAT). Training modules shall include comprehensive instructions for the operation and maintenance of the NAC-UMS® Universal Storage System and the independent spent fuel storage installation (ISFSI).

A 5.2 Pre-Operational Testing and Training Exercises

A dry run training exercise on loading, closure, handling, unloading, and transfer of the NAC-UMS® Storage System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the CANISTER. The dry run may be performed in an alternate step sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:

a. Moving the CONCRETE CASK into its designated loading area
b. Moving the TRANSFER CASK containing the empty CANISTER into the spent fuel pool
c. Loading one or more dummy fuel assemblies into the CANISTER, including independent verification
d. Selection and verification of fuel assemblies requiring preferential loading
e. Installing the shield lid
f. Removal of the TRANSFER CASK from the spent fuel pool
g. Closing and sealing of the CANISTER to demonstrate pressure testing, vacuum drying, helium backfilling, welding, weld inspection and documentation, and leak testing
h. TRANSFER CASK movement through the designated load path
i. TRANSFER CASK installation on the CONCRETE CASK
j. Transfer of the CANISTER to the CONCRETE CASK

(continued)

A 5.2 Pre-Operational Testing and Training Exercises (continued)

k. CONCRETE CASK shield plug and lid installation
l. Transport of the CONCRETE CASK to the ISFSI
m. CANISTER unloading, including reflooding and weld removal or cutting
n. CANISTER removal from the CONCRETE CASK

Appropriate mockup fixtures may be used to demonstrate and/or to qualify procedures, processes or personnel in welding, weld inspection, vacuum drying, helium backfilling, leak testing and weld removal or cutting.

A 5.3 Surveillance After an Off-Normal, Accident, or Natural Phenomena Event

A Response Surveillance is required following off-normal, accident or natural phenomena events. The NAC-UMS® SYSTEMs in use at an ISFSI shall be inspected within 4 hours after the occurrence of an off-normal, accident or natural phenomena event in the area of the ISFSI. This inspection must specifically verify that all the CONCRETE CASK inlets and outlets are not blocked or obstructed. At least one-half of the inlets and outlets on each CONCRETE CASK must be cleared of blockage or debris within 24 hours to restore air circulation.

The CONCRETE CASK and CANISTER shall be inspected if they experience a drop or a tipover.

A 5.4 Radioactive Effluent Control Program

The program implements the requirements of 10 CFR 72.44(d).

a. The NAC-UMS® SYSTEM does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. LCO 3.1.5, CANISTER Helium Leak Rate, provides assurance that there are no radioactive effluents from the NAC-UMS® SYSTEM.

(continued)

A 5.4 Radioactive Effluent Control Program (continued)

b. This program includes an environmental monitoring program. Each general license user may incorporate NAC-UMS® SYSTEM operations into their environmental monitoring program for 10 CFR Part 50 operations.

c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

A 5.5 NAC-UMS® SYSTEM Transport Evaluation Program

This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded TRANSFER CASK or CONCRETE CASK using devices, which are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable when the TRANSFER CASK or CONCRETE CASK is in the fuel building or is being handled by a device providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc.).

Pursuant to 10 CFR 72.212, this program shall evaluate the site specific transport route conditions.

a. The lift height above the transport surface prescribed in Section B3.4.6 of Appendix B to Certificate of Compliance (CoC) No. 1015 shall not exceed the limits in Table 12A5-1. Also, the program shall ensure that the transport route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than those prescribed for the reference pad surface which forms the basis for the values cited in Section B3.4.6 of Appendix B to CoC No. 1015.

b. For site specific transport conditions which are not bounded by the surface characteristics in Section B3.4.6 of Appendix B to CoC No. 1015, the program may evaluate the site specific conditions to ensure that the impact loading due to design basis drop events does not exceed 60g. This alternative analysis shall be commensurate with the drop analyses described in the Safety Analysis Report for the NAC-UMS® SYSTEM. The program shall ensure that these alternative analyses are documented and controlled.

(continued)

A 5.5 NAC-UMS® SYSTEM Transport Evaluation Program (continued)

c. The TRANSFER CASK and CONCRETE CASK may be lifted to those heights necessary to perform cask handling operations, including CANISTER transfer, provided the lifts are made with structures and components designed in accordance with the criteria specified in Section B3.5 of Appendix B to CoC No. 1015, as applicable.

A 5.6 Verification of Oxide Layer Thickness on High Burnup Fuel

A verification program is required to determine the oxide layer thickness on high burnup fuel by measurement or by statistical analysis. A fuel assembly having a burnup between 45,000 MWD/MTU and 50,000 MWD/MTU is classified as high burnup. The verification program shall be capable of classifying high burnup fuel as INTACT FUEL or DAMAGED FUEL based on the following criteria:

1. A HIGH BURNUP FUEL assembly may be stored as INTACT FUEL provided that no more than 1% of the fuel rods in the assembly have a peak cladding oxide thickness greater than 80 microns, and that no more than 3% of the fuel rods in the assembly have a peak oxide layer thickness greater than 70 microns, and that the fuel assembly is otherwise INTACT FUEL.

2. A HIGH BURNUP FUEL assembly not meeting the cladding oxide thickness criteria for INTACT FUEL or that has an oxide layer that is detached or spalled from the cladding is classified as DAMAGED FUEL.

3. A HIGH BURNUP FUEL assembly classified as DAMAGED FUEL may be stored as INTACT FUEL provided that the number of DAMAGED FUEL rods is considered in the CANISTER limit of 24 total DAMAGED FUEL rods, and that the 24 total DAMAGED FUEL rod limit for the CANISTER is not exceeded.

A fuel assembly, having a burnup between 45,000 and 50,000 MWD/MTU, must be preferentially loaded in periphery positions of the basket.

Table 12A5-1 TRANSFER CASK and CONCRETE CASK Lifting Requirements

Item	Orientation	Lifting Height Limit
TRANSFER CASK	Horizontal	None Established
TRANSFER CASK	Vertical	None Established[1]
CONCRETE CASK	Horizontal	Not Permitted
CONCRETE CASK	Vertical	< 24 inches

Note:

1. See Technical Specification A5.5(c).

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APPENDIX 12B

APPROVED CONTENTS AND DESIGN FEATURES FOR THE NAC-UMS® SYSTEM

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For the PWR fuel basket configuration, shown in Figure 12B2-1, fuel positions are numbered using the drain line as the reference point. Fuel positions 9, 10, 15 and 16 are considered to be basket center positions for the purpose of meeting the preferential loading requirement. The fuel with the shortest cooling times from among the fuel designated for loading in the CANISTER will be placed in the center positions. A single fuel assembly having the shortest cooling time may be loaded in any of these four positions. Fuel positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23 and 24 are periphery positions, where fuel with the longest cooling times will be placed. Fuel with the longest cooling times may be loaded in any of these 12 positions. Similarly, designated fuel assemblies with cooling times in the midrange of the shortest and longest cooling times will be loaded in the intermediate fuel positions – 4, 5, 8, 11, 14, 17, 20 and 21.

For the BWR fuel basket configuration, shown in Figure 12B2-2, fuel positions are also numbered using the drain line as the reference point. Fuel positions 23, 24, 25, 32, 33 and 34 are considered to be basket center positions for the purpose of meeting the preferential loading requirement. The fuel with the shortest cooling times from among the fuel designated for loading in the CANISTER will be placed in the center positions. However, the single fuel assembly having the shortest cooling time will be loaded in either position 24 or position 33. Fuel positions 1, 2, 3, 4, 5, 6, 12, 13, 19, 20, 28, 29, 37, 38, 44, 45, 51, 52, 53, 54, 55 and 56 are periphery positions, where fuel with the longest cooling times will be placed. Fuel with the longest cooling times may be loaded in any of these 23 positions. Designated fuel assemblies with cooling times in the midrange of the shortest and longest cooling times will be divided into two tiers. The fuel assemblies with the shorter cooling times in the midrange will be loaded in the inner intermediate fuel positions - 15, 16, 17, 22, 26, 31, 35, 40, 41, and 42. Fuel assemblies with the longer cooling times in the midrange will be loaded in the outer intermediate fuel positions - 7, 8, 9, 10, 11, 14, 18, 21, 27, 30, 36, 39, 43, 46, 47, 48, 49 and 50.

These loading patterns result in the placement of fuel such that the shortest-cooled fuel is in the center of the basket and the longest-cooled fuel is on the periphery. Based on engineering evaluations, this loading pattern ensures that fuel assembly allowable cladding temperatures are satisfied.

(continued)

B 2.1.3 Maine Yankee SITE SPECIFIC FUEL Preferential Loading

The estimated Maine Yankee SITE SPECIFIC FUEL inventory is shown in Table 12B2-6. As shown in this table, certain of the Maine Yankee fuel configurations must be preferentially loaded in specific basket fuel tube positions.

Corner positions are used for CONSOLIDATED FUEL, certain HIGH BURNUP FUEL and DAMAGED FUEL or FUEL DEBRIS loaded in a MAINE YANKEE FUEL CAN, for fuel assemblies with missing fuel rods or fuel assemblies with fuel rods that have been replaced by hollow Zircaloy rods. Designation for placement in corner positions results primarily from shielding or criticality evaluations of these fuel configurations. CONSOLIDATED FUEL is conservatively designated for a corner position, even though analysis shows that these lattices could be loaded in any basket position. Corner positions are positions 3, 6, 19, and 22 in Figure 12B2-1.

Preferential loading is also used for HIGH BURNUP fuel not loaded in the MAINE YANKEE FUEL CAN. This fuel is assigned to peripheral locations, positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24 in Figure 12B2-1. The interior locations must be loaded with fuel that has lower burnup and/or longer cool times to maintain the design basis heat load and component temperature limits for the basket and canister, and the spent fuel short-term temperature limits, as described in Section B 2.1.2.

One of the three loading patterns (Standard, 1.05 kW (periphery), or 0.958 kW (periphery)) shown in Table 12B2-8 must be used to load each canister. Once selected, all of the spent fuel in that canister must be loaded in accordance with that pattern. Within a pattern, mixing of enrichment and cool time is allowed, but no mixing of loading patterns is permitted. For example, choosing a Perf (1.05) pattern restricts the interior fuel to the cool times shown in the Perf (1.05i) column, and the peripherial fuel to the cool times shown in the Perf (1.05p) column.

Fuel assemblies with a control element inserted will be loaded in a Class 2 canister and basket due to the increased length of the assembly with the control element installed. However, these assemblies are not restricted as to loading position within the basket. Fuel assemblies with non-fuel items installed in corner guide tubes of the fuel assembly must also have a CEA flow plug installed and must be loaded in a basket corner fuel position in a Class 2 canister.

(continued)

Fuel assemblies with a control element assembly (CEA) inserted will be loaded in a Class 2 canister and basket due to the increased length of the assembly with the CEA installed. However, these assemblies are not restricted as to loading position within the basket.

The Transportable Storage Canister loading procedures indicates that loading of a fuel configuration with removed fuel or poison rods, CONSOLIDATED FUEL, or a MAINE YANKEE FUEL CAN with HIGH BURNUP FUEL, DAMAGED FUEL, HIGH BURNUP FUEL or UNCANNED DAMAGED FUEL, is administratively controlled in accordance with Section B 2.1.

Figure 12B2-1 PWR Basket Fuel Loading Positions

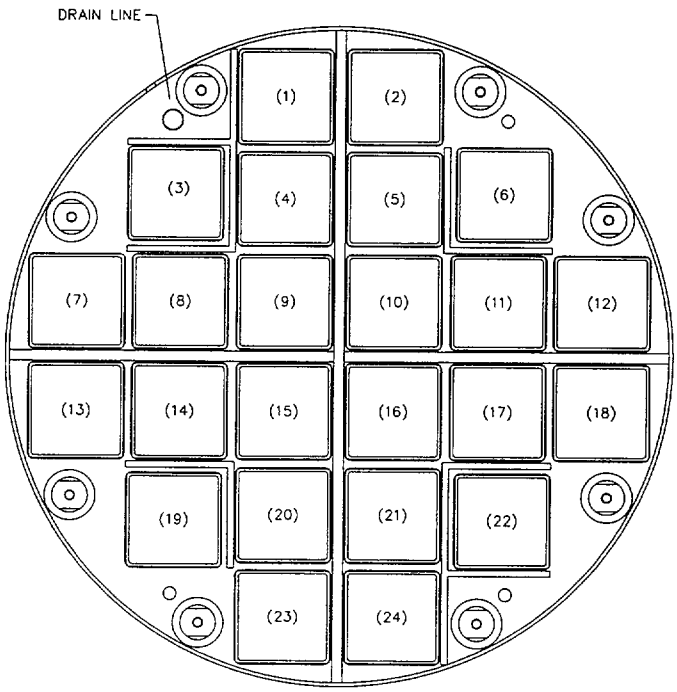

Figure 12B2-2 BWR Basket Fuel Loading Positions

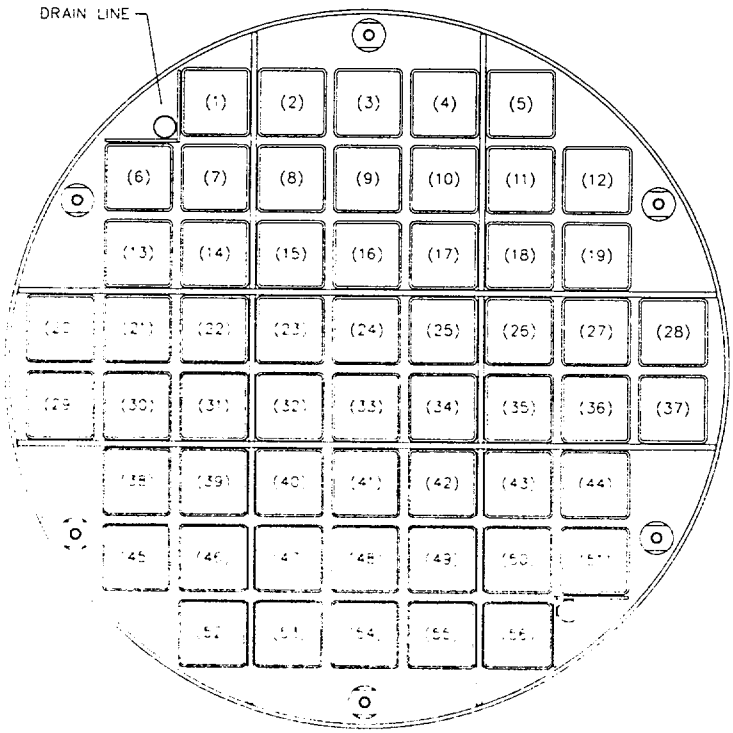

Table 12B2-1

Fuel Assembly Limits

I. NAC-UMS® CANISTER: PWR FUEL

A. Allowable Contents

1. Uranium oxide PWR INTACT FUEL ASSEMBLIES listed in Table 12B2-2 and meeting the following specifications:

a.	Cladding Type:	Zircaloy with thickness as specified in Table 12B2-2 for the applicable fuel assembly class
b.	Enrichment:	Maximum and minimum enrichments are 4.2 and 1.9 wt % ^{235}U, respectively. Fuel enrichment, burnup and cool time are related as shown in Table 12B2-4.
c.	Decay Heat Per Assembly:	≤ 958.3 watts
d.	Post-irradiation Cooling Time and Average Burnup Per Assembly:	As specified in Table 12B2-4
e.	Nominal Fresh Fuel Assembly Length (in.):	≤ 178.3
f.	Nominal Fresh Fuel Assembly Width (in.):	≤ 8.54
g.	Fuel Assembly Weight (lbs.):	≤ 1,515

B. Quantity per CANISTER: Up to 24 PWR INTACT FUEL ASSEMBLIES.

C. PWR INTACT FUEL ASSEMBLIES may contain thimble plugs and burnable poison inserts (Class 1 and Class 2 contents).

D. PWR INTACT FUEL ASSEMBLIES shall not contain control components.

E. Stainless steel spacers may be used in CANISTERS to axially position PWR INTACT FUEL ASSEMBLIES that are shorter than the available cavity length to facilitate handling.

F. Unenriched fuel assemblies are not authorized for loading.

G. The minimum length of the PWR INTACT FUEL ASSEMBLY internal structure and bottom end fitting and/or spacers shall ensure that the minimum distance to the fuel region from the base of the CANISTER is 3.2 inches.

H. PWR INTACT FUEL ASSEMBLIES with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches. End fitting damage including damaged or missing hold-down springs is allowed, as long as the assembly can be handled safely by normal means.

Table 12B2-1
Fuel Assembly Limits (continued)

II. NAC-UMS® CANISTER: BWR FUEL

A. Allowable Contents

1. Uranium oxide BWR INTACT FUEL ASSEMBLIES listed in Table 12B2-3 and meeting the following specifications:

a. Cladding Type:	Zircaloy with thickness as specified in Table 12B2-3 for the applicable fuel assembly class.
b. Enrichment:	Maximum and minimum INITIAL PEAK PLANAR-AVERAGE ENRICHMENTS are 4.0 and 1.9 wt % ^{235}U, respectively. Fuel enrichment, burnup and cooling time are related as shown in Table 12B2-5.
c. Decay Heat per Assembly:	$\leq$ 410.7 watts
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	As specified in Table 12B2-5 and for the applicable fuel assembly class.
e. Nominal Fresh Fuel Design Assembly Length (in.):	$\leq$ 176.1
f. Nominal Fresh Fuel Design Assembly Width (in.):	$\leq$ 5.51
g. Fuel Assembly Weight (lbs):	$\leq$ 683, including channels

Table 12B2-1
Fuel Assembly Limits (continued)

B. Quantity per CANISTER: Up to 56 BWR INTACT FUEL ASSEMBLIES
C. BWR INTACT FUEL ASSEMBLIES can be unchanneled or channeled with Zircaloy channels.
D. BWR INTACT FUEL ASSEMBLIES with stainless steel channels shall not be loaded.
E. Stainless steel fuel spacers may be used in CANISTERS to axially position BWR INTACT FUEL ASSEMBLIES that are shorter than the available cavity length to facilitate handling.
F. Unenriched fuel assemblies are not authorized for loading.
G. The minimum length of the BWR INTACT FUEL ASSEMBLY internal structure and bottom end fitting and/or spacers shall ensure that the minimum distance to the fuel region from the base of the CANISTER is 6.2 inches.

Table 12B2-2 PWR Fuel Assembly Characteristics

Fuel Class[1]	Vendor[2]	Array	Max. MTU	No of Fuel Rods	Max. Pitch (in)	Min. Rod Dia. (in)	Min. Clad Thick (in)	Max. Pellet Dia.(in)	Max. Active Length (in)	Min. Guide Tube Thick (in)
1	CE	14x14	0.404	176[4]	0.590	0.438	0.024	0.380	137.0	0.034
1	Ex/ANF	14x14	0.369	179	0.556	0.424	0.030	0.351	142.0	0.034
1	WE	14x14	0.362	179	0.556	0.400	0.024	0.345	144.0	0.034
1	WE	14x14	0.415	179	0.556	0.422	0.022	0.368	145.2	0.034
1	WE, Ex/ANF	15x15	0.465	204	0.563	0.422	0.024	0.366	144.0	0.015
1	Ex/ANF	17x17	0.413	264	0.496	0.360	0.025	0.303	144.0	0.016
1	WE	17x17	0.468	264	0.496	0.374	0.022	0.323	144.0	0.016
1	WE	17x17	0.429	264	0.496	0.360	0.022	0.309	144.0	0.016
2	B&W	15x15	0.481	208	0.568	0.430	0.026	0.369	144.0	0.016
2	B&W	17x17	0.466	264	0.502	0.379	0.024	0.324	143.0	0.017
3	CE	16x16	0.442	236[4]	0.506	0.382	0.025	0.325	150.0	0.035
1	Ex/ANF[3]	14x14	0.375	179	0.556	0.417	0.030	0.351	144.0	0.036
1	CE[3]	15x15	0.432	216	0.550	0.418	0.026	0.358	132.0	----
1	Ex/ANF[3]	15x15	0.431	216	0.550	0.417	0.030	0.358	131.8	----
1	CE[3]	16x16	0.403	236[4]	0.506	0.382	0.023	0.3255	136.7	0.035

1. Maximum Initial Enrichment: 4.2 wt % ^{235}U. All fuel rods are Zircaloy clad.
2. Vendor ID indicates the source of assembly base parameters, which are nominal, pre-irradiation values. Loading of assemblies meeting above limits is not restricted to the vendor(s) listed.
3. 14x14, 15x15 and 16x16 fuel manufactured for Prairie Island, Palisades and St. Lucie 2 cores, respectively. These are not generic fuel assemblies provided to multiple reactors.
4. Some fuel rod positions may be occupied by burnable poison rods or solid filler rods.

Table 12B2-3 BWR Fuel Assembly Characteristics

Fuel Class[1,5]	Vendor[4]	Array	Max. MTU	No of Fuel Rods	Max. Pitch (in)	Min. Rod Dia. (in)	Min. Clad Thick (in)	Max. Pellet Dia.(in)	Max. Active Length (in)[2]
4[5]	Ex/ANF	7 X 7	0.196	48	0.738	0.570	0.036	0.490	144.0
4	Ex/ANF	8 X 8	0.177	63	0.641	0.484	0.036	0.405	145.2
4	Ex/ANF	9 X 9	0.173	79	0.572	0.424	0.030	0.357	145.2
4	GE	7 X 7	0.199	49	0.738	0.570	0.036	0.488	144.0
4	GE	7 X 7	0.198	49	0.738	0.563	0.032	0.487	144.0
4	GE	8 X 8	0.173	60	0.640	0.484	0.032	0.410	145.2
4	GE	8 X 8	0.179	62	0.640	0.483	0.032	0.410	145.2
4	GE	8 X 8	0.186	63	0.640	0.493	0.034	0.416	144.0
5	Ex/ANF	8 X 8	0.180	62	0.641	0.484	0.036	0.405	150.0
5	Ex/ANF	9 X 9	0.167	74[3]	0.572	0.424	0.030	0.357	150.0
5[6]	Ex/ANF	9 X 9	0.178	79[3]	0.572	0.424	0.030	0.357	150.0
5	GE	7 X 7	0.198	49	0.738	0.563	0.032	0.487	144.0
5	GE	8 X 8	0.179	60	0.640	0.484	0.032	0.410	150.0
5	GE	8 X 8	0.185	62	0.640	0.483	0.032	0.410	150.0
5	GE	8 X 8	0.188	63	0.640	0.493	0.034	0.416	146.0
5	GE	9 X 9	0.186	74[3]	0.566	0.441	0.028	0.376	150.0
5	GE	9 X 9	0.198	79[3]	0.566	0.441	0.028	0.376	150.0

1. Maximum Initial Peak Planar Average Enrichment 4.0 wt % ^{235}U. All fuel rods are Zircaloy clad.
2. 150 inch active fuel length assemblies contain 6" natural uranium blankets on top and bottom.
3. Shortened active fuel length in some rods.
4. Vendor ID indicates the source of assembly base parameters, which are nominal, pre-irradiation values. Loading of assemblies meeting above limits is not restricted to the vendor(s) listed.
5. UMS Class 4 and 5 for BWR 2/3 fuel.
6. Assembly width including channel. Unchanneled or channeled assemblies may be loaded based on a maximum channel thickness of 120 mil.

Table 12B2-4 Minimum Cooling Time Versus Burnup/Initial Enrichment for PWR Fuel

Minimum Initial Enrichment wt % ^{235}U (E)	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]				30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]			
	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	5	5	5	5	7	7	5	7
2.1 ≤ E < 2.3	5	5	5	5	7	6	5	6
2.3 ≤ E < 2.5	5	5	5	5	6	6	5	6
2.5 ≤ E < 2.7	5	5	5	5	6	6	5	6
2.7 ≤ E < 2.9	5	5	5	5	6	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5	5	5	5

Minimum Initial Enrichment wt % ^{235}U (E)	35< Burnup ≤40 GWD/MTU Minimum Cooling Time [years]				40< Burnup ≤45 GWD/MTU Minimum Cooling Time [years]			
	14x14	15x15	16x16	17x17	14x14	15x15	16x16	17x17
1.9 ≤ E < 2.1	10	10	7	10	15	15	11	15
2.1 ≤ E < 2.3	9	9	7	9	14	13	10	13
2.3 ≤ E < 2.5	8	8	6	8	12	13	10	12
2.5 ≤ E < 2.7	8	8	6	8	11	13	10	12
2.7 ≤ E < 2.9	7	8	6	8	10	12	9	12
2.9 ≤ E < 3.1	7	8	6	8	9	12	9	11
3.1 ≤ E < 3.3	6	8	6	7	8	12	9	10
3.3 ≤ E < 3.5	6	8	6	7	8	12	9	10
3.5 ≤ E < 3.7	6	8	6	6	8	11	9	10
3.7 ≤ E ≤ 4.2	6	7	6	6	8	10	9	10

Table 12B2-5 Minimum Cooling Time Versus Burnup/Initial Enrichment for BWR Fuel

Minimum Initial Enrichment wt % ^{235}U (E)	Burnup ≤30 GWD/MTU Minimum Cooling Time [years]			30< Burnup ≤35 GWD/MTU Minimum Cooling Time [years]		
	7x7	8x8	9x9	7x7	8x8	9x9
1.9 ≤ E < 2.1	5	5	5	8	7	7
2.1 ≤ E < 2.3	5	5	5	6	6	6
2.3 ≤ E < 2.5	5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5	5
3.7 ≤ E ≤ 4.0	5	5	5	5	5	5

Minimum Initial Enrichment wt % ^{235}U (E)	35< Burnup ≤40 GWD/MTU Minimum Cooling Time [years]			40< Burnup ≤45 GWD/MTU Minimum Cooling Time [years]		
	7x7	8x8	9x9	7x7	8x8	9x9
1.9 ≤ E < 2.1	16	14	15	26	24	25
2.1 ≤ E < 2.3	13	12	12	23	21	22
2.3 ≤ E < 2.5	9	8	8	18	16	17
2.5 ≤ E < 2.7	8	7	7	15	14	14
2.7 ≤ E < 2.9	7	6	6	13	11	12
2.9 ≤ E < 3.1	6	6	6	11	10	10
3.1 ≤ E < 3.3	6	5	6	9	8	9
3.3 ≤ E < 3.5	6	5	6	8	7	8
3.5 ≤ E < 3.7	6	5	6	7	7	7
3.7 ≤ E ≤ 4.0	6	5	5	7	6	7

Table 12B2-6 Maine Yankee Site Specific Fuel Canister Loading Position Summary

Site Specific Spent Fuel Configurations[1]	Est. Number of Assemblies[2]	Canister Loading Position
STANDARD Fuel[3]	1,434	Any
Inserted Control Element Assembly (CEA)	168	Any
Inserted In-Core Instrument (ICI) Thimble	138	Any
Consolidated Fuel	2	Corner[4]
Fuel Rod Replaced by Rod Enriched to 1.95 wt %	3	Any
Fuel Rod Replaced by Stainless Steel Rod or Zircaloy Rod	18	Any
Fuel Rods Removed	10	Corner[4]
Variable Enrichment[6]	72	Any
Variable Enrichment and Axial Blanket[6]	68	Any
Burnable Poison Rod Replaced by Hollow Zircaloy Rod	80	Corner[4]
Damaged Fuel in MAINE YANKEE FUEL CAN	12	Corner[4]
Burnup between 45,000 and 50,000 MWD/MTU	90	Periphery[5]
MAINE YANKEE FUEL CAN	As Required	Corner[4]
UNCANNED DAMAGED FUEL	--	Periphery[5]
Inserted Startup Source	4	Corner[4]
Inserted Boronometer Source	1	Corner[4]
Inserted CEA Finger Tip or ICI String Segment	1	Corner[4]

1. All spent fuel, including that held in a Maine Yankee fuel can, must conform to the loading limits presented in Tables 12B2-8 and 12B2-9 for cool time.
2. The number of fuel assemblies in some categories may vary depending on future fuel inspections.
3. Includes fuel with inserted CEA and ICI thimbles and fuel with variable enrichment and axial blankets.
4. Corner positions are positions 3, 6, 19, and 22 in Figure 12B2-1. Corner positions are also periphery positions.
5. Periphery positions are positions 1, 2, 3, 6, 7, 12, 13, 18, 19, 22, 23, and 24 in Figure 12B2-1. Periphery positions include the corner positions.
6. Variably enriched fuel assemblies have a maximum burnup of less than 30,000 MWD/MTU and enrichments greater than 1.9 wt %. The minimum required cool time for these assemblies is 5 years.

Table 12B2-7 Maine Yankee Site Specific Fuel Limits

A. Allowable Contents

1. Combustion Engineering 14 x 14 PWR INTACT FUEL ASSEMBLIES meeting the specifications presented in Tables 12B2-1, 12B2-2 and 12B2-4.
2. PWR INTACT FUEL ASSEMBLIES may contain inserted Control Element Assemblies (CEA), In-Core Instrument (ICI) Thimbles or CEA Flow Plugs. Fuel assemblies with these components installed must be loaded in a Class 2 CANISTER and cannot be loaded in a Class 1 CANISTER.
3. PWR INTACT FUEL ASSEMBLIES with fuel rods replaced with stainless steel or Zircaloy rods or with Uranium oxide rods nominally enriched up to 1.95 wt %.
4. PWR INTACT FUEL ASSEMBLIES with fuel rods having variable enrichments with a maximum fuel rod enrichment up to 4.21 wt % ^{235}U and that also have a maximum planar average enrichment up to 3.99 wt % ^{235}U.
5. PWR INTACT FUEL ASSEMBLIES with annular axial end blankets. The axial end blanket enrichment may be up to 2.6 wt % ^{235}U.
6. PWR INTACT FUEL ASSEMBLIES with solid filler rods or burnable poison rods occupying up to 16 of 176 fuel rod positions.
7. PWR INTACT FUEL ASSEMBLIES with one or more grid spacers missing or damaged such that the unsupported length of the fuel rods does not exceed 60 inches or with end fitting damage, including damaged or missing hold-down springs as long as the assembly can be handled safely by normal means.

B. Allowable Contents requiring preferential loading based on shielding, criticality or thermal constraints. The preferential loading requirement for these fuel configurations is as described in Table 12B2-6.

1. PWR INTACT FUEL ASSEMBLIES with up to 176 fuel rods missing from the fuel assembly lattice.
2. PWR INTACT FUEL ASSEMBLIES with a burnup between 45,000 and 50,000 MWD/MTU.
3. PWR INTACT FUEL ASSEMBLIES with a burnable poison rod replaced by a hollow Zircaloy rod.

Table 12B2-7 Maine Yankee Site Specific Fuel Limits (continued)

4. PWR UNCANNED DAMAGED FUEL. UNCANNED DAMAGED FUEL must be loaded in a basket periphery fuel loading position. UNCANNED DAMAGED FUEL is limited to 24 fuel rods in each CANISTER.
5. INTACT FUEL ASSEMBLIES with a startup source in a center guide tube. The assembly must be loaded in a basket corner position and must be loaded in a Class 1 CANISTER. Only one (1) startup source may be loaded in any fuel assembly or any CANISTER. The CANISTER may not also contain the Boronometer source.
6. PWR INTACT FUEL ASSEMBLIES with CEA ends (finger tips) and/or ICI segment and/or Boronometer source, inserted in corner guide tube positions. The assembly must also have a CEA flow plug installed. The assembly must be loaded in a basket corner position and must be loaded in a Class 2 CANISTER.
7. INTACT FUEL ASSEMBLIES may be loaded in a MAINE YANKEE FUEL CAN.
8. FUEL enclosed in a MAINE YANKEE FUEL CAN. The MAINE YANKEE FUEL CAN can only be loaded in a Class 1 CANISTER. The contents that must be loaded in the MAINE YANKEE FUEL CAN are:
   a) PWR fuel assemblies with up to two INTACT or DAMAGED FUEL rods inserted in each fuel assembly guide tube or with up to two burnable poison rods inserted in each guide tube. The rods inserted in the guide tubes cannot be from a different fuel assembly. The maximum number of rods in the fuel assembly (fuel rods plus inserted rods, including burnable poison rods) is 176.
   b) A DAMAGED FUEL ASSEMBLY with up to 100% of the fuel rods classified as damaged and/or damaged or missing assembly hardware components. A DAMAGED FUEL ASSEMBLY cannot have an inserted CEA or other non-fuel component.
   c) Individual INTACT or DAMAGED FUEL rods in a rod type structure, which may be a guide tube, to maintain configuration control.
   d) FUEL DEBRIS consisting of fuel rods with exposed fuel pellets or individual intact or partial fuel pellets not contained in fuel rods.

Table 12B2-7 Maine Yankee Site Specific Fuel Limits (continued)

e) CONSOLIDATED FUEL lattice. Maximum contents is 289 fuel rods having a total lattice weight ≤ 2,100 pounds. A CONSOLIDATED FUEL lattice cannot have an inserted CEA or other non-fuel component.

f) HIGH BURNUP FUEL (45,000 to 50,000 MWD/MTU) if the total number of fuel rods classified as damaged in the CANISTER exceeds 24.

C. Unenriched fuel assemblies are not authorized for loading.

D. A canister preferentially loaded in accordance with Table 12B2-8 may only contain fuel assemblies selected from the same loading pattern..

Table 12B2-8 Loading Table for Maine Yankee CE 14 x 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable

	Burnup ≤ 30 GWD/MTU - Minimum Cool Time [years] for[1]				
Enrichment	**Standard[2]**	**Pref (0.958i)**	**Pref (0.958p)**	**Pref (1.05i)**	**Pref (1.05p)**
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time [years] for				
Enrichment	**Standard[2]**	**Pref (0.958i)**	**Pref (0.958p)**	**Pref (1.05i)**	**Pref (1.05p)**
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time [years] for				
Enrichment	**Standard[2]**	**Pref (0.958i)**	**Pref (0.958p)**	**Pref (1.05i)**	**Pref (1.05p)**
1.9 ≤ E < 2.1	7	7	6	15	5
2.1 ≤ E < 2.3	6	6	6	15	5
2.3 ≤ E < 2.5	6	6	5	14	5
2.5 ≤ E < 2.7	5	5	5	14	5
2.7 ≤ E < 2.9	5	5	5	14	5
2.9 ≤ E < 3.1	5	5	5	6	5
3.1 ≤ E < 3.3	5	5	5	6	5
3.3 ≤ E < 3.5	5	5	5	6	5
3.5 ≤ E < 3.7	5	5	5	6	5
3.7 ≤ E ≤ 4.2	5	5	5	6	5

1. Cool times for preferential loading of fuel assemblies with a decay heat of either 0.958 or 1.05 kw per assembly, loaded in either interior (i) or periphery (p) basket positions. All of the fuel assemblies in a canister must be selected using the same preferential loading pattern (Standard, 0.958 kW or 1.05 kW).
2. Fuel assemblies with cool times from 5 to 7 years must be preferentially loaded based on cool time, with fuel with the shortest cool time in the basket interior, in accordance with Section B2.1.2.

Table 12B2-8 Loading Table for Maine Yankee CE 14 x 14 Fuel with No Non-Fuel Material – Required Cool Time in Years Before Assembly is Acceptable (continued)

Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time [years] for[1]				
	Standard[2]	**Pref(0.958i)**	**Pref(0.958p)**	**Pref(1.05i)**	**Pref(1.05p)**
1.9 ≤ E < 2.1	11	20	7	Not Allowed	6
2.1 ≤ E < 2.3	9	15	7	Not Allowed	6
2.3 ≤ E < 2.5	8	15	6	Not Allowed	6
2.5 ≤ E < 2.7	8	15	6	Not Allowed	6
2.7 ≤ E < 2.9	8	14	6	Not Allowed	6
2.9 ≤ E < 3.1	8	14	6	Not Allowed	6
3.1 ≤ E < 3.3	7	14	6	Not Allowed	5
3.3 ≤ E < 3.5	6	14	6	Not Allowed	5
3.5 ≤ E < 3.7	6	13	6	Not Allowed	5
3.7 ≤ E ≤ 4.2	6	13	6	Not Allowed	5
Enrichment	**45 < Burnup ≤ 50 GWD/MTU - Minimum Cool Time [years] for**				
	Standard	**Pref(0.958i)**	**Pref(0.958p)**	**Pref(1.05i)**	**Pref(1.05p)**
1.9 ≤ E < 2.1	Not Allowed	Not Allowed	8	Not Allowed	7
2.1 ≤ E < 2.3	Not Allowed	Not Allowed	8	Not Allowed	7
2.3 ≤ E < 2.5	Not Allowed	Not Allowed	8	Not Allowed	7
2.5 ≤ E < 2.7	Not Allowed	Not Allowed	8	Not Allowed	7
2.7 ≤ E < 2.9	Not Allowed	Not Allowed	8	Not Allowed	7
2.9 ≤ E < 3.1	Not Allowed	Not Allowed	8	Not Allowed	7
3.1 ≤ E < 3.3	Not Allowed	Not Allowed	7	Not Allowed	7
3.3 ≤ E < 3.5	Not Allowed	Not Allowed	7	Not Allowed	6
3.5 ≤ E < 3.7	Not Allowed	Not Allowed	7	Not Allowed	6
3.7 ≤ E ≤ 4.2	Not Allowed	Not Allowed	7	Not Allowed	6

1. Cool times for preferential loading of fuel assemblies with a decay heat of either 0.958 or 1.05 kw per assembly, loaded in either interior (i) or periphery (p) basket positions. All of the fuel assemblies in a canister must be selected using the same preferential loading pattern.

2. Fuel assemblies with cool times from 5 to 7 years must be preferentially loaded based on cool time, with fuel with the shortest cool time in the basket interior, in accordance with Section B2.1.2.

Table 12B2-9 Loading Table for Maine Yankee CE 14 x 14 Fuel Containing CEA Cooled to Indicated Time

Enrichment	≤ 30 GWD/MTU Burnup - Minimum Cool Time in Years for				
	No CEA (Class 1)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	30 < Burnup ≤ 35 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 1)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	5	5	5	5	5
2.1 ≤ E < 2.3	5	5	5	5	5
2.3 ≤ E < 2.5	5	5	5	5	5
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	35 < Burnup ≤ 40 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 1)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	7	7	7	7	7
2.1 ≤ E < 2.3	6	6	6	6	6
2.3 ≤ E < 2.5	6	6	6	6	6
2.5 ≤ E < 2.7	5	5	5	5	5
2.7 ≤ E < 2.9	5	5	5	5	5
2.9 ≤ E < 3.1	5	5	5	5	5
3.1 ≤ E < 3.3	5	5	5	5	5
3.3 ≤ E < 3.5	5	5	5	5	5
3.5 ≤ E < 3.7	5	5	5	5	5
3.7 ≤ E ≤ 4.2	5	5	5	5	5
Enrichment	40 < Burnup ≤ 45 GWD/MTU - Minimum Cool Time in Years for				
	No CEA (Class 1)	5 Year CEA	10 Year CEA	15 Year CEA	20 Year CEA
1.9 ≤ E < 2.1	11	11	11	11	11
2.1 ≤ E < 2.3	9	9	9	9	9
2.3 ≤ E < 2.5	8	8	8	8	8
2.5 ≤ E < 2.7	8	8	8	8	8
2.7 ≤ E < 2.9	8	8	8	8	8
2.9 ≤ E < 3.1	8	8	8	8	8
3.1 ≤ E < 3.3	7	7	8	8	8
3.3 ≤ E < 3.5	6	6	7	7	7
3.5 ≤ E < 3.7	6	6	6	6	6
3.7 ≤ E ≤ 4.2	6	6	6	6	6